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COUPLED 42-GROUP NEUTRON AND 21-GROUP GAMMA RAY CROSS SECTION SETS FOR FUSION REACTOR CALCULATIONS

April 1980

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(Received March 8, 1980)

Two sets of coupled neutron and gamma ray multigroup cross sections for 40 nuclides involved in fusion reactor nuclear design have been developed and are used in the Japan Atomic Energy Research Institute. Both the cross section sets have the same 42 neutron and 21 gamma ray energy group structure and include up to P_5 Legendre scattering terms. They also include the same 40 nuclides. However, the data and derivation methods used in obtaining the two sets differ largely.

Modifications of the multigroup cross section processing codes were made in the course of development of the cross section sets. The results of calculations with both sets are compared to show discrepancies and problems. Possible means of rectifying them are considered.

Keywords; Coupled Cross Section Set, Neutron, Gamma Ray, Fusion Reactor Multigroup Energy Structure, Nuclear Design, Nuclides, Scattering Terms, Nuclear Data, Data Processing

核融合炉用４２群中性子－２１群ガンマ線結合断面積セット

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（１９８０年3月8日受理）

核融合炉核設計に関連した４０核種についての中性子とガンマ線の結合断面積セットを二組作成し使用してきた。どちらのセットも，同じ４２群中性子－２１群ガンマ線のエネルギー群構造を有し，P_5までのルジャンドル散乱項を含んでいる。また，両者はともに同じ４０核種を収納している。しかしながら，二組のセットを作成する際に用いた核データと計算法は大きく異なり，これらについて明らかにする。

これらのセットを作成する際に行った多群断面積作成計算コードの改良についても簡単に述べる。二組のセットを用いて行った計算結果の比較より，不一致と問題点を明らかにし，これらの問題の改善法を提案した。

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1. INTRODUCTION

Several coupled neutron and gamma-ray cross section libraries have been made[(1)~(4)] for shielding and fusion reactor neutronics calculations. The DLC libraries[(1)~(3)] made in the Oak Ridge National Laboratory (ORNL) are widely used for fusion reactor calculations as they are distributed through the Radiation Shielding Information Center (RSIC) in ORNL. It is convenient to use those libraries when they are available and applicable. However, the data and methods used in their preparation is not always sufficiently known and the nuclides and energy structure required for a particular task may not be satisfied by them.

Two sets of libraries for 40 nuclides of interest in fusion reactor nuclear design have been developed in the Japan Atomic Energy Research Institute (JAERI). They are called GICX40 and GICX40V4 and are extensively used for neutronics calculations related to fusion reactors in JAERI and elsewhere in Japan. Both sets have the same 42-neutron, 21-gamma-ray energy group structure with up to P_5 Legendre scattering terms. Both include cross sections for the same 40 nuclides as listed in Tables 2.1 and 2.4 but the nuclear data and method of processing used in their derivation are different. The first set GICX40 is based on the ENDF/B-III and IV nuclear data files[(5)] for neutron cross sections and on the POPOP4 Library[(6)] for gamma-ray production cross sections. The second set GICX40V4 is based entirely on the data from the ENDF/B-IV. For the processing of the former set, the RADHEAT code system[(7)] developed in JAERI was mainly used and for the processing of the latter, the NJOY system[(8)] made in the Los Alamos Scientific Laboratory was used. The GAMLEG-JR code[(9)] was used to obtain gamma-ray transport cross sections for both sets. In addition to coupled cross sections, neutron and gamma-ray kerma factors are included in the sets.

This report is produced to inform the present status of the two sets and to summarize the problems concerning the employed nuclear data and processing methods. It is hoped to improve further the cross section sets based on the responses for this report.

2. COUPLED CROSS SECTION SETS

2.1 GICX40 Set

The original portion of the coupled neutron and gamma-ray cross section set GICX40 consisting of neutron transport cross sections for 8 nuclides was derived in 1973 by T. Hiraoka.(10) Gamma-ray production and transport cross sections and kerma factors were then added and combined to enable nuclear heating calculations. The number of nuclides included increased to 40 as they came to be necessary in the course of fusion reactor related neutronics studies from 1974 to 1977. The nuclear data used were constantly updated as new data such as ENDF/B-IV(5) became available and any defective data found were replaced by appropriate ones. The main characteristics of the GICX40 set completed in 1977 are as follows:

(1) The cross sections for the 40 nuclides of interest in fusion reactor nuclear designs are included in the order shown in Table 2.1.

(2) Among the 40 nuclides, not only the candidate nuclides in the materials of blanket, shield, magnet, etc. of pure fusion reactors but also 5 fissile nuclides ^{232}Th, ^{235}U, ^{238}U, ^{237}Np and ^{239}Pu are included to enable the evaluations of fusion-fission hybrid reactors.

(3) It consists of 42-neutron and 21-gamma-ray energy group structures shown in Tables 2.2 and 2.3.

(4) It is in the format of group independent type cross section input for the ANISN code.(11) A cross section table for a P_0-component of a nuclide is shown in Fig. 2.1. Neutron and gamma-ray kerma factors are placed in the first table position and the total cross section σ_t in table position four.

(5) Anisotropy of elastic neutrons are accounted for by the up to P_5 Legendre coefficients for each nuclide.

(6) Neutron cross section data are obtained from the ENDF/B-III and IV(5) and gamma-ray production data from the POPOP4 Library.(6) The MAT numbers and ID numbers for the data in the ENDF/B and POPOP4 Libraries which are included in the set are shown in Table 2.1. Gamma-ray transport cross sections and kerma factors are calculated by the GAMLEG-JR code.(9)

(7) All neutron cross sections are the values at 300°K and all except those for ^{235}U and ^{238}U are for infinite dilution (i.e. for $\sigma_p = 10^8$). The "background" cross sections (or nonresonance isotope potential scattering

cross sections per absorber atom[12] σ_p) are 900.0 and 0.6 for ^{235}U and ^{238}U, respectively. Neutron cross sections are always weighted with a 1/E function.

(8) Although not included in GICX40 itself, tritium producing cross sections, displacement damage cross sections[13] and several libraries[14] for induced activation and dose rate calculations are prepared in the 42-group structure of Table 2.2.

(9) Reaction wise (partial) cross sections for the sensitivity analysis code SWANLAKE[15] are also prepared in the 42-neutron and 21-gamma-ray energy structure.

(10) A cross section editing program was developed to select any number of specified nuclides from GICX40 and prepare a concise microscopic or macroscopic cross section set including only those nuclides required in a particular calculation. This program can also rearrange and convert the group independent type cross sections into material wise type used in ANISN.[11] It can also invert cross sections for the use in adjoint calculations.

2.2 GICX40V4 Set

Neutron transport cross sections, gamma-ray production cross sections and neutron kerma factors of the new GICX40V4 set are calculated utilizing the full capability of the NJOY system.[8] Only the nuclear data in the ENDF/B-IV[5] are used. In the process of generating the new set, GROUPR module of the NJOY was modified to allow automatic generation of cross sections without any pre-examination of the content of the ENDF/B-IV file. All the gamma-ray production data in the file were included in GICX40V4 except for the four capture gamma-ray data given by the footnote of Table 2.4.* This table gives the gamma-ray production data included in each of the cross section of the 40 nuclides listed. It also gives the qualitative description of the status of calculated neutron kerma factors of each nuclide as they are compared in the HEATR module of the NJOY[8] with the upper and lower bound values obtained from kinematic calculations.

The capture gamma-ray data of Molybdenum was excluded because its

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* Also whenever gamma-production data for the same reaction appear in both Files 12 and 13, the data in File 12 is adopted.

value became anomalously large at around 1 $\sim$ 2 keV. The capture gamma-ray data for ^{235}U, ^{238}U and ^{239}Pu were not included simply because of input data error and their effects are considered too small to deserve recalculations.

The status of the calculated neutron kerma factors are described only qualitatively in Table 2.4. Similar energy-balance checks have been carried out for 37 nonfissionable materials from ENDF/B-V[16] by R.E. MacFarlane.[17] Table 2.4 shows that kerma factors for light nuclides ($Z \leq 9$) and ^{235}U, ^{239}Pu, are in good shape, medium weight nuclides ($11 \leq Z \leq 20$) not too bad, but the rest of them generally poor. The components nuclides of stainless steel (Cr, Fe, Ni, Mo), copper and heavy metals (W, Nb) should be treated with caution. These trends are much the same for ENDF/B-IV and V data.

Gamma-ray transport cross sections and kerma factors are calculated by the GAMLEG-JR code.[9] All cross sections and kerma factors for both neutron and gamma-ray are combined and converted to form the group independent type coupled cross section set GICX40V4. This new set also has up to P_5 Legendre coefficients in which are included not only the anisotropy of elastic neutrons but also of nonelastic neutrons whenever such data exist in ENDF/B-IV. In the NJOY calculations, neutron cross sections were always weighted with a neutron spectrum in a typical fusion reactor blanket assembly. Other details of the new set such as the energy group structure, cross section table format, the "background" cross section values are all same as those for the GICX40 set described in 2.1.

Table 2.1 Nuclear Data Identification Numbers Used in GICX40 set

No.	Nuclide	MATNO in ENDF/B-3 (4*)	POPOP4 ID Number** for Gamma-ray Producing reactions (n, γ)	(n,n'γ)	others	MT NO for ANISN input P0～P5
1	^{6}Li	1115	30101	40301		1～ 6
2	^{7}Li	1116	30101	40301		7～ 12
3	^{12}C	1165	60101	60301		13～ 18
4	^{16}O	1134	70301	86301	80201, 860601	19～ 24
5	^{4}He	1088	30301	40301		25～ 30
6	Nb	1164	404101	400301		31～ 36
7	Mo	1111	420101	400301		37～ 42
8	Cr	1121	240101	240301		43～ 48
9	Ni	1123	280101	280301		49～ 54
10	Fe	1180	260101	260301		55～ 60
11	^{1}H	1148	10101	40301		61～ 66
12	^{2}H	1120	13001	40301		67～ 72
13	^{3}He	1146	30101	40301		73～ 78
14	^{9}Be	1154	40102	40301		79～ 84
15	^{10}B	1155	40102	40301	50204	85～ 90
16	^{14}N	1133	70102	70301	74401	91～ 96
17	^{27}Al	1135	130103	180301		97～102
18	V	1017	240101	240301		103～106
19	Cu	1087	290104	280301		109～114
20	Pb	1136	820102	820301		115～120
21	^{232}Th	1117	928109	928301	925801	121～126
22	^{235}U	1157	925101	925301	925801	127～132
23	^{239}Pu	1159	928109	928301	925801	133～138
24	^{237}Np	1263 *	937101	928301	925801	139～144
25	Mg	1014	925101	925301		145～150
26	K	1150	190101	190301	190701	151～156
27	Ca	1152	200101	200301	200701	157～162
28	^{11}B	1160 *	60101	60301		163～168
29	Cl	1149 *	170101	170301		169～174
30	Na	1156 *	110101	113301		175～180
31	Cd	1281 *	480101	—		181～186
32	Ta	1285 *	730101	740301		187～192
33	^{182}W	1128 *	730103	740301		193～198
34	^{183}W	1129 *	730103	740301		199～204
35	^{184}W	1130 *	730103	740301		205～210
36	^{186}W	1131 *	740103	740301		211～216
37	F	1277 *	90102	86302		217～222
38	^{238}U	1262 *	928109	928301	925801	223～228
39	Si	1194 *	140105	140301		229～234
40	Ti	1286 *	220102	220302		235～240

* The data in ENDF/B-4 are used.

** ID number used in POPOP4 library has five or six digits. They indicate the following meaning

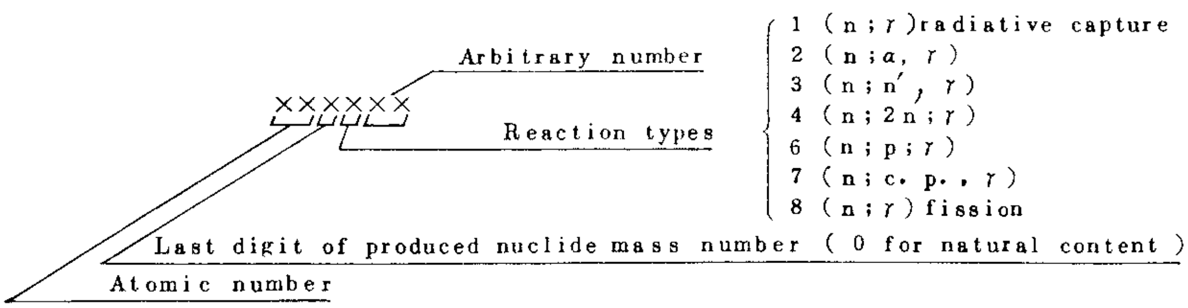

Table 2.2 42-Group Neutron Energy Group Structure

Group	Energy Limits		Mid-Point Energy	
1	15.000 - 13.720	MeV	14.360	MeV
2	13.720 - 12.549		13.135	
3	12.549 - 11.478		12.014	
4	11.478 - 10.500		10.989	
5	10.500 - 9.314		9.907	
6	9.314 - 8.261		8.788	
7	8.261 - 7.328		7.795	
8	7.328 - 6.500		6.914	
9	6.500 - 5.757		6.129	
10	5.757 - 5.099		5.428	
11	5.099 - 4.516		4.808	
12	4.516 - 4.000		4.258	
13	4.000 - 3.162		3.581	
14	3.162 - 2.500		2.831	
15	2.500 - 1.871		2.186	
16	1.871 - 1.400		1.636	
17	1.400 - 1.058		1.229	
18	1.058 - 0.800		0.929	
19	0.800 - 0.566		0.683	
20	0.566 - 0.400		0.483	
21	0.400 - 0.283		0.342	
22	0.283 - 0.200		0.242	
23	0.200 - 0.141		0.171	
24	0.141 - 0.100		0.121	
25	100.0 - 46.5	KeV	73.25	KeV
26	46.5 - 21.5		34.0	
27	21.5 - 10.0		15.75	
28	10.0 - 4.65		7.325	
29	4.65 - 2.15		3.40	
30	2.15 - 1.00		1.575	
31	1.00 - 0.465		0.733	
32	0.465 - 0.215		0.340	
33	0.215 - 0.100		0.158	
34	100.0 - 46.5	eV	73.25	eV
35	46.5 - 21.5		34.0	
36	21.5 - 10.0		15.75	
37	10.0 - 4.65		7.325	
38	4.65 - 2.15		3.40	
39	2.15 - 1.00		1.58	
40	1.00 - 0.465		0.733	
41	0.465 - 0.215		0.340	
42	0.215 - 0.001		0.108	

Table 2.3 21-Group Gamma-Ray Energy Group Structure

Group	Energy Limits (MeV)	Mid-Point Energy (MeV)
1	14.0 - 12.0	13.0
2	12.0 - 10.0	11.0
3	10.0 - 8.0	9.0
4	8.0 - 7.5	7.75
5	7.5 - 7.0	7.25
6	7.0 - 6.5	6.75
7	6.5 - 6.0	6.25
8	6.0 - 5.5	5.75
9	5.5 - 5.0	5.25
10	5.0 - 4.5	4.75
11	4.5 - 4.0	4.25
12	4.0 - 3.5	3.75
13	3.5 - 3.0	3.25
14	3.0 - 2.5	2.75
15	2.5 - 2.0	2.25
16	2.0 - 1.5	1.75
17	1.5 - 1.0	1.25
18	1.0 - 0.4	0.7
19	0.4 - 0.2	0.3
20	0.2 - 0.1	0.15
21	0.1 - 0.01	0.055

Table 2.4 Gamma-Ray Production Data and Kerma Factor Status of GICX40V4 Set

No.	Nuclide	MATNO in ENDF/B-IV	Gamma-Ray Production Data File No., MF (Reaction Type No., MT)	Status of Neutron Kerma Factor
1	^{6}Li	1271	12(52, 102)	O.K.
2	^{7}Li	1272	12(51, 102)	O.K.
3	^{12}C	1274	12(102), 13(51)	O.K.
4	^{16}O	1276	12(102), 13(4,22,103,107)	O.K.
5	^{4}He	1270	none	O.K.
6	^{93}Nb	1189	12(102), 13(3)	too low below 6 MeV negative below 900 keV
7	Mo	1287	12(102)*, 13(3)	large negative values
8	Cr	1191	12(102), 13(3)	partially negative
9	Fe	1192	12(3,51,52,102)	too high below 1.2 keV high or low up to 4 MeV
10	Ni	1190	12(102), 13(3)	too low below 1.6 MeV negative below 10 keV
11	^{1}H	1269	12(102)	O.K.
12	^{2}H	1120	12(102)	O.K.
13	^{3}He	1146	none	O.K.
14	^{9}Be	1289	12(102, 741)	too low below 8.8 keV negative below 5 eV
15	^{10}B	1273	12(781), 13(4, 103)	O.K.
16	^{14}N	1275	12(102),13(4,103,104,105,107)	O.K.
17	^{27}Al	1193	12(102), 13(4,28,103)	almost O.K.
18	V	1196	12(3,16,22,28,102)	too low at 237 keV - 2.4 MeV too high above 2.4 MeV
19	Cu	1295	12(102), 13(3)	mostly too high
20	Pb	1288	12(3)	mostly too low
21	^{232}Th	1296	none	too high at every energy
22	^{235}U	1261	12(4,8,102†), 13(3)	O.K.
23	^{239}Pu	1264	12(4,8,102†), 13(3)	O.K.
24	^{237}Np	1263	none	too high at every energy
25	Mg	1280	12(102)	a little too high above 26 keV
26	K	1150	12(51-54,91,102,103,107) 13(55, 56)	too high at every energy
27	Ca	1195	12(4,22,28,102,103,107)	almost O.K.
28	^{11}B	1160	none	mostly too high
29	Cl	1149	12(51-61, 91, 102), 13(3)	too high below 3.2 MeV
30	^{23}Na	1156	12(51-68, 91, 102)	mostly too high
31	Cd	1281	none	too high at every energy
32	^{181}Ta	1285	12(102), 13(3)	mostly too high
33	^{182}W	1128	12(102), 13(4)	too high at 150 keV - 620 keV too low above 780 keV
34	^{183}W	1129	12(102), 13(4)	too low above 825 keV
35	^{184}W	1130	12(102), 13(4)	negative above 1.2 MeV
36	^{186}W	1131	12(102), 13(4)	negative above 1.5 MeV
37	F	1277	13(3, 4, 102, 107)	almost O.K.
38	^{238}U	1262	12(8, 102†), 13(3)	too high below 95 keV
39	Si	1194	12(102),13(4,22,28,103,107)	almost O.K.
40	Ti	1286	12(102), 13(3)	too high below 95 keV

* Capture gamma-ray data of Mo are excluded because it is too large.

† Capture gamma-ray data of ^{235}U, ^{239}Pu and ^{238}U are not included.

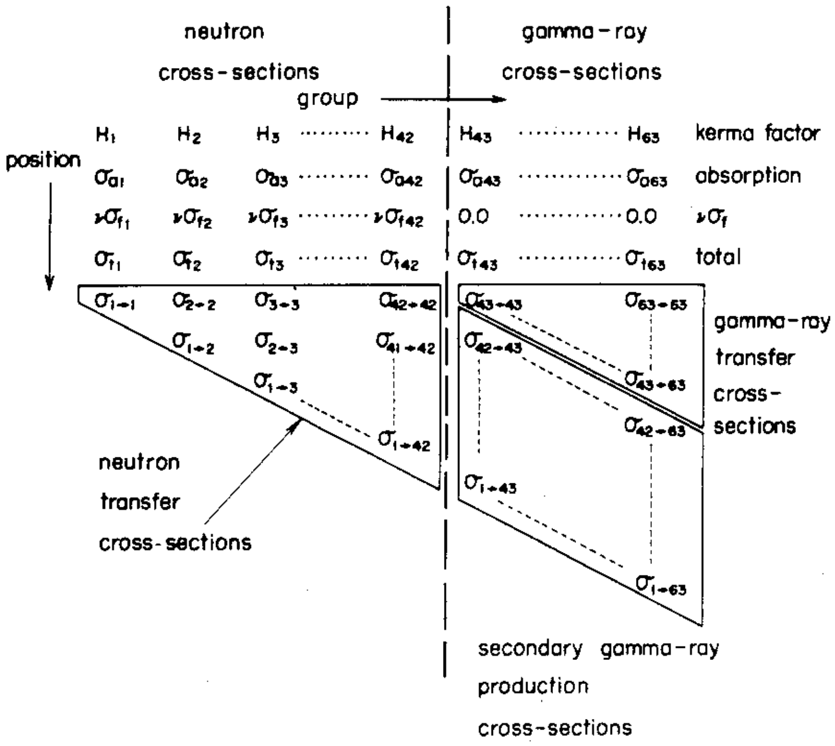

Fig. 2.1 Table of 42-Group Neutron and 21-Group Gamma-Ray Coupled Cross Sections

3. CALCULATIONAL METHODS

3.1 Coupled Cross Section Calculations Based on RADHEAT

The coupled cross section set GICX40 was produced by the code system shown in Fig. 3.1. This code system is based on the code system RADHEAT[7] developed in JAERI on 1974 for the radiation heating analysis of a nuclear reactor. The main differences between the original RADHEAT system and the one shown in Fig. 3.1 are as follows; the use of the SPTG4Z code[18] for neutron cross section calculations from the ENDF/B-IV data,[5] the use of the MACK code [19] for neutron kerma factor calculations and the processing of the cross section set of a number of nuclides at a time instead of one by one.

Neutron transport cross sections were processed by the codes SUPERTOG-JR[20] and SPTG4Z[18] from the nuclear data files ENDF/B-III and IV, respectively. Neutron kerma factors were calculated by the MACK code[19] using the ENDF/B-IV data. The original MACK code was revised to overcome the trouble encountered in the processing of inelastic level scattering data of ^{182}W and also to calculate and include fission energy deposition into the neutron kerma factor.

Gamma-ray production cross sections were calculated using the neutron cross sections obtained by the SUPERTOG-JR and SPTG4Z calculations and the POPOP4 Library data.[6] Gamma-ray transport cross sections and kerma factors are calculated using the GAMLEG-JR code.[9]

All the cross sections and kerma factors for neutron and gamma-ray are combined by the COUPLING code into the coupled cross section set of the form shown in Fig. 2.1. The nuclide wise coupled cross section set is then converted to the group independent type for the use in ANISN,[11] DOT-3.5[21] and MORSE[22] calculations.

3.2 Coupled Cross Section Calculations Based on NJOY

The coupled cross section set GICX40V4 was processed mainly by the nuclear data processing system NJOY[8] using the ENDF/B-IV data.[5] This NJOY system was partly implemented to the computer in JAERI by D.W. Muir of the Los Alamos Scientific Laboratory in October 1977 and was fully implemented later in 1978.

The following modifications were made in the modules of the NJOY system to simplify the input data preparation before it was used to process GICX40V4 in 1979:

The GROUPR module was modified to allow automatic generation of partial cross sections including all gamma-ray production cross sections. When all type of partial cross section numbers (MT) are given as the input for the processing of a nuclide, the modified GROUPR selects and processes only the reaction cross sections with the data available for that nuclide in the ENDF/B-IV. The execution will not be terminated even if the input specifies reactions without the data in the file.

The UNRESR module was also revised not to terminate even if unresolved parameters are absent in the file and requested by the input.

The DTFR module was revised to give ANISN[11] type cross section matrix instead of the DTF type.[23] That is $(2\ell + 1)$ factors were multiplied to P_ℓ scattering terms and absorption cross section σ_a is placed in the table position one.

The NJOY code was used to produce neutron cross sections, kerma factors and gamma-ray production cross sections from the ENDF/B-IV. As for the calculation of gamma-ray transport (interaction) cross sections and kerma factors, the GAMINR module of the NJOY system could not be used because the required ENDF/B tape including the photon interaction data was not available. The GAMLEG-JR code[9] was used for the calculation of gamma-ray transport cross sections and kerma factors. All the cross sections and kerma factors were then combined and converted to the group independent form for the use in radiation transport calculations.

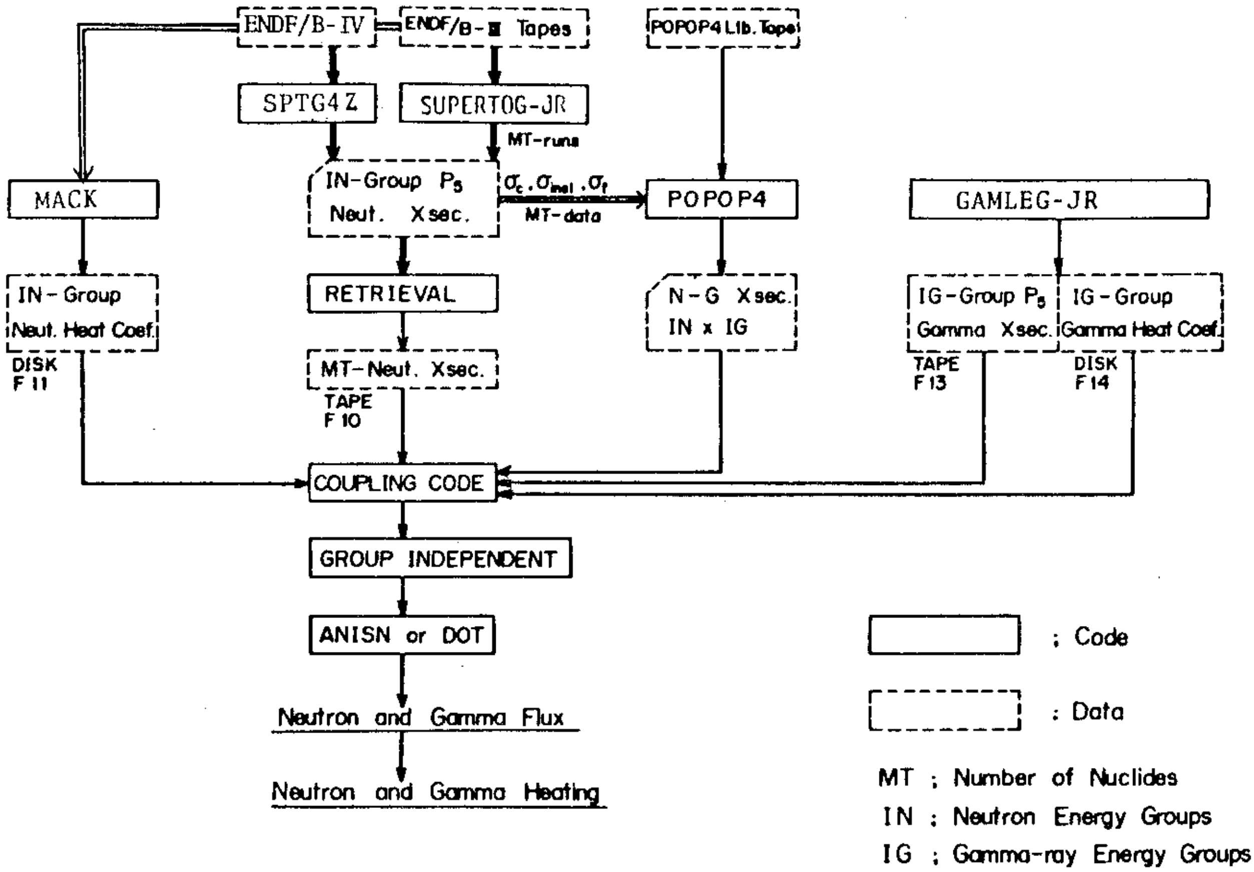

Fig. 3.1 Nuclear Heating Calculation Code System Based on RADHEAT

4. COMPARISON OF CALCULATED RESULTS

Tritium breeding reaction rates and nuclear heating rates integrated over regions of two fusion reactor models are calculated using both the GICX40 and GICX40V4 sets. The results of calculations using two sets are compared in this section.

4.1 Comparison for the Benchmark Blanket Model

For the comparison of the two coupled cross section sets, the benchmark blanket model used in the comparison[(24),(25)] of neutronics calculational methods is also used. The calculated model described in Ref. 24 is shown as Fig. 4.1. In the two calculations using the respective sets, the $P_5 - S_8$ approximation is used in the ANISN[(11)] calculations and calculational conditions are identical except for the cross section set used.

Table 4.1 shows the ^{6}Li(n,α)t and ^{7}Li(n,n'α)t reactions (denoted by T_6 and T_7) integrated over each lithium region of the model and their sums. Although region integrated values of both reactions differ as much as 20% when different cross section sets are used, the sum over the whole model agree well with only 1% difference. When GICX40V4 is used in place of GICX40, low energy neutrons (causing the ^{6}Li(n,α)t reactions) seem to be reflected more by the graphite reflector and hence less low energy neutrons teach the region 10. On the other hand, more high energy neutrons (causing the ^{7}Li(n,n'α)t reactions) reach region 10 when GICX40V4 is used. Both of these results can be interpreted as due to the exact treatment of nonelastic neutron angular distribution by the NJOY code[(17)] in the processing of the GICX40V4 set. The angular distribution was considered to be isotropic by SUPERTOG-JR[(20)] when the GICX40 set was produced. Compared with the recommended values of T_6 and T_7 in Ref. 25 of 0.911 and 0.528, respectively, the corresponding values in Table 4.1 for both sets are considerably (3∼9%) large.

Neutron and gamma-ray heating rate integrated over each region and their respective sum over the whole model are shown in Table 4.2. Neutron heating in the regions 3 and 5 became negative in case of GICX40V4 because of negative kerma factors for Niobium (see Table 2.4). Except for these two regions, calculated neutron heating is in good agreement for the two sets with the difference being less than 2%. Region integrated gamma-ray heating rate by the GICX40V4 set is always greater than the case using the GICX40 by more than 28%. This is caused by the improper gamma-ray production data in the ENDF/B-IV data as described in 2.2. In order to have better agreement, the elimination of redundant data and renormalization of the data for gamma-ray production are required. An automatic adjustment of gamma-ray production data by revising the processing codes is being considered.

Table 4.1 Comparison of tritium producing reactions in the Benchmark Blanket

^{6}Li(n,α)t	GICX40	GICX40V4	V4/40	^{7}Li(n,n'α)t	GICX40	GICX40V4	V4/40
$T_6(4)$	0.047	0.050	1.06	$T_7(4)$	0.078	0.081	1.04
$T_6(6)$	0.289	0.304	1.05	$T_7(6)$	0.301	0.305	1.02
$T_6(7)$	0.239	0.252	1.05	$T_7(7)$	0.133	0.132	1.00
$T_6(8)$	0.298	0.291	0.98	$T_7(8)$	0.057	0.057	1.00
$T_6(10)$	0.064	0.051	0.80	$T_7(10)$	0.001	0.001	1.20
T_6	0.936	0.948	1.01	T_7	0.570	0.577	1.01
Total = $T_6 + T_7$					1.506	1.524	1.01

$T_6(J)$: ^{6}Li(n,α)t reactions per source neutron in region J.

$T_7(J)$: ^{7}Li(n,n'α)t reactions per source neutron in region J.

$$T_6 = \sum_J T_6(J), \quad T_7 = \sum_J T_7(J)$$

Table 4.2 Comparison of neutron and gamma heating in the Benchmark Blanket

(MeV/source neutron)

neutron heating	GICX40	GICX40V4	V4/40	gamma heating	GICX40	GICX40V4	V4/40
$H_n(3)$	0.064	-0.005		$H_\gamma(3)$	0.264	0.358	1.355
$H_n(4)$	1.208	1.197	0.991	$H_\gamma(4)$	0.166	0.215	1.295
$H_n(5)$	0.048	-0.016		$H_\gamma(5)$	0.257	0.343	1.334
$H_n(6)$	5.115	5.058	0.989	$H_\gamma(6)$	0.940	1.206	1.283
$H_n(7)$	2.808	2.794	0.995	$H_\gamma(7)$	0.633	0.953	1.505
$H_n(8)$	2.110	2.102	0.996	$H_\gamma(8)$	0.387	0.784	2.025
$H_n(9)$	0.417	0.425	1.020	$H_\gamma(9)$	0.308	1.262	4.100
$H_n(10)$	0.256	0.257	1.005	$H_\gamma(10)$	0.013	0.076	6.056
H_n	12.024	11.812	0.982	H_γ	2.998	5.195	1.733
Total heating = $H_n + H_\gamma$					15.022	17.007	1.132

$H_n = \mathrm{Sum}_J\, H_n(J)$, $\quad H = \mathrm{Sum}_J\, H_\gamma(J)$

4.2 Comparison for the INTOR-J Outboard Model

A calculational model for the outboard blanket, shield and toroidal field coil of the JAERI Proposal for International Tokamak Reactor (INTOR-J)[26] is shown in Fig. 4.2. In the model, lithium oxide (Li_2O) is loaded in the regions 4 and 5 in the form of pebbles and blocks for tritium production. Hence these regions are called Li_2O Pebble and Li_2O Block regions.

The tritium producing reactions in the Li_2O Pebble and Block regions are shown in Table 4.3. When the GICX40 set is replaced by the GICX40V4 set, more of low energy neutrons (causing the $^6Li(n,\alpha)t$ reactions) seems to be reflected back to the Pebble region. However, the larger penetration of high energy neutrons seen in Table 4.1 by the replacement cannot be observed from this table.

Table 4.4 shows region integrated neutron and gamma-ray heating in the INTOR-J Outboard Model. The neutron heating values calculated by the GICX40V4 set become negative or take smaller values compared with those by the GICX40 set. This is mainly due to the negative kerma factors of Mo, Cr and Ni (c.f. Table 2.4). The effect of the large negative kerma factor of Mo is particularly large.

The gamma-ray heating values calculated by the GICX40V4 set are always greater than those calculated by the GICX40 set. The difference becomes greater at regions farther away from the plasma where capture gamma-rays caused by low energy neutron capture become dominant. This indicates the special need for the proper selection of gamma-ray production data for low energy neutrons. However, without any reliable experimental bases, it is difficult at present to evaluate the absolute magnitude of calculated errors in the gamma-heating for either calculations.

In spite of the large differences in the region integrated values between the two sets seen in Tables 4.2 and 4.4, total heating over the whole model agrees within 14%. This is due to the compensation of the excess gamma-productions by the negative kerma factors conducted in the HEATR module of the NJOY system.[17]

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* The model in Fig. 4.2 does not exactly correspond to the one considered in Ref. 26. However the slight difference is of no consequence as a model for the cross section sets comparison.

Table 4.3 Comparison of tritium producing reactions in the INTOR-J Blanket

(reactions/source neutron)

Region integrated reaction rate	GICX40	GICX40V4	V4/40
$^{6}Li(n,\alpha)t$ in Pebble Region	0.184	0.212	1.15
Block Region	0.596	0.614	1.03
$^{7}Li(n,n'\alpha)t$ in Pebble Region	0.072	0.069	0.96
Block Region	0.111	0.105	0.95
Total	0.963	1.000	1.04

Table 4.4 Comparison of neutron and gamma-ray heating in the INTOR-J Outboard Model

(MeV/source neutron)

neutron heating	GICX40	GICX40V4	V4/40	gamma heating	GICX40	GICX40V4	V4/40
$H_n(2)$	8.69-2*	-2.86-1		$H_\gamma(2)$	9.50-1	1.22	1.28
$H_n(3)$	9.96-1	7.58-1	0.761	$H_\gamma(3)$	1.86	2.33	1.25
$H_n(4)$	2.51	2.54	1.01	$H_\gamma(4)$	9.51-1	1.00	1.06
$H_n(5)$	6.48	6.42	0.99	$H_\gamma(5)$	2.12	2.50	1.18
$H_n(6)$	1.96-2	7.74-3	0.40	$H_\gamma(6)$	2.27-1	5.63-1	2.48
$H_n(8)$	9.13-2	5.54-2	0.60	$H_\gamma(8)$	2.91-1	1.06	3.63
$H_n(10)$	4.48-9	-4.29-9		$H_\gamma(10)$	1.11-7	7.11-7	6.39
$H_n(11)$	8.60-9	-2.24-9		$H_\gamma(11)$	1.55-7	6.74-7	6.06
H_n	10.19	9.49	0.931	H_γ	6.40	8.68	1.36
Total heating = $H_n + H_\gamma$					16.59	18.17	1.10

$H_n = \mathrm{Sum}_J\, H_n(J)$, $\quad H_\gamma = \mathrm{Sum}_J\, Hn(J)$

* Read as 8.69×10^{-2}

Distances in cm: 0 | 150 | 200 | 200.5 | 203.5 | 204 | 264 | 294 | 300

Origin →	Plasma	Vacuum	Nb	94 % Li 6% Nb	Nb	20 cm	94% Li 6% Nb 20 cm	20 cm	C	94 % Li 6% Nb
Zone Number	1	2	3	4	5		6		7	8
Region Number	1	2	3	4	5	6	7	8	9	10
Material	A	B	C	D	C		D		E	D
Number of Intervals Per Zone	1	1	3	6	3		30		15	3
Thickness (cm)	150	50	0.5	3	0.5		60		30	6

Comment: The intervals in each zone are of equal step length. There are 62 intervals all together.

Fig. 4.1 Configuration of the Benchmark Blanket Model From Ref.24

Distances in cm: 0 | 133.5 | 134.5 | 137.0 | 155.0 | 200.0 | 205.0 | 290.0 | 390.0 | 400.0 | 405.0 | 500.0

Origin →	Plasma / Vacuum	Mo	SS	Li_2O Pebble: Li_2O(21%) SS(5%) He (21%)	Li_2O Block: Li_2O(67%) SS (8%) He (8%)	SS	Vacuum	SS(90%) Water (10%)	Air	SS	SCM: SS(60%) Cu(29%) Nb(7%)
Region Number	1	2	3	4	5	6	7	8	9	10	11
Number of Intervals Per Region	2	2	2	6	26	2	1	40	1	2	40
Thickness (cm)	133.5	1.0	2.5	18.0	45.0	5.0	85.0	100.0	10.0	5.0	95.0

Fig. 4.2 Calculational model for the INTOR-J Outboard Model

5. CONCLUSIONS

Two coupled cross section sets GICX40 and GICX40V4, which have the same format cross sections for the same 40 nuclides but of different origins are produced for the use in fusion reactor nuclear design in the Japan Atomic Energy Research Institute. The following conclusions have been obtained from the comparison of the results calculated with the use of two cross section sets.

1. Although region integrated values of tritium production reactions differed as much as 20% when different cross sections are used, the sums over the whole model agreed within 4%. Most of the differences may be attributed to the consideration of nonelastic neutron anisotropy by the NJOY code in the processing of the GICX40V4 in contrast to the neglect of the anisotropy in the GICX40 set.
2. Inadequate gamma-production data in ENDF/B-IV resulted in negative neutron kerma factors and gross overestimation of gamma-production when the GICX40V4 set was used. However, total heating in the whole calculational system agreed within 14% due to the compensation procedure in the NJOY system to conserve energy balance.
3. An automatic selection of proper gamma-production data and their renormalization should improve the present situation. More appropriate gamma-production data are requested. At the same time, integral experiments are strongly requested for the verification of neutron and gamma-ray heating rate in a mock-up model of a fusion reactor blanket, shield and magnet.

ACKNOWLEDGMENTS

The GICX40 set originates from the neutron cross sections for 8 nuclides obtained by T. Hiraoka in 1973.[10] A. Sagara of Nagoya University helped us compile coupled cross sections for 30 nuclides in 1974. H. Nakajima and Y. Nakao of Kyushu University added fissile nuclides, F and ^{11}B. M. Ebisuya of Osaka University added the rest of several nuclides to make the total of 40 nuclides. H. Narita and M. Igarashi of Century Research Center Corporation converted and modified some processing codes used in making GICX40.

The NJOY system used in the production of the GICX40V4 set was partly implemented to the computer in JAERI by D.W. Muir of the Los Alamos Scientific Laboratory in 1977. The conversion of NJOY was completed and further modified by K. Minami, M. Kimijima and J. Sakamaki of Fujitsu Limited. H. Kawasaki of Century Research Center Corporation assisted us in the compilation of the GICX40V4 set.

The authors deeply thank all of the above people for their contributions.

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