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AGS, A 7090 COMPUTER PROGRAM FOR THE CALCULATION OF AVERAGE GROUP CROSS SECTIONS

M. E. Tsagaris

ABSTRACT

This program calculates infinite-medium thermal spectra for each composition in a reactor. The spectra are obtained iteratively from the neutron balance with the energy transfer probability kernel, the absorption cross sections, and the slowing-down sources. Group transfer cross sections are calculated from point values of the flux spectrum together with point values of the transfer probability kernel, and group cross sections are obtained by averaging over the computed spectra. Macroscopic group cross sections are calculated for each composition; microscopic cross sections are calculated for each nuclide in each composition.

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M. E. Tsagaris

Introduction

The AGS program has been written to calculate group cross sections for use in multigroup thermalization calculations. This program consists of 3 chain links and is designed to work in conjunction with a family of neutron thermalization programs.[1] The first chain is a revised version of IMS,[1] which in addition to obtaining transfer probabilities from the kernel library tape and generating the $\phi(E)$ spectrum, permits flux values to be read in. Chain two calculates group transfer cross sections, and chain three computes the remainder of the average group microscopic and macroscopic cross sections for up to a maximum of twenty-five nuclides.

Description of the Problem — Chain 1

The first link obtains a numerical solution to the space independent Boltzmann equation by means of a normalized extrapolated Gauss iteration scheme. The equation which is solved is

$$\phi(E) = \frac{\int_{E_1}^{E_2} \Sigma(E' \to E)\ \phi(E')\ dE' + S(E)}{\int_{E_1}^{E_2} \Sigma(E \to E')\ dE' + \Sigma_a(E)} \tag{1}$$

where $\Sigma(E \to E')$ is the transfer probability that a neutron will scatter at energy E into a unit energy interval at E', $S(E)$ is the outside source term, $\phi(E)$ is the flux per unit energy at the energy E, and $\Sigma_a(E)$ is the absorption cross section which includes the leakage term.

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[1] C. A. Preskitt, E. A. Nephew, and M. E. Tsagaris, "Neutron Thermalization Programs for the IBM 7090," USAEC Report ORNL-TM-461, Oak Ridge National Laboratory.

Since a detailed description of the problem and method of solution are fully given in Ref. 1, they will not be repeated here.

Preparation of Input Data for Chain 1

The first link consists of a revised IMS program and its associated subroutines. Although the input data are generally the same as for the IMS code, they will be repeated along with the points of difference in order to incorporate all input for AGS in one report.

The input data required for the first chain calculation is divided into two distinct parts, card input information to logical tape 10, and data obtained from the kernel library tape, which is designated as logical tape 2. The card input includes the title of the calculation, cross section data, and the kernel number of the kernel to be selected from the KERNEL LIBRARY TAPE. The selection of a kernel automatically specifies for the calculation an energy mesh, an integration interval, a set of source values, S(E), and the model by which the transfer probabilities are calculated. The card input data must be assembled in the following order

1. Card 1. Columns 1-72 may contain any alphabetic or numeric information which identifies the calculation. The contents of this card and the kernel title which is read from logical tape 2 will be printed at the top of each page of output.

2. Card 2. Seven numbers are punched which give the following information.

Fields 1 and 2. These contain a neutron temperature (°K) and a value for λ for a modified Campbell-Freemantle initial guess for the spectrum. If T = 0, then the initial guess for the flux is taken to be the converged flux from the previous case in the same run.

Field 3. Each element in the table of transfer probabilities is multiplied by this number. Consequently, this field should normally contain the macroscopic free atom scattering cross section or the atomic number density. An exception, however, is in the case where it is desired to run several calculations using the same kernel without reading the

kernel from tape each time. Re-reading of the kernel can be avoided by placing a 0.0 in field 1 of card 3. If this is done, however, field 3 of card 2 must contain a 1.0 in order to retain the same normalization of the transfer probabilities as before.

Field 4. Any non-zero number in this field is used as the value at 0.0253 ev of a 1/v cross section (cm^{-1}). If this field is punched, then no cross section cards will be read and the problem will use the 1/v cross section alone. Conversely, if this field is left blank, cross section data will be supplied.

Field 5. This field is ignored unless field 4 contains zero or is blank. If field 4 contains a zero or is blank, then this field must contain a number between 1.0 and 25.0 inclusive which indicates the number of sets of cross sections that are to be read in. This number includes leakage cross sections, if one wishes to use them.

Field 6. The convergence criterion is supplied in this field. Iteration will continue until the maximum fractional change in the flux at any point in the mesh is less than this number or until the maximum number of iterations specified in the next field has been reached. A value of 10^{-3} to 10^{-4} should be satisfactory for most cases.

Field 7. This field specifies the maximum number of flux iterations to be undertaken. If left blank the number of iterations is automatically set as 100.

3. Card 3. Field 1 of card 3 contains the number of the kernel which is to be selected from the KERNEL LIBRARY TAPE and used in the calculation. A zero in this field will cause the same kernel used in the previous calculation to be re-used. In this case, field 3 of card 2 must contain a 1.0 to prevent a renormalization of the transfer probabilities, unless a renormalization is desired. Field 2 contains the extrapolation factor, α, to be used in the normalized extrapolated Gauss iteration scheme employed in the flux iterations. An optimum choice of this factor can greatly reduce the running time of the program. For initial runs, a factor $\alpha \cong 1.0$ to 1.2 is a reasonable choice to make. Field 3 contains one of three option numbers. Their significance is:

0.0 – calculate $\phi(E)$'s as described above.

1.0 – read in $\phi(E)$ values, seven numbers per card.

–1.0 – by-pass entire chain and use $\phi(E)$'s calculated in previous case.

4. Cards 4. If field 4 of card 2 contains a zero or is blank, then these cards must contain up to seven number densities each (controlled by field 5 of card 2) which are used to form a composite macroscopic cross section from the sets of microscopic cross sections which follow. Units are cm^{-1} per barn.

5. Cards 5. If the option of reading cross section cards has been selected in field 4 of card 2 and card 4 has been supplied, then cards 5 must contain as many sets of microscopic absorption cross section decks as are specified in field 5 of card 2. If leakage is calculated, there must be one less set of cross sections than specified in field 5 of card 2. A maximum of 25 sets may be supplied and must be consistent in order with the number supplied on card 4. Fields 1 through 6 contain three pairs of numbers (less if required on the last card), giving values of the energy in ev and the cross section at that energy. Each card of a cross section deck must contain an identifying 4-punch in column 72, and the energy range specified must exceed the energy range over which an interpolation is to be made. The card records on logical 10 are read one by one and a running count is kept of the number of energy mesh point values. This is continued until a blank or zero number in an energy field (1, 3, or 5) is encountered. This blank signifies that the end of the cross section deck has been reached. Hence, to those cross section decks with a number of energy mesh points exactly divisible by three, a final card must be added with all blanks except for a 4-punch in column 72.

6. Cards 6. These cards contain the leakage cross sections and are considered by the program to be a set of cross sections. The format of these cards is exactly as described for cards 05, except that an identifying 8-punch is required in column 72. This 8-punch selects an option which allows a straight linear interpolation of the leakage rather than a linear interpolation of the logarithm of the cross section versus the logarithm of the energy as it does with cards 05.

Output

A complete listing of the output is obtained from logical tape 9 and consists of the following lines.

1. Title of the flux calculation.
2. Title of the kernel used in the calculation.
3. Number of iterations, degree of convergence, extrapolation factor used.
4. A list of the energy mesh points and the computed fluxes.

Description of the Problem – Chain 2

Chain 2 generates group transfer cross sections by the solution to Eq. (2).

$$\Sigma(g_i \rightarrow g_j) = \frac{\int_{g_i} dE_i \int_{g_j} dE_j \; \Sigma(E_i \rightarrow E_j) \; \phi(E_i)}{\int_{g_i} \phi(E_i) \; dE_i} \qquad (2)$$

The flux spectrum values and the transfer probabilities are obtained from the first link.

Preparation of Input Data for Chain 2

These input data cards immediately follow the data cards described above.

1. Card 1. Columns 1-72 may contain any alphabetic or numeric information which identifies the calculation.

2. Card 2. Field 1 contains the number of group boundaries listed on cards 3. The value should equal the number of groups desired plus one.

3. Card 3. Contain seven numbers per card (or less, if required on the last card) which list the group boundaries in ev.

Output

The output consists of the following information.

1. Title of the case as described in card 1.
2. The computed group transfer cross sections from each group to all other groups, and a tabulation of their lower and upper boundaries in ev.
3. A listing of the average group fluxes.

Description of the Problem — Chain 3

The last chain was written to perform the numerical integrations encountered in the calculation of spectrum averaged cross sections. The microscopic fission and absorption group cross sections for each nuclide are given by Eqs. (3) and (4).

$$\bar{\sigma}_{a_g} = \frac{\int_g \phi(E)\ \sigma_a(E)\ dE}{\int_g \phi(E)\ dE} \tag{3}$$

$$\bar{\sigma}_{f_g} = \frac{\int_g \phi(E)\ \sigma_f(E)\ dE}{\int_g \phi(E)\ dE} \tag{4}$$

In addition, this program calculates the macroscopic cross sections and the diffusion coefficients by Eqs. (5-10).

$$\Sigma_a(g) = \sum_i N_i \sigma_a{}^i \tag{5}$$

$$\Sigma_f(g) = \sum_i N_i \sigma_f{}^i \tag{6}$$

$$\Sigma_s(g) = \left[\sum_g \Sigma(g \rightarrow g')\right] + \Sigma(g \rightarrow g) \tag{7}$$

$$\Sigma_R(g) = \left[\sum_{g'} \Sigma(g \rightarrow g')\right] + \Sigma_a(g) \tag{8}$$

$$\Sigma_T(g) = \Sigma_R(g) + \Sigma(g \rightarrow g) \tag{9}$$

$$D(g) = \frac{1}{3[\Sigma_T(g) - \bar{\mu}\,\Sigma_s(g)]} \tag{10}$$

All integrations in this program are done by a generalized Simpson's rule subroutine which does not require equally spaced intervals or an odd number of mesh points. The integration routine is explained in detail in Ref. 1.

Preparation of Input Data for Chain 3

The following cards are inserted as the last data cards for any given case.

1. Card 1. Columns 1-72 may contain any alphabetic or numeric information which identifies the calculation.

2. Cards 2. Contain the names of the elements and must be punched in the same order as their corresponding cross sections. Each element has a field width of 10 columns with the name punched in the first 6 columns of the field. As many as seven nuclides may be punched on one card.

3. Card 3. Field 1 of card 3 contains $\bar{\mu}$, the average cosine of the scattering angle for neutrons in the center of mass system.

4. Cards 4. Contain ν, the average number of neutrons produced per fission. If elements listed in cards 2 have no fission cross sections, 0.0 must be punched in the corresponding ν fields.

5. Cards 5. Contain the absorption cross sections. Refer to cards 5 description in chain 1 input section.

6. Cards 6. If a value of ν for an element has been punched in cards 4, the fission cross section deck follows cards 5 in the same format as previously described for cross section decks.

Output

The output consists of the following information.

1. Title of the average group cross sections case.
2. The value of $\bar{\mu}$.
3. Computed macroscopic group cross sections for Σ_a, Σ_f, $\nu\Sigma_f$, Σ_T, Σ_R, and D, the diffusion coefficient.
4. The microscopic group cross sections, $\bar{\sigma}_a$, $\bar{\sigma}_f$, and $\nu\bar{\sigma}_f$ for each individual nuclide.

Considerable freedom is allowed in the format of the input numbers. Unless otherwise described, all numbers must contain a decimal point even if they are integers. The most convenient form is that of a decimal fraction with 10 or less digits including the decimal point (5.12, 64.3, etc.), but an exponential notation may be used. If this is done, the exponent must be separated from the characteristic by a plus or minus sign and must be written in the appropriate exponent field beginning in column 9, 19, etc. Both columns in the exponent field must be filled, with a zero or sign in the first column if necessary.

To operate the program, an input tape is prepared on a card-to-tape unit using the input data described above. This tape is selected as logical tape 10 and, together with the kernel library tape selected as logical tape 2, constitutes all of the input data. An auxout tape 6 is selected from which the deck of cards containing the energy mesh and computed fluxes are punched.

The AGS chain program has been compiled and loaded on a saved tape and is "file protected" to guard the information and eliminate the loading time. Any number of cases may be stacked for the calculation of more than one problem.

An input data form, list of data cards, and output from a sample problem are given on the following pages.

Acknowledgment

The author wishes to thank R. S. Carlsmith and C. A. Preskitt for their helpful suggestions.

TEN DIGIT DATA FORM

PROBLEM	REQUEST
AGS — CHAIN 1	3325

CODER	DATE	PAGE
MET	2/18/63	1 of 3

1	9	11	19	21	29	31	39	41	49	51	59	61	69	71	73 80	
← AGS.		COLUMNS 1-72 ARE AVAILABLE FOR CASE IDENTIFICATION. →													1	
T(°K)		λ		Σ_s		Σ_a^o		XN		ϵ		XNI			2	
XNK		α		XF											3	
N_1		N_2		etc.		N(XN)									4	
E_1		$\sigma_a(E_1)$		E_2		$\sigma_a(E_2)$		E_3		$\sigma_a(E_3)$				04	5	cross section deck 1
etc.														04	5	
E_1		$L(E_1)$		E_2		$L(E_2)$		E_3		$L(E_3)$				08	6	leakage
etc.														08	6	

Note: Cards 4, 5, and 6 are omitted unless $\Sigma_a^o = 0.0$.

T(°K) = neutron temperature for initial flux guess. The initial guess is $\phi(E) = M(E) + \lambda F(E)$.

Σ_s = free atom macroscopic scattering cross section.

Σ_a = 2200 m/s value of the macroscopic absorption cross section.

XN = number of microscopic cross section decks which will be provided, including leakage deck.

ϵ = degree of convergence desired.

XNI = maximum number of iterations allowed. If left blank, program assumes 100 iterations.

XNK = the kernel number to be selected from the library tape.

α = extrapolation factor of the iteration scheme.

XF = option for calculating or reading in $\phi(E)$'s.

N_1(etc.) = atomic number density of the component associated with cross section deck 1.

TEN DIGIT DATA FORM

PROBLEM	REQUEST
CHAIN 2	3325

CODER	DATE	PAGE
MET	2/18/63	2 of 3

1	9	11	19	21	29	31	39	41	49	51	59	61	69	71	73 80
→ CØLUMNS 1-72 ARE AVAILABLE FØR CASE IDENTIFICATIØN.														→	1
EICT															2
E_1		E_2		E_3		E_4		E_5		E_6		E_7			3
E_8		.....				E(EICT)									3

EICT = number of group boundaries to be listed in cards 3.

E_1 (etc.) = values of the group boundary energies in electron volts.

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AGS INPUT DATA FØRM

TEN DIGIT DATA FORM

PROBLEM	REQUEST
CHAIN 3	3325

CODER	DATE	PAGE
MET	2/18/63	3 of 3

1	9	11	19	21	29	31	39	41	49	51	59	61	69	71	73 80	
COLUMNS 1-72 ARE AVAILABLE FOR CASE IDENTIFICATION															1	
NAME(1)		NAME(2)		NAME(3)		etc.		NAME(NS)							2	
$\bar{\mu}$															3	
ν_1		ν_2		ν_3		etc.		ν(NS)							4	
E_1		$\sigma_a(E_1)$		E_2		$\sigma_a(E_2)$		E_3		$\sigma_a(E_3)$				04	5	cross section deck 1
etc.		etc.												04	5	
E_1		$\sigma_f(E_1)$		E_2		$\sigma_f(E_2)$		E_3		$\sigma_f(E_3)$				04	6	fission cross section deck
etc.		etc.												04	6	

NOTE: Cards 6 are required only if an element has fission cross sections.

NAME(1),etc. = name of the element associated with cross section deck 1.

$\bar{\mu}$ = cosine of the scattering angle.

ν_1,etc. = number of neutrons produced per fission.

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AGS INPUT DATA FORM

TEN DIGIT DATA FORM

PROBLEM	REQUEST
AGS	3325

CODER	DATE	PAGE
MET	2/18/63	1 of 1

1	9	11	19	21	29	31	39	41	49	51	59	61	69	71	73 80
SAMPLE-PRØBLEM-CHAIN-1.-CØRE SPECTRUM AT 850K.-12-THERMAL-GRØUPS.															1
1380.0		0.4		0.2379		0.0		5.0		0.0001					2
2.0		1.0		0.0											3
0.0505		9.14	-5	0.67	-5	9.7	-5	1.0							4
1.001	-5	0.201		10.0		2.01	-4							04	Graphite
1.001	-5	3.8	+2	10.0		0.38								04	Thorium
1.001	-5	1.36	+2	10.0		0.136								04	U-238
0.0001		11640.0		0.0002		8231.0		etc.		etc.				04	U-235
0.0066		-0.0140541		0.02205		-0.0182331		etc.		etc.				08	Leakage
SAMPLE-PRØBLEM-CHAIN-2.-12-THERMAL-GRØUP-TRANSFERS.															1
13.0															2
1.001	-4	0.0131		0.031		0.0564		0.072		0.0894		0.109			3
0.153		0.265		0.406		0.675		1.89		5.0					3
SAMPLE-PRØBLEM-CHAIN-3.-12-THERMAL-GRØUP-CRØSS-SECTIØNS.															1
CARBON		TH-232		U-238		U-235									2
0.056															3
0.0		0.0		0.0		2.43									4
Cross section cards as listed in chain 1, omitting the leakage terms, are inserted here.															

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SAMPLE PRØBLEM INPUT

SAMPLE PROBLEM CHAIN I. CORE SPECTRUM AT 850K. 12 THERMAL GROUPS.

IT#850K A#12 MESH(.0001(161)5.0) SOURCES(TKS GAM-I KERNEL 1) 2.00000

NUMBER OF ITERATIONS# 42 MAXIMUM DIVERGENCE# 0.930E-04 EXTRAPOLATION FACTOR#1.100

ENERGY	FLUX	ENERGY	FLUX	ENERGY	FLUX	ENERGY	FLUX	ENERGY	FLUX
1.0010E-04	2.3098E-01	1.4100E-01	1.7754E 02	6.2600E-01	4.2564E 01	1.6900E 00	1.5969E 01	3.2600E 00	8.1976E 00
1.8000E-04	4.1647E-01	1.5300E-01	1.7353E 02	6.5000E-01	4.0825E 01	1.7300E 00	1.5590E 01	3.3200E 00	8.0928E 00
3.0000E-04	6.9664E-01	1.6500E-01	1.6837E 02	6.7500E-01	3.9165E 01	1.7700E 00	1.5226E 01	3.3700E 00	7.9494E 00
5.0000E-04	1.1673E 00	1.7800E-01	1.6205E 02	7.0100E-01	3.7588E 01	1.8100E 00	1.4878E 01	3.4300E 00	7.7809E 00
7.0000E-04	1.6422E 00	1.9100E-01	1.5524E 02	7.2700E-01	3.6137E 01	1.8500E 00	1.4527E 01	3.4900E 00	7.6115E 00
1.0010E-03	2.3643E 00	2.0500E-01	1.4752E 02	7.5300E-01	3.4806E 01	1.8900E 00	1.4196E 01	3.5400E 00	7.4605E 00
1.4200E-03	3.3842E 00	2.1900E-01	1.3957E 02	7.8000E-01	3.3530E 01	1.9300E 00	1.3868E 01	3.6000E 00	7.2836E 00
2.1000E-03	5.0708E 00	2.3400E-01	1.3112E 02	8.0800E-01	3.2314E 01	1.9800E 00	1.3486E 01	3.6600E 00	7.2908E 00
2.8200E-03	6.8936E 00	2.4900E-01	1.2290E 02	8.3500E-01	3.1224E 01	2.0200E 00	1.3235E 01	3.7200E 00	7.2541E 00
3.7000E-03	9.1662E 00	2.6500E-01	1.1463E 02	8.6400E-01	3.0139E 01	2.0600E 00	1.3024E 01	3.7800E 00	7.1877E 00
4.7000E-03	1.1803E 01	2.7500E-01	1.0981E 02	8.9300E-01	2.9124E 01	2.1100E 00	1.2759E 01	3.8400E 00	7.1035E 00
5.8000E-03	1.4758E 01	2.8100E-01	1.0710E 02	9.2200E-01	2.8178E 01	2.1500E 00	1.2517E 01	3.9000E 00	7.0113E 00
7.0300E-03	1.8114E 01	2.9000E-01	1.0359E 02	9.5200E-01	2.7260E 01	2.2000E 00	1.2245E 01	3.9600E 00	6.9124E 00
8.4000E-03	2.1909E 01	2.9700E-01	1.0096E 02	9.8200E-01	2.6392E 01	2.2400E 00	1.2020E 01	4.0200E 00	6.8136E 00
9.8400E-03	2.5948E 01	3.0700E-01	9.7497E 01	1.0100E 00	2.5620E 01	2.2900E 00	1.1761E 01	4.0800E 00	6.7054E 00
1.1500E-02	3.0646E 01	3.1400E-01	9.5203E 01	1.0400E 00	2.4822E 01	2.3400E 00	1.1510E 01	4.1500E 00	6.5918E 00
1.3100E-02	3.5191E 01	3.2500E-01	9.1746E 01	1.0700E 00	2.4010E 01	2.3800E 00	1.1312E 01	4.2100E 00	6.4923E 00
1.6900E-02	4.6024E 01	3.3200E-01	8.9726E 01	1.1100E 00	2.2973E 01	2.4300E 00	1.1081E 01	4.2700E 00	6.3962E 00
2.1100E-02	5.7897E 01	3.4000E-01	8.7501E 01	1.1400E 00	2.2416E 01	2.4800E 00	1.0854E 01	4.3400E 00	6.2938E 00
2.5800E-02	7.0694E 01	3.4900E-01	8.5081E 01	1.1700E 00	2.2211E 01	2.5300E 00	1.0640E 01	4.4000E 00	6.1995E 00
3.1000E-02	8.4204E 01	3.6000E-01	8.2182E 01	1.2100E 00	2.1884E 01	2.5800E 00	1.0431E 01	4.4700E 00	6.1058E 00
3.6700E-02	9.8098E 01	3.6800E-01	8.0148E 01	1.2400E 00	2.1530E 01	2.6300E 00	1.0228E 01	4.5300E 00	6.0167E 00
4.2800E-02	1.1177E 02	3.8700E-01	7.5665E 01	1.2800E 00	2.1009E 01	2.6800E 00	1.0025E 01	4.6000E 00	5.9287E 00
4.9400E-02	1.2497E 02	4.0600E-01	7.1541E 01	1.3100E 00	2.0578E 01	2.7300E 00	9.8328E 00	4.6600E 00	5.8462E 00
5.6400E-02	1.3728E 02	4.2600E-01	6.7524E 01	1.3500E 00	2.0052E 01	2.7800E 00	9.6436E 00	4.7300E 00	5.7600E 00
6.4000E-02	1.4869E 02	4.4600E-01	6.3819E 01	1.3800E 00	1.9613E 01	2.8300E 00	9.4652E 00	4.8000E 00	5.6738E 00
7.2000E-02	1.5845E 02	4.6700E-01	6.0278E 01	1.4200E 00	1.9086E 01	2.8800E 00	9.2866E 00	4.8600E 00	5.6022E 00
8.0500E-02	1.6655E 02	4.8800E-01	5.7067E 01	1.4500E 00	1.8686E 01	2.9400E 00	9.0817E 00	4.9300E 00	5.5122E 00
8.9400E-02	1.7308E 02	5.1000E-01	5.4031E 01	1.4900E 00	1.8191E 01	3.0000E 00	8.8891E 00	5.0000E 00	5.4483E 00
9.8900E-02	1.7769E 02	5.3200E-01	5.1290E 01	1.5300E 00	1.7699E 01	3.0400E 00	8.7509E 00	2.3098E-01	9.2409E-40
1.0900E-01	1.8027E 02	5.5500E-01	4.8794E 01	1.5700E 00	1.7235E 01	3.1000E 00	8.5706E 00	4.1647E-01	-0.
1.1900E-01	1.8096E 02	5.7800E-01	4.6571E 01	1.6100E 00	1.6790E 01	3.1500E 00	8.4062E 00	6.9664E-01	0.2296E-40
1.3000E-01	1.7997E 02	6.0200E-01	4.4473E 01	1.6500E 00	1.6370E 01	3.2100E 00	8.2549E 00	1.1673E 00	5.0500E-02

SAMPLE PROBLEM CHAIN 2. 12 THERMAL GROUP TRANSFERS.

AVERAGE GROUP TRANSFER CROSS SECTIONS.

FROM GROUP 1 1.0010E-04 EV TO 1.3100E-02 EV.

1, 1.0627E-01 2, 1.1935E-01 3, 7.1264E-02 4, 1.8286E-02 5, 1.0348E-02 6, 5.4386E-03
7, 3.6146E-03 8, 6.7159E-04 9, 5.8802E-06 10, 1.3974E-08 11, 1.1780E-13 12, 0.

FROM GROUP 2 1.3100E-02 EV TO 3.1000E-02 EV.

1, 3.0956E-02 2, 9.6116E-02 3, 8.5734E-02 4, 2.5352E-02 5, 1.5854E-02 6, 9.2677E-03
7, 7.1417E-03 8, 1.7203E-03 9, 2.4562E-05 10, 9.8026E-08 11, 1.8758E-12 12, 0.

FROM GROUP 3 3.1000E-02 EV TO 5.6400E-02 EV.

1, 8.8050E-03 2, 3.9742E-02 3, 9.6554E-02 4, 4.1974E-02 5, 2.8737E-02 6, 1.8515E-02
7, 1.6401E-02 8, 5.0716E-03 9, 1.1518E-04 10, 7.5829E-07 11, 3.2293E-11 12, 0.

FROM GROUP 4 5.6400E-02 EV TO 7.2000E-02 EV.

1, 3.3331E-03 2, 1.7306E-02 3, 6.0072E-02 4, 5.4915E-02 5, 4.4258E-02 6, 3.0306E-02
7, 2.9344E-02 8, 1.0659E-02 9, 3.2302E-04 10, 2.8827E-06 11, 1.9618E-10 12, 0.

FROM GROUP 5 7.2000E-02 EV TO 8.9400E-02 EV.

1, 1.6809E-03 2, 9.6468E-03 3, 3.6668E-02 4, 3.9248E-02 5, 5.4303E-02 6, 4.3153E-02
7, 4.4528E-02 8, 1.8246E-02 9, 6.9021E-04 10, 7.8757E-06 11, 7.9790E-10 12, 0.

FROM GROUP 6 8.9400E-02 EV TO 1.0900E-01 EV.

1, 8.1842E-04 2, 5.2282E-03 3, 2.1909E-02 4, 2.4958E-02 5, 3.9818E-02 6, 5.4694E-02
7, 6.6844E-02 8, 3.0786E-02 9, 1.4452E-03 10, 2.1032E-05 11, 3.1713E-09 12, 0.

FROM GROUP 7 1.0900E-01 EV TO 1.5300E-01 EV.

1, 2.7657E-04 2, 2.0553E-03 3, 9.9294E-03 4, 1.2409E-02 5, 2.1127E-02 6, 3.4187E-02
7, 9.3348E-02 8, 6.6923E-02 9, 4.3824E-03 10, 9.4755E-05 11, 2.8141E-08 12, 0.

FROM GROUP 8 1.5300E-01 EV TO 2.6500E-01 EV.

1, 3.2151E-05 2, 3.1176E-04 3, 1.9515E-03 4, 2.8932E-03 5, 5.5872E-03 6, 1.0219E-02

```
 7,  4.2582E-02  8,  1.4472E-01  9,  3.2959E-02 10,  1.5163E-03 11,  1.7477E-06 12,  4.7562E-23

FROM GROUP   9          2.6500E-01 EV TO 4.0600E-01 EV.

   1,  3.8349E-06  2,  1.0386E-05  3,  1.0404E-04  4,  2.0780E-04  5,  5.0293E-04  6,  1.1466E-03
 7,  6.6913E-03  8,  6.9796E-02  9,  1.3601E-01 10,  2.6095E-02 11,  1.0606E-04 12,  6.1104E-19

FROM GROUP  10          4.0600E-01 EV TO 6.7500E-01 EV.

   1,  1.4663E-08  2,  1.1137E-07  3,  1.8527E-06  4,  5.0657E-06  5,  1.5757E-05  6,  4.6100E-05
 7,  4.0303E-04  8,  9.1486E-03  9,  6.1432E-02 10,  1.5772E-01 11,  1.1740E-02 12,  5.1784E-14

FROM GROUP  11          6.7500E-01 EV TO 1.8900E 00 EV.

   1,  2.1198E-06  2,  1.4892E-11  3,  5.5163E-10  4,  2.4249E-09  5,  1.1228E-08  6,  4.8867E-08
 7,  8.3526E-07  8,  7.1800E-05  9,  1.6685E-03 10,  3.3484E-02 11,  2.0338E-01 12,  1.1943E-03

FROM GROUP  12          1.8900E 00 EV TO 5.0000E 00 EV.

   1,  9.1445E-07  2,  0.          3,  3.0513E-25  4,  1.2422E-23  5,  2.1342E-22  6,  3.3953E-21
 7,  6.1028E-19  8,  6.6307E-15  9,  1.5881E-11 10,  1.5625E-07 11,  3.7352E-02 12,  2.0181E-01


AVERAGE GROUP FLUXES.


 1,  2.2340E-01
 2,  1.0780E 00
 3,  2.8618E 00
 4,  2.3181E 00
 5,  2.8954E 00
 6,  3.4775E 00
 7,  7.8694E 00
 8,  1.6227E 01
 9,  1.2698E 01
10,  1.3999E 01
11,  2.7212E 01
12,  2.6198E 01
```

SAMPLE PROBLEM CHAIN 3. 12 THERMAL GROUP CROSS SECTIONS.

MU, 5.6000E-02

MACROSCOPIC CROSS SECTIONS.

GROUP	UPPER E	SIGMA A	SIGMA F	NU SIGMA F	SIGMA S	SIGMA T	SIGMA R	DIFF. COEFF.
1	1.31000E-02	1.29063E-01	1.08306E-01	2.63183E-01	3.35247E-01	4.64310E-01	3.58040E-01	7.48162E-01
2	3.10000E-02	7.18895E-02	6.04133E-02	1.46804E-01	2.72167E-01	3.44057E-01	2.47941E-01	1.01374E 00
3	5.64000E-02	4.86431E-02	4.10760E-02	9.98146E-02	2.55916E-01	3.04559E-01	2.08005E-01	1.14852E 00
4	7.20000E-02	3.84344E-02	3.23808E-02	7.86854E-02	2.50518E-01	2.88952E-01	2.34038E-01	1.21246E 00
5	8.94000E-02	3.35098E-02	2.80049E-02	6.80519E-02	2.48172E-01	2.81682E-01	2.27379E-01	1.24478E 00
6	1.09000E-01	2.93280E-02	2.50954E-02	6.09819E-02	2.46522E-01	2.75850E-01	2.21157E-01	1.27205E 00
7	1.53000E-01	2.51661E-02	2.10075E-02	5.10482E-02	2.44732E-01	2.69898E-01	1.76551E-01	1.30110E 00
8	2.65000E-01	2.17165E-02	1.75542E-02	4.26568E-02	2.42775E-01	2.64492E-01	1.19771E-01	1.32857E 00
9	4.06000E-01	1.93486E-02	1.58177E-02	3.84369E-02	2.40678E-01	2.60027E-01	1.24012E-01	1.35200E 00
10	6.75000E-01	9.15617E-03	7.74944E-03	1.88311E-02	2.40508E-01	2.49665E-01	9.19485E-02	1.41126E 00
11	1.89000E 00	5.53464E-03	4.37088E-03	1.06212E-02	2.39798E-01	2.45332E-01	4.19563E-02	1.43738E 00
12	5.00000E 00	2.56968E-03	1.67006E-03	4.05825E-03	2.39165E-01	2.41735E-01	3.99227E-02	1.45980E 00

MICROSCOPIC CROSS SECTIONS.

GROUP 1 1.0010E-04 EV TO 1.3100E-02 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	7.33752E-03	0.	0.
TH-232	1.38719E 01	0.	0.
U-238	4.96469E 00	0.	0.
U-235	1.31331E 03	1.11656E 03	2.71323E 03

GROUP 2 1.3100E-02 EV TO 3.1000E-02 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	4.25001E-03	0.	0.
TH-232	8.03484E 00	0.	0.
U-238	2.87563E 00	0.	0.
U-235	7.31147E 02	6.22818E 02	1.51345E 03

GROUP 3 3.1000E-02 EV TO 5.6400E-02 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	3.03843E-03	0.	0.
TH-232	5.74429E 00	0.	0.
U-238	2.05585E 00	0.	0.
U-235	4.94339E 02	4.23464E 02	1.02902E 03

GROUP 4 5.6400E-02 EV TO 7.2000E-02 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	2.51006E-03	0.	0.
TH-232	4.74538E 00	0.	0.
U-238	1.69835E 00	0.	0.
U-235	3.90335E 02	3.33823E 02	8.11190E 02

GROUP 5 7.2000E-02 EV TO 8.9400E-02 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	2.23936E-03	0.	0.
TH-232	4.23362E 00	0.	0.
U-238	1.51519E 00	0.	0.
U-235	3.40202E 02	2.88710E 02	7.01566E 02

GROUP 6 8.9400E-02 EV TO 1.0900E-01 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	2.02022E-03	0.	0.
TH-232	3.81932E 00	0.	0.
U-238	1.36692E 00	0.	0.
U-235	2.97606E 02	2.58716E 02	6.28679E 02

GROUP 7 1.0900E-01 EV TO 1.5300E-01 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	1.76364E-03	0.	0.
TH-232	3.33425E 00	0.	0.
U-238	1.19331E 00	0.	0.
U-235	2.55302E 02	2.16572E 02	5.26270E 02

GROUP 8 1.5300E-01 EV TO 2.6500E-01 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	1.41676E-03	0.	0.
TH-232	2.67845E 00	0.	0.
U-238	9.58602E-01	0.	0.
U-235	2.20553E 02	1.80971E 02	4.39760E 02

GROUP 9 2.6500E-01 EV TO 4.0600E-01 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	1.11281E-03	0.	0.

TH-232	2.10382E 00	0.	0.
U-238	7.52945E-01	0.	0.
U-235	1.96857E 02	1.63069E 02	3.96257E 02

GROUP 10 4.0600E-01 EV TO 6.7500E-01 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	8.82816E-04	0.	0.
TH-232	1.66901E 00	0.	0.
U-238	5.97328E-01	0.	0.
U-235	9.23200E 01	7.98911E 01	1.94135E 02

GROUP 11 6.7500E-01 EV TO 1.8900E 00 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	6.03584E-04	0.	0.
TH-232	1.14110E 00	0.	0.
U-238	4.08395E-01	0.	0.
U-235	5.56405E 01	4.50606E 01	1.09497E 02

GROUP 12 1.8900E 00 EV TO 5.0000E 00 EV.

ELEMENT	SIGMA A	SIGMA F	NU SIGMA F
CARBON	3.65879E-04	0.	0.
TH-232	6.91712E-01	0.	0.
U-238	2.47560E-01	0.	0.
U-235	2.56322E 01	1.72171E 01	4.18377E 01

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