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FENDL Multigroup Libraries

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Abstract: Selected neutron reaction nuclear data libraries and photon-atomic interaction cross section libraries for elements of interest to the IAEA's program on Fusion Evaluated Nuclear Data Library (FENDL) have been processed into MATXSR format using the NJOY system on the VAX4000 computer of the IAEA. This document lists the resulting multigroup data libraries. All the multigroup data generated are available cost-free upon request from the IAEA Nuclear Data Section.

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FENDL Multigroup Libraries

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1. INTRODUCTION

The IAEA Nuclear Data Section, in co-operation with several national nuclear data centers and research groups, is creating an internationally available Fusion Evaluated Nuclear Data Library (FENDL), which will serve as a comprehensive source of processed and tested nuclear data tailored to the requirements of the Engineering and Development Activities (EDA) of the International Thermonuclear Experimental Reactor (ITER) Project and other fusion-related development projects. The FENDL activity [1-4] is supported by several IAEA Coordinated Research Programs. Within the scope of the FENDL project [1-4], the International Atomic Energy Agency is committed to a program of nuclear data processing activities in order to create a modern and internationally available Fusion Evaluated Nuclear Data Library(FENDL). The nuclear data processing code system NJOY developed at Los Alamos National Laboratory, USA [5] with all modifications and improvements as available is employed in this study. Selected neutron interaction libraries and photon-atomic interaction cross section libraries for elements of interest to the IAEA's program on FENDL have been processed into MATXSR format using the NJOY system on the VAX4000 computer of the IAEA. This document lists the resulting multigroup data libraries.

2. SOURCE OF BASIC DATA FOR FENDL AND STATUS OF PROGRESS IN PROCESSING

The coupled neutron-gamma data files for elements and isotcpes of FENDL are to be derived from ENDF/B-VI, JENDL-3.1 and BROND-2 for the following materials following the recommendation of the FENDL meetings [1-3]:

ENDF/B-VI:

Neutron interaction and photon production cross sections:

^{1}H, ^{3}H, ^{6}Li, ^{7}Li, ^{9}Be, ^{10}B, ^{11}B, C, ^{16}O, ^{19}F, P, S, Cl, K, ^{55}V,

$^{50,52\text{-}54}Cr$, ^{55}Mn, $^{54,56\text{-}58}Fe$, ^{59}Co, $^{58,60\text{-}62,64}Ni$, $^{63,65}Cu$, $^{134\text{-}138}Ba$,

$^{182\text{-}184,186}W$, $^{206\text{-}208}Pb$.

At the time of writing this report, all the 'point' ENDF i.e. files in 'PENDF' format, all the 'group' ENDF i.e. files in 'GENDF' format and in 'MATXSR' format for all the above nuclides except ^{19}F from ENDF/B-VI have been generated and are available cost-free from the NDS upon request. The MATXSR files have been generated in the following way:

1. MATXSR files: P-5 neutron interaction and photon production cross sections.

2. MATXSR files: P-5 neutron interaction, photon production and photon interactions cross sections.

Presently MATXSR does not permit the use of different Legendre orders for neutrons and photons. Users interested to maintain a higher Legendre order for photon interactions in their neutronic calculations may have to perform a photon source term calculations using P-5 MATXSR files for neutron interactions and photon production and follow it up with a P-8 calculation for photon transport subsequently.

JENDL 3.1:

Neutron interaction and photon production cross sections:

^{23}Na, Mg, ^{27}Al, Ca, Ti, ^{55}Mn, Mo, ^{181}Ta, ^{209}Bi.

At the time of writing this report, all the multigroup cross sections in GENDF and MATXSR format for all the above nuclides except Mo from JENDL-3.1 have been generated and are available cost-free from the NDS upon request. Note that the coupled library i.e. the MATXSR files (P-5 neutron interaction, photon production and photon interactions) uses ENDF/B-VI for photon interaction data.

BROND-2:

Neutron interaction and photon production cross sections:

^{2}H, ^{14}N, ^{15}N, Si, $^{90\text{-}92,94,96}$Zr, ^{93}Nb, Sn.

At the time of writing this report, the processing of BROND-2 materials was incomplete. The reasons are: Compatibility problems of BROND-2 with ENDF/B-VI formats as required by the latest version of NJOY, absence of photon production cross sections for the isotopes etc. In particular, the HEATR module of NJOY which generates kerma and damage cross sections could not be run because of these problems. The PENDF, GENDF and MATXSR files for neutron cross sections for all the isotopes of Zirconium from BROND-2 for only the neutron interactions (i.e. without the photon production part) have been generated and are available cost-free from the NDS upon request. The problems associated with processing of BROND-2 library are under discussion with the evaluation experts of BROND-2 [6].

ENDF/B-VI:

Photon-atomic interaction cross sections for all the elements.

At the time of writing this report, all the point ENDF, i.e. PENDF, Group ENDF, i.e., GENDF and MATXSR files (P-8 photon interactions) for all the following elements from ENDF/B-VI have been generated and are available cost-free from the NDS upon request.

H, Li, Be, B, C, N, O, F, Na, Mg, Al, Si, P, S, Cl, K, Ca, Ti, V, Cr, Mn, Fe, Co, Ni, Cu, Zr, Nb, Mo, Sn, Ba, Ta, W, Pb, Bi.

The basic evaluated data are drawn from the basic libraries available from the NDS [7].

3. SPECIFICATIONS FOR MULTIGROUP PROCESSING

The specifications of multigroup processing as used in the NJOY runs for the FENDL project to produce a coupled neutron-gamma library is given below:

Neutron Groups: 175 (Vitamin-J structure)

Gamma groups: 42 (Vitamin-J structure)

Neutron weight function:

Thermal + 1/E +Fission + Fusion (IWT=6 in NJOY)

Gamma weight function:

1/E with rolloffs (IWT=3 in NJOY)

Legendre order for neutrons: P_6 for transport correction to P_5

Legendre order for gammas: P_8 for transport correction to P_7

Temperatures: 300, 900 and 1500 Kelvin

Dilution factors: 10^0, 10^1, 10^2, 10^3, 10^4, and 10^{10} barns

Reconstruction, linearization and thinning tolerances: 0.1 %

No of digits for resonance reconstruction: 7 (Maximum possible with the VAX system with NJOY at present).

Thermal Scattering law included for Be in Be metal, C in graphite and H in water.

In spite of the limited manpower available for processing activities at the IAEA, evaluated files for most of the materials of FENDL have been successfully processed, and including up to the generation of Group ENDF or "GENDF" files. These GENDF files were further post-processed into MATXSR format. A coupled neutron-gamma library in MATXSR format for use in transport codes has also been successfully produced and is freely available internationally to users as results of this program of nuclear data processing activity. The NDS obtained in April 1992 consulting assistance from the NJOY and TRANSX [8] author R.E. MacFarlane on these problems. Currently, efforts towards production of a point library for use in Monte-Carlo code, in particular the MCNP code, are underway.

REFERENCES

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6. S. Ganesan, Private communication to V. Pronyaev (June 1992).

7. H.D. Lemmel, "Index of Data Libraries available on magnetic tape from the IAEA Nuclear Data Section," IAEA-NDS-7 (Rev.91/10), October 1991.

8. R.E. MacFarlane, "TRANSX 2: A Code for Interfacing MATXS Cross Section Libraries to Nuclear Transport Codes," LA-12312-MS (To be published).

9. R.E. MacFarlane (Private communication).

LIST OF FILES, SIZES, CONTENTS ETC.

GENDF Formatted files:

1. GENDF files for neutron interaction cross sections and photon production cross section data (output of GROUPR module).

2. GENDF files for photon interaction cross section data (output of GAMINR module).

MATXSR formatted files:

The above mentioned GENDF files have been post-processed into MATXSR format.

Generated and made available are the following three types of multigroup data files in MATXSR format.

3. Photon interaction cross section data, P-8 Legendre order, 42 energy groups in VITAMIN-J group structure derived from ENDF/B-VI for all the FENDL elements mentioned in the text.

4. Neutron interaction cross section data and photon production cross section data, P-5 Legendre order, 175 energy groups in VITAMIN-J group structure derived from ENDF/B-VI, JENDL-3.1 or BROND-2.

5. A coupled neutron-gamma, 175 neutron -42 photon groups, P-5 Legendre order, for neutron interaction cross section data derived from ENDF/B-VI, JENDL-3.1 or BROND-2, and, photon production cross section data and photon interaction cross section data derived from ENDF/B-VI.

MATXSR formatted files:

DATA SET: **XNOM.GANFEN.MATXSRF** (Member Name)

Photon interaction cross section data, P-8 Legendre order, 42 energy groups in VITAMIN-J group structure derived from ENDF/B-VI for all the 34 FENDL elements mentioned in the text.

Number of 80 column records: 1006 each.

Element	NDS internal name
H	HG
Li	LIG
BE	BEG
B	BG
C	CG
N	NG
O	OG
F	FG
Na	NAG
Mg	MGG
Al	ALG
Si	SIG
P	PG
S	SG
Cl	CLG
K	KG
Ca	CAG
Ti	TIG
V	VG
Cr	CRG
Mn	MNG
Fe	FEG
Co	COG
Ni	NIG
Cu	CUG
Zr	ZRG
Nb	NBG
Mo	MOG
Sn	SNG
Ba	BAG
Ta	TAG
W	WG
Pb	PBG
Bi	BIG

MATXSR formatted files:

DATA SET: **XNOM.GANFEN.MATXSRF** (Member Name)

Neutron interaction cross section data and photon production cross section data, P-5 Legendre order, 175 energy groups in VITAMIN-J group structure derived from ENDF/B-VI, JENDL-3.1 or BROND-2.

Element	NDS internal name	Number of 80 column records	Basic File used
^{1}H	H1N	302550	ENDF/B-VI
^{3}H	H3N	78699	ENDF/B-VI
^{6}Li	LI6N	73373	ENDF/B-VI
^{7}Li	LI7N	66511	ENDF/B-VI
^{9}Be	BE9N	38229	ENDF/B-VI
^{10}B	B10N	49874	ENDF/B-VI
^{11}B	B11N	44122	ENDF/B-VI
C	CN	32978	ENDF/B-VI
^{14}N	N14N	44750	ENDF/B-VI
^{15}N	N15N	30513	ENDF/B-VI
O	ON	37904	ENDF/B-VI
^{23}Na	NA23J3N	37786	JENDL-3.1
Mg	MGJ3N	40689	JENDL-3.1
^{27}Al	AL27N	31363	JENDL-3.1
^{31}P	P31N	20135	ENDF/B-VI
S	SN	34929	ENDF/B-VI
Cl	CLN	22586	ENDF/B-VI
K	KN	22841	ENDF/B-VI
Ca	CAJ3N	38847	JENDL-3.1
Ti	TIJ3N	37574	JENDL-3.1
V	VN	26363	ENDF/B-VI

Element	NDS internal name	Number of 80 column records	Basic File used
^{50}Cr	CR50N	36326	ENDF/B-VI
^{52}Cr	CR52N	36450	ENDF/B-VI
^{53}Cr	CR53N	40917	ENDF/B-VI
^{54}Cr	CR54N	29499	ENDF/B-VI
^{55}Mn	MN55J3N	29547	JENDL-3.1
^{55}Mn	MN55N	47817	ENDF/B-VI
^{54}Fe	FE54N	36318	ENDF/B-VI
^{56}Fe	FE56N	43433	ENDF/B-VI
^{57}Fe	FE57N	41851	ENDF/B-VI
^{58}Fe	FE58N	34978	ENDF/B-VI
^{59}Co	CO59N	25323	ENDF/B-VI
^{58}Ni	NI58N	38796	ENDF/B-VI
^{60}Ni	NI60N	39112	ENDF/B-VI
^{61}Ni	NI61N	40925	ENDF/B-VI
^{62}Ni	NI62N	34804	ENDF/B-VI
^{64}Ni	NI64N	31817	ENDF/B-VI
^{63}Cu	CU63N	48783	ENDF/B-VI
^{65}Cu	CU65N	41455	ENDF/B-VI
^{90}Zr	ZR90BRN	17598	BROND-2
^{90}Zr	ZR91BRN	16993	BROND-2
^{92}Zr	ZR92BRN	18368	BROND-2
^{94}Zr	ZR94BRN	16650	BROND-2
^{96}Zr	ZR96BRN	14278	BROND-2
^{134}Ba	BA134N	13544	ENDF/B-VI
^{135}Ba	BA135N	14659	ENDF/B-VI
^{136}Ba	BA136N	14102	ENDF/B-VI

Element	NDS internal name	Number of 80 column records	Basic File used
^{136}Ba	BA137N	13944	ENDF/B-VI
^{136}Ba	BA138N	14427	ENDF/B-VI
^{181}Ta	TA181J3N	30464	JENDL-3.1
^{182}W	W182N	34644	ENDF/B-VI
^{183}W	W183N	34375	ENDF/B-VI
^{184}W	W184N	34732	ENDF/B-VI
^{186}W	W186N	35390	ENDF/B-VI
^{206}Pb	PB206N	37372	ENDF/B-VI
^{207}Pb	PB207N	37715	ENDF/B-VI
^{208}Pb	PB208N	27570	ENDF/B-VI
^{209}Bi	BI209N	23271	JENDL-3.1

MATXSR formatted files:

DATA SET: **XNOM.GANFEN.MATXSRF** (Member Name)

A coupled neutron-gamma, 175 neutron - 42 photon groups, P-5 Legendre order, for neutron interaction cross section data derived from ENDF/B-VI, JENDL-3.1 or BROND-2, and, photon production cross section data and photon interaction cross section data derived from ENDF/B-VI.

Element	NDS internal name	Number of 80 column records	Basic File used
^{1}H	H1NG	303395	ENDF/B-VI
^{3}H	H3NG	79690	ENDF/B-VI
^{6}Li	LI6NG	74284	ENDF/B-VI
^{7}Li	LI7NG	67485	ENDF/B-VI
^{9}Be	BE9NG	39220	ENDF/B-VI
^{10}B	B10NG	50865	ENDF/B-VI
^{11}B	B11NG	45113	ENDF/B-VI
C	CNG	33969	ENDF/B-VI
^{14}N	N14NG	45741	ENDF/B-VI
^{15}N	N15NG	31504	ENDF/B-VI
O	ONG	38895	ENDF/B-VI
^{23}Na	NA23J3NG	38777	JENDL-3.1
Mg	MGJ3NG	41680	JENDL-3.1
^{27}Al	AL27NG	32354	JENDL-3.1
^{31}P	P31NG	21126	ENDF/B-VI
S	S00NG	35920	ENDF/B-VI
Cl	CLNG	23577	ENDF/B-VI
K	KNG	23832	ENDF/B-VI
Ca	CAJ3NG	39838	JENDL-3.1
Ti	TIJ3NG	38565	JENDL-3.1

Element	NDS internal name	Number of 80 column records	Basic File used
V	VNG	27354	ENDF/B-VI
^{50}Cr	CR50NG	37310	ENDF/B-VI
^{52}Cr	CR52NG	37434	ENDF/B-VI
^{53}Cr	CR53NG	41901	ENDF/B-VI
^{54}Cr	CR54NG	30485	ENDF/B-VI
^{55}Mn	MN55NG	48808	ENDF/B-VI
^{55}Mn	MN55NGJE	30537	JENDL-3.1
^{54}Fe	FE54NG	37309	ENDF/B-VI
^{56}Fe	FE56NC	44424	ENDF/B-VI
^{57}Fe	FE57NG	42834	ENDF/B-VI
^{58}Fe	FE58NG	35962	ENDF/B-VI
^{59}Co	CO59NG	26314	ENDF/B-VI
^{58}Ni	NI58NG	39778	ENDF/B-VI
^{60}Ni	NI60NG	40091	ENDF/B-VI
^{61}Ni	NI61NG	41913	ENDF/B-VI
^{62}Ni	NI62NG	35788	ENDF/B-VI
^{64}Ni	NI64NG	32801	ENDF/B-VI
^{63}Cu	CU63NG	49774	ENDF/B-VI
^{65}Cu	CU65NG	42446	ENDF/B-VI
^{134}Ba	BA134NG	14535	ENDF/B-VI
^{135}Ba	BA135NG	15650	ENDF/B-VI
^{136}Ba	BA136NG	15093	ENDF/B-VI
^{136}Ba	BA137NG	14935	ENDF/B-VI
^{138}Ba	BA138NG	15418	ENDF/B-VI
^{181}Ta	TA181NGJ	31455	JENDL-3.1
^{182}W	W182NG	35635	ENDF/B-VI

Element	NDS internal name	Number of 80 column records	Basic File used
^{183}W	W183NG	35366	ENDF/B-VI
^{184}W	W184NG	35723	ENDF/B-VI
^{186}W	W186NG	36381	ENDF/B-VI
^{206}Pb	PB206NG	38363	ENDF/B-VI
^{207}Pb	PB207NG	38706	ENDF/B-VI
^{208}Pb	PB208NG	28561	ENDF/B-VI
^{209}Bi	BI209NG	24262	JENDL-3.1

GENDF FILES: NEUTRON INTERACTION AND PHOTON PRODUCTION DATA

DATA SET: **XNOM.GANFEN.GROUPRX** (Member Name)

Element	NDS internal name	Number of 80 column records	Basic File used
^{1}H	H1E6	317945	ENDF/B-VI
^{3}H	H3E6	290691	ENDF/B-VI
^{6}Li	LI6E6	291030	ENDF/B-VI
^{7}Li	LI7E6	282970	ENDF/B-VI
^{9}Be	BE9E6	53231	ENDF/B-VI
^{10}B	B10E6	69054	ENDF/B-VI
^{11}B	B11E6	59242	ENDF/B-VI
C	CE6	47514	ENDF/B-VI
^{14}N	N14E6	62735	ENDF/B-VI
^{15}N	N15E6	45059	ENDF/B-VI
O	OE6	53039	ENDF/B-VI
^{23}Na	NA23J3	55197	JENDL-3.1
Mg	MGJ3	60250	JENDL-3.1
^{27}Al	AL27J3	48109	JENDL-3.1
Si	SIJ3	56536	JENDL-3.1
^{31}P	P31E6	34559	ENDF/B-VI
S	SE6	54682	ENDF/B-VI
Cl	CLE6	38443	ENDF/B-VI
K	KE6	38801	ENDF/B-VI
Ca	CAJ3	58576	JENDL-3.1
Ti	TIJ3	58066	JENDL-3.1
V	VE6	44685	ENDF/B-VI
^{50}Cr	CR50E6	52330	ENDF/B-VI

Element	NDS internal name	Number of 80 column records	Basic File used
^{52}Cr	CR52E6	52890	ENDF/B-VI
^{53}Cr	CR53E6	58695	ENDF/B-VI
^{54}Cr	CR54E6	44662	ENDF/B-VI
^{55}Mn	MN55E6	65535	ENDF/B-VI
^{55}Mn	MN55J3	49164	JENDL-3.1
^{54}Fe	FE54E6	52092	ENDF/B-VI
^{56}Fe	FE56E6	62269	ENDF/B-VI
^{57}Fe	FE57E6	59087	ENDF/B-VI
^{58}Fe	FE58E6	49783	ENDF/B-VI
^{59}Co	CO59E6	43027	ENDF/B-VI
^{58}Ni	NI58E6	55258	ENDF/B-VI
^{60}Ni	NI60E6	56221	ENDF/B-VI
^{61}Ni	NI61E6	58485	ENDF/B-VI
^{62}Ni	NI62E6	50068	ENDF/B-VI
^{64}Ni	NI64E6	46714	ENDF/B-VI
^{63}Cu	CU63E6	69005	ENDF/B-VI
^{65}Cu	CU65E6	59930	ENDF/B-VI
^{134}Ba	BA134E6	29706	ENDF/B-VI
^{135}Ba	BA135E6	31748	ENDF/B-VI
^{136}Ba	BA136E6	30421	ENDF/B-VI
^{137}Ba	BA137E6	30437	ENDF/B-VI
^{138}Ba	BA138E6	29105	ENDF/B-VI
^{181}Ta	TA181J3	52001	JENDL-3.1
^{182}W	W182E6	52453	ENDF/B-VI
^{183}W	W183E6	53165	ENDF/B-VI
^{184}W	W184E6	52585	ENDF/B-VI

Element	NDS internal name	Number of 80 column records	Basic File used
^{186}W	W186E6	53707	ENDF/B-VI
^{206}Pb	PB206E6	60479	ENDF/B-VI
^{207}Pb	PB207E6	59854	ENDF/B-VI
^{208}Pb	PB208E6	43947	ENDF/B-VI
^{209}Bi	BI209J3	40316	JENDL-3.1
^{90}Zr	ZR90BR	33879	BROND-2
^{91}Zr	ZR91BR	33864	BROND-2
^{92}Zr	ZR92BR	35319	BROND-2
^{94}Zr	ZR94BR	32974	BROND-2
^{96}Zr	ZR96BR	29524	BROND-2

GENDF FILES: PHOTON INTERACTION DATA

DATA SET: **XNOM.GANFEN.GAMINR** (Member Name)

Number of 80 coulumn records: 1673 for each.

All the GENDF files have been derived from ENDF/B-VI

Element	**NDS internal name**
H	H
Li	LI
Be	BE
B	B
C	C
N	N
O	O
F	F
Na	NA
Mg	MG
Al	AL
Si	SI
P	P
S	S
Cl	CL
K	K
Ca	CA
Ti	TI
V	V
Cr	CR
Mn	MN
Fe	FE
Co	CO
Ni	NI
Cu	CU
Zr	ZR
Nb	NB
Mo	MO
Sn	SN
Ba	BA
Ta	TA
W	W
Pb	PB
Bi	BI

A BRIEF DESCRIPTION OF "MATXSR" FORMAT

The FENDL multigroup library is in MATXSR format. The MATXS files are produced by MATXSR module of NJOY. The MATXSR module of NJOY reformats mutigroup constants from the GENDF tape (i.e the output file of GROUPR for neutron interaction cross sections and photon production cross sections or the output file of GAMINR for photon atomic interaction cross sections) into the MATXS interface format. The MATXS file has a very general organization to hold arbitrary vectors and matrices. The file is first divided into "data types" such as neutron scattering, photon production, gamma scattering, and neutron thermal data. Each data type is asigned a name (NSCAT, NG, GSCAT, NTHERM). Data types are distinguished by the choice of incident and secondary group structures. Each data type is divided into materials specified by nuclide, temperature, and background cross section. Each material is further subdivided into "vector partials" and "matrix partials". These reaction partials are labeled with Hollerith names so there is no limit on the quantities that can be stored. MATXSR reads cross sections from the GENDF tape, assigns the Hollerith names, and packs the cross sections into MATXS format. The code TRANSX 2.0 (ref. 8), for instance, serves to interface MATXS cross section libraries to nuclear transport codes such as ANISN. TRANSX 2.0 reads nuclear data from a library in MATXS format and produces transport tables compatible with many discrete-ordinates (S_N) and diffusion codes. The FENDL multigroup library may be post-processed using TRANSX 2.0 to produce tables for neutron, photon or coupled transport for specific application calculations.

The MATXS format is briefly described below (Ref. 9):

Material cross section file:

This file contains cross section vectors and matrices for all particles, materials, and reactions; delayed neutron spectra by time group; and decay heat and photon spectra. Formats given are for file exchange only.

File structure:

Record type	**Present if**
File identification	Always
File control	Always
Set hollerith identification	Always
File data	Always
********** (Repeat for all particles)	
Group structures	Always
**********	
********** (Repeat for all materials)	
* material control	Always
*	
* ********* (Repeat for all submaterials)	
* * vector control	N1DB.GT.0

Record type	Present if
* * * * ********* (Repeat for all vector blocks) * * * Vector block	N1DB.GT.0
* * ********* * * * * ********* (Repeat for all matrix blocks) * * * Matrix control	N2D.GT.0
* * * * * * ******* (Repeat for all sub-blocks) * * * * Matrix sub-block	N2D.GT.0
* * * ******* * * * * * * Constant sub-block	JCONST.GT.0
* * * *************	

File identification

HNAME,(HUSE(I),I=1,2),IVERS

1+3*MULT

FORMAT(4H OV ,A8,1H*,2A8,1H*,I6)

HNAME	Hollerith file name - MATXS - (A8)
HUSE	Hollerith user identifiation (A8)
IVERS	File version number
MULT	Double precision parameter 1- A8 word is single word 2- A8 word is double precision word

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File control:

NPART,NTYPE,NHOLL,NMAT,MAXW,LENGTH

FORMAT(4H 1D ,4I6)

NPART	Number of particles for which group structures are given
NTYPE	Number of data types present in set
NHOLL	Number of words in set hollerith identification record
NMAT	Number of materials on file
MAXW	Maximum record size for sub-blocking
LENGTH	Length of file

Set hollerith identification:

(HSETID(I),I=1,NHOLL)

NHOLL*MULT

FORMAT(4H 2D ,8A8/(9A8))

HSETID	Hollerith identification of set (A8) (To be edited out 72 characters per line)

File data:

(HPRT(J),J=1,NPART),(HTYPE(K),K=1,NTYPE),(HMATN(I),I=1,NMAT),
(NGRP(J),J=1,NPART),(JINP(K),K=1,NTYPE,(JOUTP(K),K=1,NTYPE),
(NSUBM(I)I=1,NMAT),(LOCM(I),I=1,NMAT)

(NPART+NTYPE+NMAT)*MULT+2*NTYPE+NPART+2*NMAT

FORMAT(4H 3D ,8A8/(9A8)) HPRT,HTYPE,HMATN
FORMAT(12I6) NGRP,JINP,JOUTP,NSUBM,LOCM

HPRT(J)	Hollerith identification for particle J	
	N	Neutron
	G	Gamma
	P	Proton
	D	Deuteron
	T	Triton
	H	He-3 Nucleus
	A	Alpha (He-4 nucleus)
	B	Beta
	R	Residual or recoil (Heavier than alpha)
HTYPE(K)	Hollerith identification for data type K	
	NSCAT	Neutron scattering
	NG	Neutron induced gamma production
	GSCAT	Gamma scattering
	PN	Proton induced neutron production
	DKN	Delayed neutron data
	DKHG	Decay heat and gamma data
	DKB	Decay beta data
HMATN(I)	Hollerith identification for material I	
NGRP(J)	Number of energy groups for particle J	
JINP(K)	Type of incident particle associated with data type K. For dk data types, JINP is 0.	
JOUTP(K)	Type of outgoing particle associated with data type K	
NSUBM(I)	Number of submaterials for material I	
LOCM(I)	Location of material I	

Group structure:

(GPB(I),I=1,NGR),EMIN

NGR=NGRP(J)

NGRP(J)+1

FORMAT(4H 4D ,1P5E12.5/(6E12.5))
GPB(I) Maximum energy bound for group I for particle J
EMIN Minimum energy bound for particle J

Material control:

HMAT,AMASS,(TEMP(I),SIGZ(I),ITYPE(I),N1D(I),N2D(I),LOCS(I),
I=1,NSUBM)MULT+1+6*NSUBM

FORMAT(4H 6D ,A8,1H*,1P2E12.5/(2E12.5,5I6))

HMAT	Hollerith material identifier
AMASS	Atomic weight ratio
TEMP	Ambient temperature or other parameters for submaterial I.
SIGZ	Dilution factor or other parameters for submaterial I
ITYPE	Data type for submaterial I
N1D	Number of vectors for submaterial I
N2D	Number of matrix blocks for submaterial I
LOCS	Location of submaterial I

Vector control:

(HVPS(I),I=1,N1D),(NFG(I),I=1,N1D),(NLG(I),I=1,N1D)
(MULT+2)*N1D
FORMAT(4H 7D ,8A8/(9A8)) HVPS
FORMAT(12I6) IBLK,NFG,NLG

HVPS(I) Hollerith identifier of vector

NELAS	Neutron elastic scattering
N2N	(n,2n)
NNF	Second chance fission
GABS	Gamma absorption
P2N	Protons in, 2 neutrons out
.	.
.	.
.	.

NFG(I) Number of first group in band for vector I
NLG(I) Number of last group in band for vector I

Vector block

(VPS(I),I=1,KMAX)

KMAX=Sum over group band for each vector in block J

KMAX

FORMAT(4H 8D ,1P5E12.5/(6E12.5))

VPS(I) Data for group bands for vectors in block J. The block size is determined by taking all the group bands that have a total length less than or equal to MAXW.

Scattering matrix control:

HMTX,LORD,JCONST,
1(JBAND(L),L=1,NOUTG(K)),(IJJ(L),L=1,NOUTG(K))

MULT+2+2*NOUTG(K)

FORMAT(4H 9D ,A8/(12I6)) HMTX,LORD,JCONST, JBAND,IJJ

HMTX	Hollerith identification of block
LORD	Number of orders present
JCONST	Number of groups with constant spectrum
JBAND(L)	Bandwidth for group L
IJJ(L)	Lowest group in band for group L

Scattering sub-block:

(SCAT(K),K=1,KMAX)

MAX=LORD times the sum over all JBAND in the group range of this sub-block
FORMAT(5H 10D ,1P5E12.5/(6E12.5))

KMAX

SCAT(K) : Matrix data given as bands of elements for initial groups that lead to each final group. The order of the elements is as follows: Band for P0 of of group I, band for P1 of group I, ... , band for P0 of group I+1, band for P1 of group I+1, etc. The groups in each band are given in descending order. The size of each sub-block is determined by the total length of a group of bands that is less than or equal to MAXW. If JCONST.GT.0, The contributions from the JCONST low-energy groups are given separately.

Constant sub-block:

```
(SPEC(L),L=1,NOUTG(K)),(PROD(L),L=L1,NING(K))

L1=NING(K)-JCONST+1

NOUTG(K)+JCONST

FORMAT(4H11D ,1P5E12.5/(6E12.5))
```

SPEC	Normalized spectrum of final particles for initial particles in groups L1 To NING(K)
PROD	Production cross section (e.g., NU*SIGF) for initial groups L1 through NING(K)

This option is normally used for the energy-independent neutron and photon spectra from fission and radiative capture usually seen at low energies.