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LAWRENCE LIVERMORE LABORATORY

University of California/Livermore, California

MORSE CROSS SECTION LIBRARY TAPES

Unclassified

Tom Wilcox

July 9, 1973

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LAWRENCE LIVERMORE LABORATORY

July 9, 1973

MOR-13

MEMORANDUM

TO: Distribution

FROM: T. P. Wilcox

SUBJECT: MORSE Cross Section Library Tapes

I INTRODUCTION

This memorandum discusses the complete topic of MORSE cross section library tapes. These tapes may be used with the MORSE-L, MORSE-C, or ANISN-L computer codes. Other codes, making use of the information contained herewithin, can also be written or modified to use these tapes.

A number of cross section libraries now exist in the MORSE tape format. These are explained and described in detail. It should be pointed out that the L-Division neutron and coupled neutron-gamma libraries have recently been updated. This memorandum describes the new versions rather than the versions that have been in existence for the last two years. Users should read the comments in Section V and then use the newer data.

It should be noted that there is not just one library which is to be used with the above three codes. Due to the fact that these codes use cross sections averaged over energy groups, the group structure and the weighting used to obtain the averaged values has a very direct effect on the calculated results. In the solution of any specific problem, care must be taken to insure that the group structure provides sufficient cross section energy detail where detail is needed and that the proper weighting spectra has been used. (An iterative procedure might be necessary in order to obtain a good spectra.) There has been a tendency to think that a large number of energy groups solves the need for worrying about cross section weighting.* This is not universally true for all problems. The libraries presented here will enable the user to start on the solution to many commonly run problems. Uncommon problems, and exact solutions, will probably require different libraries than those outlined in this memorandum.

*Fifteen years ago a large number was 16 groups - today it is up to around 100.

II LIBRARY TAPE FORMATS

MORSE library tapes are formed of a series of binary files (each separated by end-of-file markers). Each of these files contains either sets of isotope cross sections all with the same energy group structure or the file may contain data describing an albedo for a single material. The last file on each tape looks like a cross section file but contains special flags indicating that no more data is to be found on the tape. A single tape may contain just one library set (two physical files) or may contain as many sets as the tape will physically hold. The simple sketch below illustrates what a typical library tape might look like:

Each of these types of files are described below in greater detail.

A. Cross Section Files

Each cross section file contains a set (or library) of microscopic cross section data which may be used to solve transport or criticality problems. Within each set will be cross sections for a number of different isotopes or elements or mixtures. These cross sections have been averaged over specific energy groups to form what is called "multigroup cross sections." In mathematical terms:

$$\bar{\sigma}_g \equiv \frac{\int_{E_g}^{E_{g+1}} \sigma(E) \psi(E) dE}{\int_{E_g}^{E_{g+1}} \psi(E) dE}$$

Where: $\bar{\sigma}_g$ is the average cross section in the g^{th} group

E_g and E_{g+1} are the energy limits of this group

$\sigma(E)$ is the energy dependent cross section

and $\psi(E)$ is some weighting function used to obtain the average. In many cases, $\psi(E)$ is set equal to the scalar flux

A set of cross sections may then be identified by the number of groups it contains and by the specific energy bounds of those groups. Obviously, cross section sets with different numbers of groups may not be used together in the same problem. Even sets with an identical number of groups but different group boundaries may not be used together. Each cross section file is therefore identified by parameters which specify the number of groups which define neutron cross sections, the number of groups defining gamma cross sections and a group energy specification number. This latter number would allow, for example, two thirty group neutron sets; one with only high energy groups and one with only low energy groups.

The first record in each cross section file contains four words which define the library contained in the file. These four words are:

NNG	The number of neutron groups
NGG	The number of gamma groups
NE	The energy group specification number
LEN	The length of the second record

The second record contains the following quantities in the order listed with unneeded items omitted.

- a) energy limits of the neutron groups (NNG + 1 values). Highest energy first, followed by next highest; etc. In MEV (several older libraries still have energies in ev.).

- b) energy limits of the gamma groups. Similar to neutron groups. (NGG + 1 values).
- c) average velocity of particles in each group. (NNG + NGG values). In cm/sec, for gamma rays this is the velocity of light; for neutrons this is a group averaged value which is related to the flux weighting which has been used to obtain the cross sections.
- d) fission source - the probability that a fission neutron is born into a given energy group. (NNG + NGG values if NNG > 0). At present, this matrix is independent of the energy of the neutron causing fission. There is no reason why this data cannot be written so as to be dependent upon this quantity. In that case, there would be NNG * NGG values.

The remaining records in the cross section file contain data describing individual isotopes, elements, or mixtures. Data for isotopes and elements are in units of barns while the data for mixtures are in units of cm²/g. When making macroscopic cross sections from the library, isotope and element data must be multiplied by the atomic number density in units of atoms/barn-cm while mixture data is multiplied by density values in units of gm/cc. In all other ways, the data contained in the library for isotopes, elements and mixtures are identical. The following discussion will talk about isotopes but either elements or mixtures may be used interchangeably.

A given isotope's cross sections must be identified by a series of identification symbols. The first such identification is the name of the isotope. The library allows three names per cross section set in order to accommodate different methods of identification. One of these names is usually the chemical symbol followed by the approximate atomic weight in the case of isotopes - i.e., U represents natural uranium while U235 represents the isotope of uranium with a weight of 235. The second set of identification calls the cross section by a number which is 1000 times the atomic number plus the approximate atomic weight. In the case of naturally occurring elements, the atomic weight is set to zero. In the previous example, uranium would thus be number 92000 while U²³⁵ would be 92235. On the library, all names are BCD words with a maximum of ten letters or numbers. In the example given above, the cross sections for natural uranium could be reached by looking for a material with a name of either U or 92000. The third allowed name in this case is not used.

The second identification necessary to find a desired set of cross sections is an integer number describing the flux weighting used when the group averaged cross sections were generated. From the previously written expression for these quantities, it is evident that a change in the weighting function, $\psi(E)$, will produce a different set of averaged cross sections

even though the group structure remains the same. The set-up of the library allows the use of up to ten different flux weightings with any one specified isotope. By definition, the integer division of the flux weighting parameter by 10 (remainder is dropped, not rounded up) will produce the energy group specification number which defines the library. I.e.; flux weighting parameters of zero to nine are used with energy group specification number zero; flux weighting parameters of ten to nineteen with number one; etc. The flux weighting parameter may also be used to separate cross section sets obtained from different sources, the use or non-use of transport corrections and the existence of special changes in a given cross section set.

Angular scattering is represented in MORSE/ANISN type cross sections by expanding the group averaged values in a Legendre series:

$$\begin{aligned} \bar{\sigma}_{g \rightarrow g'}(\mu) = & \bar{\sigma}_{g \rightarrow g'}^{\ell=0} P_0(\mu) + \bar{\sigma}_{g \rightarrow g'}^{\ell=1} P_1(\mu) \\ & + \bar{\sigma}_{g \rightarrow g'}^{\ell=2} P_2(\mu) + \dots + \bar{\sigma}_{g \rightarrow g'}^{\ell=L} P_L(\mu) \end{aligned}$$

Each term in this series includes an entire set (or table) of values for possible scattering between each energy group. Thus the required cross sections are sometimes referred to as multi-group, multi-table cross section sets. It is necessary when specifying a desired set of cross sections to know the maximum number of tables ($L + 1$) associated with that isotope. Therefore, the third identification necessary for a given isotope is the maximum order of the Legendre expansion (L).

Each isotope, within a cross section file, contains two or more records of information. The first identifies the isotope, the second and remaining records contain cross section data. The first file is always 17 words long. The following list details the contents of this record:

1	NN1	First ID name (BCD)
2	NN2	Second ID name (BCD)
3	NN3	Third ID name (BCD)
4	NNG	Number of neutron groups
5	NGG	Number of gamma groups
6	NTG	Total number of groups (sum of the above two quantities). Set equal to - IHS if upscattering occurs in the cross section sets.

7	L	Maximum order of Legendre expansion. There will be L + 1 additional records of information.
8	NSO	Spectrum weighting number (NSO/10 must equal NE)
9	LENI	Length of each data record
10-17	TITLE(8)	Title associated with this isotope (8A10). The first word should contain a description of the isotope (i.e., "TIN") since it is used in MORSE-L to identify cross section sets. The remaining data is printed whenever the cross section set is used, it should include the date when added to the library and any other descriptive information.

The data records contain cross section information in the form usually input to ANISN or MORSE. Data for the first (highest energy neutron) group is input first followed by the second group, etc. Within each group, the data has the form:

$$\sigma_a, \nu\sigma_f, \sigma_g, \sigma_{g+NUP \rightarrow g}, \sigma_{g+NUP-1 \rightarrow g}, \dots$$

$$\sigma_{g \rightarrow g}, \sigma_{g-1 \rightarrow g}, \dots, \sigma_{g-NDOWN \rightarrow g}$$

Where: σ_a is the absorption cross section in group g (the subscript g is normally not written). (This is normally the difference between the total cross section and the sum of all the scattering cross sections from this group to other groups. Due to (n, 2n) and pair production reactions, σ_a is often negative.)

$\nu\sigma_f$ is the product of nu (number of neutrons per fission) and the fission cross section in group g

$\sigma_{g' \rightarrow g}$ is the scattering cross section for transfer from group g' to group g. This input is set to zero if group g' is not within the group structure (i.e. transfer from group -1 to g.)

There are NUP + NDOWN + 4 input values in each group, where NUP is the maximum number of groups upscatter allowed and NDOWN is the maximum number of groups downscatter allowed. Three positions of certain cross

sections have been traditionally been used to define the set - IHT the position of the total cross section (σ_g); IHS the position of the in-group scatter term ($\sigma_{g \rightarrow g}$); and IHM the position of the last scatter term ($\sigma_{g \rightarrow \text{NDOWN} \rightarrow g}$). These may be related thusly;

$$\text{IHT} = 3$$

$$\text{IHS} = 4 + \text{NUP}$$

$$\text{IHM} = \text{NUP} + \text{NDOWN} + 4$$

A thirty group set with no upscatter and full downscatter would have IHT = 3, IHS = 4 and IHM = 33.

The first cross section record (second record in each isotope set) contains cross sections for the $\ell=0$ legendre expansion. The second contains values for $\ell=1$; the third for $\ell=2$; etc. For values of ℓ greater than zero, the absorption, fission and total cross sections are set equal to zero. A factor of $(2\ell + 1)$ is included in the transfer cross sections written on to the library. Care should be taken to insure that this factor has been included.

The last isotope written in each cross section file is a dummy material which flags the end of data in that file. It is identical in format to the other isotopes except that in the first record $\text{NNG} = \text{NGG} = -9999$.

In summary, a cross section file looks like the following sketch:

while each isotope would look like:

B. Albedo Files

An albedo file is like a cross section file that contains the first two records only and without isotope records. The data in these records are defined differently, however. In the first record; the four words now are defined as:

NIDR	The number of incident directions in the albedo data.
NRDR	The number of return directions.
ID	The identification number of this albedo. This number equals 100 times the number of groups in this data plus some positive identifying integer number. The identifying number must be less than 100.
LEN	The length of the second record.

The second record contains the following data in the order listed:

- a) incident neutron directions (NIDR values). The direction cosines of the incident neutron directions. Smallest cosine first followed by the next smallest, etc.
- b) return neutron directions (NRDR values). The direction cosines of the points on the cumulative distribution function where the function is defined by input data. Smallest cosine first followed by next smallest, etc. All cosines greater than zero. Last value must be equal to one.
- c) probability of return ($[NNG + NGG] \times NIDR$ values). Number of particles which return for one incident particle. Order of input is for incident group 1, incident angle 1; incident group 1, incident angle 2; etc.
- d) cumulative probability distribution functions ($[NNG + NGG] \times [NNG + NGG + 1] \times NIDR \times NRDR \times 1/2$ values). Cumulative probability for return of each incident energy and angle combination. Starts with incident energy group one and angle one and goes thusly: energy group 1, angle 1; group 1, angle 2 group 2, angle 1 For each incident combination, the cumulative probability of return is ordered thusly: return to same energy group, angle 1; same group angle 2 same group, last angle; return to next lower group, energy group, angle 1, etc. It should be noted that the final value in each incident combination will be one with all other values between zero and one in increasing order.

C. End-of-Data Marker

The last file on the tape contains markers to indicate that fact. The four word record at the beginning of this file is identical to that

for cross section data except that $NNG = NGG = -9999$.

This file should be present on each library tape or on each disk file which simulates a library tape. In cases where this file has been omitted, problem errors in specifying cross section requirements frequently show up as system malfunctions rather than simple indications of the real problem.

III SOURCES OF LIBRARY DATA

While the library tape format is important to insure the transmittal of data in the correct order, the cross section data itself will determine the correctness of a given problem's solution. The data on the present MORSE libraries has come from a wide variety of sources. In this section, we will attempt to describe those sources and, in so doing, will attempt to provide the reader with an understanding that will allow him to generate his own library of cross section data.

The easiest way to acquire a cross section library is to find someone else who has one and then convert it to your use. This is exceedingly easy to do with MORSE and ANISN since these two codes are used in every major installation in this country and many installations abroad. The cross section data input to these codes is the same throughout the world so it is very easy to interchange data. For each case where this is done, it is necessary only to be able to read the new data (either on cards or tape) and then to write a MORSE library tape with the formats previously described. (The data is the same at different installations - the format XLIBUPDT code¹ will read the data and make, or add to, a library. For more data than this, a user may easily write his own translation code. The 16 and 66 group neutron libraries (described in Section V) were obtained in this manner.

Neutron cross section information is available at LLL on our evaluated nuclear data library (ENDL)². T Division has a code, called CLYDE³, which processes the point data on the ENDL in order to obtain group averaged cross sections such as required by MORSE. The output of CLYDE is a magnetic tape. Up to now, we have read this tape with a code called ANTRAN⁴ and punched the required data onto cards. The cards from a number of CLYDE problems were then read by a routine called NCS and the data contained thereon written onto a MORSE library tape. This flow of data is shown on figure 1. Recently, the author wrote a version of ANTRAN, called MOTRAN, which eliminates the punched cards and goes directly from the CLYDE output tape to a MORSE tape. (On the CDC-7600, the CLYDE output may be a disk file and MOTRAN may be run immediately after CLYDE, thus saving time.) MOTRAN will be written up and described in a future MORSE note. The difficulty in the new method lies in the CLYDE code. For higher order Legendre expansions, this code takes excessive running times and it is not possible to generate data for a complete library in one run. We have been insured that this problem will be corrected in the near future. Sometime, in the little more distant future, the ENDL will also contain cross section information

(a) "old" method

(b) "new" method

Figure 1 Production of MORSE Library Tapes From CLYDE Code

for gamma rays and coupled neutron-gamma data. CLYDE will then provide all the information needed to generate a cross section library.

Figure 2 illustrates the flow of information followed when a cross section library of gamma interactions is formed. T Division maintains a library of point gamma ray cross section data⁵. We have written a code, labeled SIGMA, that is able to read this library and form averaged cross sections for a specified group structure. These are punched onto cards. A code, called COMPTON, is also run with this group structure to generate the angular scattering Legendre coefficients from compton scattering off of free electrons. This code is similar to MUG⁶ code of ORNL or the GAMLEG⁷ code of LASL. The punched card output of COMPTON and SIGMA are combined in a code labeled CONV and converted to the proper MORSE tape formats. CONV, like NCS used to handle the neutron data, is a "book-keeping" code and performs no physical calculations. The SIGMA code is able to process data over the entire range of data on the basic T Division library (100 Mev to 1 Kev). The assumptions of free electron scattering in the COMPTON code limit the lower energy of valid data, however. No provision has been made for fluorescence or bremsstrahlung because of these limits. Future work is expected to eliminate these restrictions and lower the bottom energy of the gamma cross sections to 1 Kev.

The coupled neutron-gamma libraries have been made (figure 3) by reading neutron data from an already prepared neutron library and gamma data from a gamma library and then reading the gamma production cross sections. The latter data comes from the CB code which has been described in an existing MORSE note⁸. Again, a code, called CLMK, exists to perform the required bookkeeping. At sometime in the future, the CB code will be replaced by CLYDE and use will be made of the LLL-ENDF for gamma production cross sections.

IV MIXTURES

It has been found that it is very convenient to have mixture data available on the library tapes. Such materials save the user the time and trouble of locating a valid composition and at the same time allow the use of a more complete description. All too often, a user will represent stainless steel as 100% iron (which it isn't) or air as all nitrogen (which it isn't).

Table I indicates the compositions used to date on MORSE library tapes for mixtures while Table II indicates the composition of two isotopes each made up of two or more other isotopes. If the compositions listed are not correct for a given problem, a user may still mix his own in a given problem or may use XLIBUPDT¹ to put additional mixtures onto a library tape.

Figure 2 Production of MORSE Gamma Cross Section Library

Figure 3 Production of MORSE Coupled Neutron-Gamma Library

Table I

STANDARD MIXTURE COMPOSITIONS

<u>Mixture</u>	<u>Composition for 1gm/cc</u>		<u>Notes, Reference</u>
	<u>Isotope</u>	<u>Atomic Density</u>	
Air	N	.03246 at/b-cm	dry air at sea level (Mark's ME Handbook, sec. 6, pg. 9)
	O	.00874	
	A	.00020	
	C	6.3×10^{-6}	
	H1	4.2×10^{-6}	
Beryllium Oxide	Be	.02407	
	O	.02407	
CH	H1	.04632	
	C	.04632	
CH ₂	H1	.08588	polyethylene, polythene
	C	.04294	
Concrete	H	.006026	criticality library only (the composition of con- crete varies - it is best to use an analysis of your specific con- crete rather than a tabulated composition)
	C	.000052	
	O	.01992	
	Na	.00043	
	Mg	.000044	
	Al	.000757	
	Si	.004622	
	K	.00020	
	Ca	.00066	
	Fe	.00015	
Deuterium Oxide (heavy water)	H2	.06014	criticality library only
	O	.03007	
Explosives:			
LX040, LX041	H1	.01554	$C_{1.55}H_{2.58}N_{2.3}O_{2.3}F_{0.52}$ (UCRL-14592)
	C	.009339	
	N	.01386	
	O	.01386	
	F	.00313	

Table I

STANDARD MIXTURE COMPOSITIONS (CONT.)

<u>Mixture</u>	<u>Composition for 1gm/cc</u>		<u>Notes, Reference</u>
	<u>Isotope</u>	<u>Atomic Density</u>	
Explosives (Cont.):			
LX07	C	.008917	criticality library only (UCRL-18592)
	H1	.01579	
	N	.01464	
	O	.01464	
	F	.002109	
LX09	C	.00861	criticality library only (UCRL-14592)
	H1	.01650	
	N	.01560	
	O	.01638	
	F	.00009	
LX10	C	.008492	criticality library only (UCRL-14592)
	H1	.01602	
	N	.01542	
	O	.01542	
	F	.001084	
PBX9409, PBX9404	H1	.01657	$C_{1.4}H_{2.75}N_{2.57}O_{2.69}$ $Cl_{0.3}P_{0.1}$ (UCRL-14592) Cl & P omitted in criticality library
	C	.00844	
	N	.01548	
	O	.01621	
	Cl	.00018	
	F	.00006	
Mock HE, LM-04-0	C	.01570	criticality library only (from Pu array experiments)
	H1	.0201415	
	N	.0132337	
	O	.0130129	
Fiberglass	H1	.00082	$Al_{8.6}C_{2.43}N_{1.8}O_{3.92}Si_{1.9.8}$ $Mg_{3.9}H_{2.5}$
	C	.00799	
	N	.00059	
	O	.01290	
	Mg	.00128	
	Al	.00283	
	Si	.00651	

Table I

STANDARD MIXTURE COMPOSITIONS (CONT.)

<u>Mixture</u>	Composition for 1gm/cc		<u>Notes, Reference</u>
	<u>Isotope</u>	<u>Atomic Density</u>	
Lithium Hydride	Li	.07577	
	H1	.07577	
Lithium-7 Hydride	H1	.07504	
	Li7	.07504	
Lithium Fluoride	Li	.02323	
	F	.02323	
Lucite	H1	.04812	C ₅ H ₈ O ₂
	C	.03008	
	O	.01203	
Plutonium, Alpha Phase	PU239	.00239	criticality library only
	PU240	.0001297	
Plutonium, Delta Phase	PU239	.00237	criticality library only
	PU240	.0001287	
Silicon Dioxide (quartz)	Si	.01003	
	O	.02006	
Sodium Iodide	Na	.004019	used only in gamma libraries
	I	.004019	
Stainless Steel-304	C	.00002	neutron cross section mixtures used Ni-58 for Ni, Mn for Cr (Etherington - NE Handbook)
	Si	.0001	
	Cr	.0022	
	Mn	.0001	
	Fe	.00754	
	Ni	.0010	
Teflon	C	.01205	CF ₂
	F	.02410	
Uranium, Depleted (D38)	U235	1.845 x 10 ⁻⁵	criticality library only
	U238	.002512	

Table I

STANDARD MIXTURE COMPOSITIONS (CONT.)

<u>Mixture</u>	<u>Composition for 1gm/cc</u>		<u>Notes, Reference</u>
	<u>Isotope</u>	<u>Atomic Density</u>	
Uranium, Oralloy	U235	.002388	criticality library only
	U238	.0001721	
Water	H1	.06686	
	O	.03343	

Table II

COMPOSITE ISOTOPE COMPOSITIONS

<u>Isotope</u>	Composition	
	<u>Isotope</u>	<u>Atom Fraction</u>
Lithium	LI6	.0742
	LI7	.9258
Fission Product (Gamma Cross Sections Only)	SE	.0015
	BR	.0031
	KR	.0430
	RB	.0138
	SR	.1160
	Y	.0332
	ZR	.0982
	MO	.0946
	RU	.0776
	PD	.0017
	SB	.0069
	TE	.0501
	I	.0639
	XE	.0731
	CS	.0302
	BA	.0619
	LA	.0614
	CE	.0931
	ND	.0713
	PM	.0026
	SM	.0028

V AVAILABLE LIBRARIES

A. Sixteen Group Neutron Library

This library is used for criticality safety problems and should not be used for transport calculations. There are two sets of cross section data on the library - the LASL Hansen-Roach set and a LRL set made up from Howerton's cross sections. Both, of course, have the same energy specifications (Table III).

The Hansen-Roach set have been assigned a flux weighting parameter of zero (with the exception of a $1/E$ weighted deuterium which has been given a value of one). Hydrogen has been replaced with a hydrogen made up of P_3 Howerton cross sections in the first 14 groups and Hansen-Roach cross sections (P_1) in the last two groups. This avoids the problems encountered in MORSE by the existence of negative transport corrected cross sections in the P_0 and P_1 hydrogen. Several materials (Pb, Rh, Ag) in this group were obtained from Howerton data to supplement the Hansen-Roach set.

Data obtained from Howerton (about 1970-1) was weighted with either a GODIVA fast spectra (assigned a flux weighting number of 2) or with a thermal spectra (assigned a number of 6). All isotopes have a P_3 expansion. Hydrogen as modified above is used since the Howerton hydrogen is valid only for free atomic hydrogen and does not represent bound hydrogen at lower energies. The scattering has been made isotropic in this part of the library for all group to group transfers where the probability of transfer is less than 5×10^{-5} . This avoids blow-ups in MORSE-C caused by non-convergence in the CLYDE code for these transfers.

The materials available in this library are summarized in Table IV.

Table III

ENERGY STRUCTURE FOR 16 GROUP NEUTRON LIBRARY

<u>Group</u>	<u>E Top</u>	<u>v ave</u>	<u>Fission Spectra</u>
1	10 Mev	2.85×10^9 cm/sec	0.204
2	3	1.99×10^9	0.344
3	1.4	1.47×10^9	0.168
4	0.9	1.10×10^9	0.180
5	0.4	6.70×10^8	0.090
6	0.1	2.70×10^8	0.014
7	17 Kev	1.14×10^8	0
8	3	4.80×10^7	0
9	550 ev	2.06×10^7	
10	100	1.01×10^7	
11	30	5.66×10^6	0
12	10	3.19×10^6	0
13	3	1.79×10^6	0
14	1	1.09×10^6	0
15	0.4	6.06×10^5	0
16	0.1	2.18×10^5	0
	0		

Table IV

CONTENTS OF SIXTEEN-GROUP NEUTRON LIBRARY

<u>Material</u>			<u>Allowed Symbols</u>	
Hansen-Roach Data (flux weighting number = 0)				
(a) Isotopes (barns)				
Hydrogen	(P ₃)	H	1000	1001
Deuterium	(Po)	D	1001	H2
Deuterium	(P ₁)	D	1001	H2
Lithium	(Po)	LI	3000	
Lithium-6	(Po)	LI6	3006	
Lithium-7	(Po)	LI7	3007	
Beryllium	(Po)	BE	4000	4009
Boron	(Po)	B	5000	
Boron-10	(Po)	B10	5010	
Carbon	(Po)	C	6000	6012
Nitrogen	(Po)	N	7000	7014
Oxygen	(Po)	O	8000	8016
Fluorine	(Po)	F	9000	9019
Sodium	(Po)	NA	11000	11023
Aluminum	(Po)	AL	13000	13027
Silicon	(Po)	SI	14000	
Chlorine	(Po)	CL	17000	
Potassium	(Po)	K	19000	
Chromium	(Po)	CR	24000	
Iron	(Po)	FE	26000	
Cobalt	(Po)	CO	27000	
Nickel	(Po)	NI	28000	
Gallium	(Po)	GA	31000	
Zirconium	(Po)	ZR	40000	
Niobium	(Po)	NB	41000	
Molybdenum	(Po)	MO	42000	

Table IV

CONTENTS OF SIXTEEN-GROUP NEUTRON LIBRARY (CONT.)

<u>Material</u>			<u>Allowed Symbols</u>	
Hansen-Roach Data (flux weighting number = 0)				
(a) Isotopes (barns) (Cont.)				
Rhenium	(Po)	RH	45000	45103
Silver	(Po)	AG	47000	
Cadmium	(Po)	CD	48000	
Cerium	(Po)	CE	58000	
Europium	(Po)	EU	63000	
Hafnium	(Po)	HF	72000	
Tantalum	(Po)	TA	73000	
Lead	(Po)	PB	82000	90000
Therium	(Po)	TH	90232	
Uranium-233	(Po)	U233	92233	
Uranium-235	(Po)	U235	92235	
Uranium-238	(Po)	U238	92238	
Plutonium-239	(Po)	PU239	94239	
Plutonium-240	(Po)	PII240	84240	
(b) Mixtures (cm ² /gm)				
Concrete	(P ₃)	CONCRETE	CONCR	
Water	(P ₃)	WATER	H2O	
Polyethene	(P ₃)	CH2	POLYETH	
Alpha Phase Plutonium	(Po)	APU	ALPHA-PU	
Delta Phase Plutonium	(Po)	DPU	DELTA-PU	
Beryllium Oxide	(Po)	BE0	BE-OXIDE	
Oralloy	(Po)	OY	ORALLOY	
Depleted Uranium	(Po)	D38	DEPLETED-U	
Deuterium Oxide	(Po)	D20	D-OXIDE	
Deuterium Oxide	(P ₁)	D20	D-OXIDE	

Table IV

CONTENTS OF SIXTEEN-GROUP NEUTRON LIBRARY (CONT.)

<u>Material</u>			<u>Allowed Symbols</u>	
(b) Mixtures (cm ² /gm) (Cont.)				
High Explosives:				
PBX9404	(P ₃)	PX94	PBX9404	HEPBX9404
LX040	(P ₃)	HELX040	HELX040	LX040
LX07	(P ₃)	LX07	HELX07	
LX09	(P ₃)	LX09	HELX09	
LX10	(P ₃)	LX10	HELX10	
Mock	(P ₃)	LM-04-0		
CH	(P ₃)	CH		

Hansen-Roach Data (flux weighting number = 1 [1/E])

(a) Isotopes (barns)

Deuterium	(P ₀)	D	1001	H2
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Howerton Data (flux weighting number = 2 [GODIVA] and flux weighting number = 6 [Thermal])

(a) Isotopes (barns)

Hydrogen-Atomic	(P ₃)	H-ATOMIC		
(do not use GODIVA flux weighted set in MORSE code)				
Hydrogen	(P ₃)	H	1000	1001
Deuterium	(P ₃)	D	H2	1002
Tritium	(P ₃)	T	H3	1003
Helium-3	(P ₃)	HE3	2003	
Helium-4	(P ₃)	HE4	2004	
Lithium-6	(P ₃)	LI6	3006	
Lithium-7	(P ₃)	LI7	3007	
Beryllium	(P ₃)	BE	4000	4009
Boron	(P ₃)	B	5000	

Table IV

CONTENTS OF SIXTEEN-GROUP NEUTRON LIBRARY (CONT.)

<u>Material</u>		<u>Allowed Symbols</u>	
(a) Isotopes (barns) (Cont.)			
Boron-10	(P ₃)	B10	5010
Carbon	(P ₃)	C	6000 6012
Nitrogen	(P ₃)	N	7000 7014
Oxygen	(P ₃)	O	8000 8016
Fluorine	(P ₃)	F	9000 9019
Sodium	(P ₃)	NA	11000 11023
Magnesium	(P ₃)	MG	12000
Aluminum	(P ₃)	AL	13000 13027
Silicon	(P ₃)	SI	14000
Phosphorus	(P ₃)	P	15000 15031
Sulfur	(P ₃)	S	16000 16032
Chlorine	(P ₃)	CL	17000
Argon	(P ₃)	A	18000 18040
Potassium	(P ₃)	K	19000
Calcium	(P ₃)	CA	20000
Titanium	(P ₃)	TI	22000
Manganese	(P ₃)	MN	25000 25055
Iron	(P ₃)	FE	26000
Nickel-58	(P ₃)	NI58	28058
Copper	(P ₃)	CU	29000
Gallium	(P ₃)	GA	31000
Niobium	(P ₃)	NB	41000
Molybdenum	(P ₃)	MO	42000
Cadmium	(P ₃)	CD	48000
Tin	(P ₃)	SN	50000
Barium-138	(P ₃)	BA138	56138
Europium	(P ₃)	EU	63000

Table IV

CONTENTS OF SIXTEEN-GROUP NEUTRON LIBRARY (CONT.)

<u>Material</u>			<u>Allowed Symbols</u>	
(a) Isotopes (barns) (Cont.)				
Gadolinium	(P ₃)	GD	64000	
Helminum	(P ₃)	HO	67000	67165
Tantalum	(P ₃)	TA	73000	73181
Tungsten	(P ₃)	W	74000	
Gold	(P ₃)	AU	79000	79197
Lead	(P ₃)	PB	82000	
Thorium	(P ₃)	TH	90000	90232
Uranium-233	(P ₃)	U233	92233	
Uranium-234	(P ₃)	U234	92234	
Uranium-235	(P ₃)	U235	92235	
Uranium-236	(P ₃)	U236	92236	
Uranium-237	(P ₃)	U237	92237	
Uranium-238	(P ₃)	U238	92238	
Uranium-239	(P ₃)	U239	92239	
Uranium-240	(P ₃)	U240	92240	
Plutonium-238	(P ₃)	PU238	94238	
Plutonium-239	(P ₃)	PU239	94239	
Plutonium-240	(P ₃)	PU240	94240	
Plutonium-241	(P ₃)	PU241	94241	
Americium-242	(P ₃)	AM242	95242	

B. Sixty-Six Group Neutron Library

This library was made from the Howerton library at the same time the 16 group library was formulated (1970-71). The group structure (Table V) is the same as used in the SORS monte carlo code. Since there has been major changes in the Howerton evaluated library since the time these cross sections were generated, and since the origin of this data is somewhat obscure, it is not recommended that this library be used for anything but criticality safety evaluations which compare results with the previous 16 group data.

As previously, two flux weighting spectra were used - a fast GODIVA spectra and a thermal spectra. The first spectra has been assigned a flux weighting number of 2 and the thermal a number of 6 (same as in the 16 group library). All isotopes previously listed as being present in the 16 group library for Howerton data, with the exception of bound hydrogen, are available with the fast spectrum weighting. Only the following are available with the thermal weighting: Li-6, Li-7, Be, C, O, Ti, Ni-58, Cu, Ta, W, Pb, U-234, U-235, U-238, Pu-238, Pu-239, Pu-240 and Pu-241.

All materials in this library contain P_3 expansions. Only 30 group downscatter exists in the cross sections.

Table V

SIXTY-SIX GROUP NEUTRON LIBRARY

<u>Group</u>	<u>Top Energy</u>	<u>Velocity</u>	<u>Fission Spectra</u>
1	14.60 Mev	5.12×10^9 cm/sec	.000368
2	13.09	4.78	.000490
3	11.08	4.36	.00265
4	9.24	3.97	.00607
5	7.57	3.55	.01638
6	6.07	3.16	.03700
7	4.73	2.76	.07609
8	3.56	2.37	.12877
9	2.55	1.99	.18028
10	1.72	1.64	.16002
11	1.18	1.40	.10314
12	0.888	1.20	.09338
13	0.638	9.98×10^8	.07703
14	0.429	7.99	.05784
15	0.261	5.98	.03676
16	0.136	4.24	.01583
17	65.35 Kev	3.01	.00515
18	33.98	2.17	.00193
19	18.01	1.70	.000496
20	12.68	1.43	.000257
21	8.89	1.21	.000147
22	6.65	1.05	.000110
23	5.00	9.25×10^7	0
24	4.01	8.50	0
25	3.56	8.00	0
26	3.14	7.50	0
27	2.75	7.00	0
28	2.39	6.50	0

Table V

SIXTY-SIX GROUP NEUTRON LIBRARY (CONT.)

<u>Group</u>	<u>Top Energy</u>	<u>Velocity</u>	<u>Fission Spectra</u>
29	2.05	6.00	0
30	1.73	5.50	0
31	1.44	5.00	0
32	1.18	4.55	
33	0.991	4.22	0
34	0.879	3.98	0
35	0.776	3.73	0
36	0.678	3.53	0
37	0.622	3.40	0
38	0.587	3.28	0
39	0.536	3.10	0
40	0.487	2.95	0
41	0.426	2.72	0
42	0.354	2.50	0
43	0.302	2.27	0
44	0.243	2.02	0
45	0.189	1.82	0
46	0.160	1.70	0
47	0.143	1.60	0
48	0.126	1.48	0
49	0.103	1.32	0
50	81.82 ev	1.20	0
51	69.27	1.10	0
52	57.78	1.00	0
53	47.32	9.00 x 10 ⁶	0
54	37.90	8.00	0
55	29.54	7.00	0
56	22.22	6.00	0

Table V

SIXTY-SIX GROUP NEUTRON LIBRARY (CONT.)

<u>Group</u>	<u>Top Energy</u>	<u>Velocity</u>	<u>Fission Spectra</u>
57	15.95	5.00	0
58	10.72	4.00	0
59	6.54	3.00	0
60	3.40	1.98	0
61	1.31	1.18	0
62	0.428	7.96×10^5	0
63	0.261	5.87	0
64	0.136	4.20	0
65	0.065	3.03	0
66	0.034	2.37	0
	0.025		

C. Thirty Group Neutron Library

The thirty group neutron library was designed to perform penetration calculations with high energy (> 0.5 Mev) neutrons. It does have some cross sections all the way down to thermal neutron energies but the group structure is quite broad in this range. The resulting answers for thermal problems will probably be only a good estimate of the true result. It is not adequate for criticality problems since half of the energy groups occur above the energy region of interest.

This library was made from the LLL Evaluated Nuclear Data Library as it existed in May of 1973. This is the latest currently available version of that library. The CLYDE code was used and a composite spectra formed from a 20°C Maxwellian, a $1/v$, a Watt fission and a D-T infinite media fusion spectrum assumed. All materials carry a flux weight number of zero and all are P_3 expansions. (Since it was impossible to evaluate the P_2 and P_3 matrices for isotopes greater than tantalum within a reasonable amount of computation time [tungsten was still going after 80 minutes of CDC-7600 time], all isotopes with atomic weights equal to or greater than 82 have zero P_2 and P_3 cross section matrices. When the CLYDE code has been corrected to eliminate this difficulty, these matrices will be computed and added to the library.) The library's cross sections are not transport corrected.

The cross sections for hydrogen in the lower two energy groups have been altered to simulate bound hydrogen. This has been done by using the old Hansen-Roach data and large errors should be attached to the library values.

Table VI indicates the energy grouping of this library while Table VII indicates the isotopes available. In the last table, it should be noted that fissionable isotopes with and without fission gammas included are represented on this neutron only library by the same set of cross sections. In the coupled neutron-gamma library, to be described later, this is not true.

The fission neutron spectra contained in this library is a combined spectra derived from the fission of U235, U238 and Pu239 caused by 14 Mev neutrons. Table VIII lists these various fission spectra as well as a U235 spectra caused by 1 Mev neutrons. The user may well wish to override the library spectra for a given problem.

The thirty group library described here is an updated version of a library which has existed for the past two years. The newer library contains more isotopes, mixtures and updated cross section data. It should be used rather than the old version.

Table VI

ENERGY STRUCTURE FOR 30 GROUP NEUTRON LIBRARY

<u>Group</u>	<u>E Top</u>	<u>v ave</u>	<u>ΔE</u>
1	18.000 Mev	5.67×10^9 cm/sec	2.388 Mev
2	15.612	5.18×10^9	2.071
3	13.541	4.97×10^9	1.796
4	11.745	4.58×10^9	1.558
5	10.187	4.26×10^9	1.351
6	8.835	3.97×10^9	1.172
7	7.663	3.70×10^9	1.017
8	6.647	3.44×10^9	0.882
9	5.765	3.21×10^9	0.765
10	5.000	2.96×10^9	0.796
11	4.205	2.72×10^9	0.669
12	3.536	2.49×10^9	0.563
13	2.973	2.28×10^9	0.473
14	2.500	2.07×10^9	0.512
15	1.988	1.84×10^9	0.407
16	1.581	1.64×10^9	0.324
17	1.257	1.47×10^9	0.257
18	1.000	1.32×10^9	0.159
19	0.841	1.21×10^9	0.134
20	0.707	1.11×10^9	0.113
21	0.595	1.02×10^9	0.095
22	0.500	7.91×10^8	0.276
23	0.224	5.29×10^8	0.124
24	0.100	2.49×10^8	0.087
25	12.61 Kev	8.86×10^7	11.02 Kev
26	1.59	3.14×10^7	1.39
27	200 ev	1.11×10^7	174.8 ev
28	25.2	3.96×10^6	22.0
29	3.18	1.41×10^6	2.78
30	0.40	2.48×10^5	0.40
	0.0001		

Table VII

CONTENTS OF L-DIVISION NEUTRON AND COUPLED NEUTRON-GAMMA LIBRARIES

<u>Elements or Isotopes</u>	<u>Allowable Names</u>		
Hydrogen, Hydrogen-1	H	H1	1001
Deuterium	D	1002	H2
Tritium	T	1003	H3
Helium-3	HE3	2003	
Helium-4	HE4	2004	
Lithium	LI	3000	
Lithium-6	LI6	3006	
Lithium-7	LI7	3007	
Beryllium	BE	4000	4009
Boron	B	5000	
Boron-10	B10	5010	
Boron-11	B11	5011	
Carbon	C	6000	6012
Nitrogen	N	7000	7014
Oxygen	O	8000	8016
Fluorine	F	9000	9019
Sodium	NA	11000	11023
Magnesium	MG	12000	
Aluminum	AL	13000	13027
Silicon	SI	14000	
Phosphorus	P	15000	15031
Sulfur	S	16000	16032
Chlorine	CL	17000	
Argon	A	18000	18040
Potassium	K	19000	
Calcium	CA	20000	
Titanium	TI	22000	
Manganese	MN	25000	25055

Table VII

CONTENTS OF L-DIVISION NEUTRON AND COUPLED NEUTRON-GAMMA LIBRARIES (CONT.)

<u>Elements or Isotopes</u>	<u>Allowable Names</u>	
Iron	FE	26000
Nickel-58	NI58	28058
Copper	CU	29000
Gallium	GA	31000
Niobium	NB	41000 41093
Molybdenum	MO	42000
Cadmium	CD	48000
Tin	SN	50000
Barium-138	BA138	56138
Europium	EU	63000
Gadolinium	GD	64000
Holmium	HO	67000
Tantalum	TA	73000 73181
Tungsten	W	74000
Gold	AU	79000 79197
Lead	PB	82000
Thorium	TH	90232 90000
Thorium (w/o fission gammas)	TH*	
Uranium-233	U233	92233
Uranium-233 (w/o fission gammas)	U233*	
Uranium-234	U234	92234
Uranium-234 (w/o fission gammas)	U234*	
Uranium-235	U235	92235
Uranium-235 (w/o fission gammas)	U235*	
Uranium-236	U236	92236
Uranium-236 (w/o fission gammas)	U236*	
Uranium-237	U237	92237
Uranium-237 (w/o fission gammas)	U237*	

Table VII

CONTENTS OF L-DIVISION NEUTRON AND COUPLED NEUTRON-GAMMA LIBRARIES (CONT.)

<u>Elements or Isotopes</u>	<u>Allowable Names</u>	
Uranium-238	U238	92238
Uranium-238 (w/o fission gammas)	U238*	
Uranium-239	U239	92239
Uranium-239 (w/o fission gammas)	U239*	
Uranium-240	U240	92240
Uranium-240 (w/o fission gammas)	U240*	
Plutonium-238	PU238*	
Plutonium-239	PU239	94239
Plutonium-239 (w/o fission gammas)	PU239*	
Plutonium-240	PU240	94240
Plutonium-240 (w/o fission gammas)	PU240*	
Plutonium-241	PU241	94241
Plutonium-241 (w/o fission gammas)	PU241*	
Fission Products	FISS-PROD	99120
<u>Mixtures</u>		
Water	H2O	WATER
Air	AIR	
CH	CH	
CH ₂	CH2	POLYETH
LX040	LX040	LX041 HELX040
PBX9409	PBX9409	HEPBX9409
LiH	LIH	
Lucite	LUCITE	
Fiberglass	FIBERGLASS	FG
Li ⁷ H	LI7H	
Stainless Steel-304	SS304	
SiO ₂	SI02	QUARTZ

Table VII

CONTENTS OF L-DIVISION NEUTRON AND COUPLED NEUTRON-GAMMA LIBRARIES (CONT.)

<u>Mixtures</u>	<u>Allowable Names</u>
LiF	LIF
BeO	BE0
Teflon	TEFLON CF2

Table VIII

FISSION NEUTRON SPECTRA

Energy Group	²³⁵ U 14 MEV	²³⁸ U 14 MEV	²³⁹ Pu 14 MEV	Combined 14 MEV	²³⁵ U 1 MEV
1	0	0	0	0	0
2	0	0	0	0	0
3	1.106×10^{-5}	1.435×10^{-4}	1.669×10^{-4}	1.07×10^{-4}	2.444×10^{-5}
4	1.597×10^{-3}	1.636×10^{-3}	1.926×10^{-3}	1.69×10^{-3}	9.727×10^{-4}
5	3.610×10^{-3}	3.509×10^{-3}	4.703×10^{-3}	3.94×10^{-3}	2.358×10^{-3}
6	6.471×10^{-3}	6.745×10^{-3}	1.246×10^{-2}	8.73×10^{-3}	4.912×10^{-3}
7	1.180×10^{-2}	1.152×10^{-2}	1.744×10^{-2}	1.36×10^{-2}	9.030×10^{-3}
8	1.815×10^{-2}	1.797×10^{-2}	2.398×10^{-2}	2.00×10^{-2}	1.490×10^{-2}
9	2.612×10^{-2}	2.576×10^{-2}	3.189×10^{-2}	2.79×10^{-2}	2.243×10^{-2}
10	4.322×10^{-2}	4.287×10^{-2}	4.904×10^{-2}	4.50×10^{-2}	3.904×10^{-2}
11	5.544×10^{-2}	5.517×10^{-2}	5.937×10^{-2}	5.67×10^{-2}	5.230×10^{-2}
12	6.578×10^{-2}	6.558×10^{-2}	6.761×10^{-2}	6.63×10^{-2}	6.423×10^{-2}
13	7.319×10^{-2}	7.283×10^{-2}	7.661×10^{-2}	7.42×10^{-2}	7.334×10^{-2}
14	1.025×10^{-1}	1.013×10^{-1}	1.046×10^{-1}	1.028×10^{-1}	1.046×10^{-1}
15	1.020×10^{-1}	9.962×10^{-2}	1.012×10^{-1}	1.009×10^{-1}	1.049×10^{-1}
16	9.545×10^{-2}	9.186×10^{-2}	9.189×10^{-2}	9.31×10^{-2}	9.804×10^{-2}
17	8.463×10^{-2}	8.116×10^{-2}	7.953×10^{-2}	8.18×10^{-2}	8.665×10^{-2}
18	5.553×10^{-2}	5.356×10^{-2}	5.110×10^{-2}	5.34×10^{-2}	5.663×10^{-2}
19	4.795×10^{-2}	4.681×10^{-2}	4.346×10^{-2}	4.61×10^{-2}	4.882×10^{-2}
20	4.056×10^{-2}	4.039×10^{-2}	3.628×10^{-2}	3.91×10^{-2}	4.135×10^{-2}
21	3.372×10^{-2}	3.443×10^{-2}	2.990×10^{-2}	3.27×10^{-2}	3.448×10^{-2}
22	8.969×10^{-2}	9.736×10^{-2}	7.889×10^{-2}	8.86×10^{-2}	9.346×10^{-2}
23	2.956×10^{-2}	3.480×10^{-2}	2.636×10^{-2}	3.02×10^{-2}	3.263×10^{-2}
24	1.202×10^{-2}	1.428×10^{-2}	1.119×10^{-2}	1.25×10^{-2}	1.437×10^{-2}
25	4.290×10^{-4}	4.860×10^{-4}	4.261×10^{-4}	4.47×10^{-4}	5.649×10^{-4}
26	6.301×10^{-6}	7.905×10^{-6}	6.952×10^{-6}	7.05×10^{-6}	8.313×10^{-6}
27	2.851×10^{-8}	1.251×10^{-7}	1.100×10^{-7}	8.79×10^{-8}	3.761×10^{-8}
28	0	1.986×10^{-9}	1.746×10^{-9}	0	0
29	0	3.162×10^{-11}	2.781×10^{-11}	0	0
30	0	5.084×10^{-13}	4.470×10^{-13}	0	0

D. Fifteen Group Gamma Library

The fifteen group gamma library consists of fifteen equal lethargy groups between 10 and 0.5 Mev (Table IX). The cross section data was obtained by assuming a flat weighting spectra and compton scattering from free electrons. All materials carry a flux weighting number of zero and have a P_3 expansion. No transport corrections have been made.

Table X lists the contents of this library. Besides the elements and mixtures given in the table, the library contains two additional sets of cross section data. The first is a P_0 matrix labeled "X** COMPTON". It is used in the MORSE-L program by the point flux estimator to make an analytic solution to compton scattering. If this set of data is needed in a MORSE-L problem, the code calls for it without the user being aware of its use. The second set of data is a P_9 matrix labeled "COMPTON". This material may be used in a problem to extend the P_3 element data to P_9 data (or anywhere between P_3 and P_9). Each mixed material, in those cases that desire additional angular accuracy, should have added to their mixing instructions the additional component:

COMPTON 9 0 N

where N is equal to the electron density of the given material (the sum over all elements in the mixture of the product of atomic density and atomic number).

Table IX

ENERGY STRUCTURE FOR 15 GROUP GAMMA LIBRARY

<u>Group</u>	<u>E Top</u>	<u>E ave</u>	<u>ΔE</u>	<u>E ave ΔE</u>
1	10.0000	9.0948	1.8104	16.4649
2	8.1896	7.4483	1.4826	11.0430
3	6.7070	6.0999	1.2142	7.4066
4	5.4928	4.9956	0.9944	4.9676
5	4.4984	4.0912	0.8144	3.3318
6	3.6840	3.3506	0.6669	2.2346
7	3.0171	2.7440	0.5462	1.4988
8	2.4709	2.2472	0.4473	1.0052
9	2.0236	1.8404	0.3663	0.6742
10	1.6572	1.5072	0.3000	0.4522
11	1.3572	1.2344	0.2457	0.3033
12	1.1115	1.0109	0.2012	0.2034
13	0.9103	0.8279	0.1648	0.1364
14	0.7455	0.6780	0.1350	0.0915
15	0.6105	0.5552	0.1105	0.0614
	0.5000			

Table X

CONTENTS OF L-DIVISION GAMMA LIBRARIES

<u>Elements</u>		<u>Allowable Names</u>	
Hydrogen	H H3 1000 1003	H1 D 1001	H2 T 1002
Helium	HE 2000	HE3 2003	HE4 2004
Lithium	LI 3000	LI6 3006	LI7 3007
Beryllium	BE	4000	4009
Boron	B 5000	B10 5010	B11 5011
Carbon	C	6000	6012
Nitrogen	N	7000	7014
Oxygen	O	8000	8016
Fluorine	F	9000	9019
Neon	NE	10000	
Sodium	NA	11000	11023
Magnesium	MG	12000	
Aluminum	AL	13000	13027
Silicon	SI	14000	
Phosphorus	P	15000	
Sulfur	S	16000	16032
Chlorine	CL	17000	
Argon	A	18000	
Potassium	K	19000	
Calcium	CA	20000	
Scandium	SC	21000	
Titanium	TI	22000	
Vanadium	V	23000	
Chromium	CR	24000	

Table X

CONTENTS OF L-DIVISION GAMMA LIBRARIES (CONT.)

<u>Elements</u>		<u>Allowable Names</u>	
Manganese	MN	25000	25055
Iron	FE	26000	
Cobalt	CO	27000	
Nickel	NI	NI58	28000
Copper	CU	29000	
Zinc	ZN	30000	
Gallium	GA	31000	
Germanium	GE	32000	
Arsenic	AS	33000	
Selenium	SE	34000	
Bromine	BR	35000	
Krypton	KR	36000	
Rubidium	RB	37000	
Strontium	SR	38000	
Yttrium	Y	39000	
Zirconium	ZR	40000	
Niobium	NB	41000	41093
Molybdenum	MO	42000	
Technetium	TC	43000	
Ruthenium	RU	44000	
Rhodium	RH	45000	
Palladium	PD	46000	
Silver	AG	47000	
Cadmium	CD	48000	
Indium	IN	49000	
Tin	SN	50000	
Antimony	SB	51000	
Tellurium	TE	52000	

Table X

CONTENTS OF L-DIVISION GAMMA LIBRARIES (CONT.)

<u>Elements</u>		<u>Allowable Names</u>	
Iodine	I	53000	
Xenon	XE	54000	
Cesium	CS	55000	
Barium	BA 56138	56000	BA138
Lanthanum	LA	57000	
Cerium	CE	58000	
Praseodymium	PR	59000	
Neodymium	ND	60000	
Promethium	PM	61000	
Samarium	SM	62000	
Europium	EU	63000	
Gadolinium	GD	64000	
Terbium	TB	65000	
Dysprosium	DY	66000	
Holmium	HO	67000	67165
Erbium	ER	68000	
Thulium	TM	69000	
Ytterbium	YB	70000	
Lutetium	LU	71000	
Hafnium	HF	72000	
Tantalum	TA	73000	73181
Tungsten	W	74000	
Rhenium	RE	75000	
Osmium	OS	76000	
Iridium	IR	77000	
Platinum	PT	78000	
Gold	AU	79000	79197

Table X

CONTENTS OF L-DIVISION GAMMA LIBRARIES (CONT.)

<u>Elements</u>		<u>Allowable Names</u>	
Mercury	HG	80000	
Thallium	TL	81000	
Lead	PB	82000	
Bismuth	BI	83000	
Polonium	PO	84000	
Astatine	AT	85000	
Radon	RN	86000	
Francium	FR	87000	
Radium	RA	88000	
Actinium	AC	89000	
Therium	TH 90232	90000	TH*
Protactinium	PA	91000	
Uranium	U U238 92235 U233* U234* U236* U237* U239* U240*	U235 U238* 92238 92233 92234 92236 92237 92239 92240	U235* 92000 U233 U234 U236 U237 U239 U240
Neptunium	NP	93000	93237
Plutonium	PU 94000 PU238* PU240* PU241*	PU239 94239 94238 94240 94241	PU239* PU238 PU240 PU241
Fission Products	FISS-PROD	99120	

Table X

CONTENTS OF L-DIVISION GAMMA LIBRARIES (CONT.)

<u>Mixtures</u>		<u>Allowable Names</u>	
Water	WATER	H2O	
Air	AIR		
CH	CH		
CH ₂	CH2	POLYETH	
LX040	LX040	LX041	HELX040
PBX9409	PBX9409	HEPBX9409	
LiH	LIH		
Lucite	LUCITE		
Fiberglass	FIBERGLASS	FG	
Teflon	TEFLON	CF2	
Li ⁷ H	LI7H		
Stainless Steel-304	SS304		
SiO ₂	SI02	QUARTZ	
NaI	NAI		
LiF	LIF		
BeO	BE0		

E. Thirty-Five Group Gamma Library

This library is identical to the fifteen group gamma library in composition and make-up except that there are thirty-five equal lethargy groups between 10.0 and 0.125 Mev. Table XI lists these energies while Table X may be used for allowable element and mixture names.

Table XI

ENERGY STRUCTURE FOR 35 GROUP GAMMA LIBRARY

<u>Group</u>	<u>E Top</u>	<u>E ave</u>	<u>ΔE</u>	<u>E ave ΔE</u>
1	10.0000	9.4116	1.1768	11.0756
2	8.8232	8.3040	1.0383	8.6222
3	7.7849	7.3268	0.9161	6.7123
4	6.8688	6.4646	0.8083	5.2254
5	6.0604	5.7038	0.7132	4.0679
6	5.3472	5.0326	0.6293	3.1668
7	4.7180	4.4404	0.5552	2.4654
8	4.1628	3.9178	0.4899	1.9192
9	3.7629	3.4568	0.4322	1.4941
10	3.2407	3.0500	0.3814	1.1631
11	2.8593	2.6911	0.3365	0.9055
12	2.5228	2.3744	0.2969	0.7049
13	2.2259	2.0950	0.2619	0.5488
14	1.9640	1.8484	0.2311	0.4272
15	1.7329	1.6309	0.2039	0.3326
16	1.5289	1.4390	0.1799	0.2589
17	1.3490	1.2696	0.1588	0.2016
18	1.1903	1.1202	0.1401	0.1569
19	1.0502	0.9884	0.1236	0.1222
20	0.9266	0.8721	0.1090	0.0951
21	0.8176	0.7695	0.0962	0.0740
22	0.7213	0.6789	0.0849	0.0576
23	0.6365	0.5990	0.0749	0.0449
24	0.5616	0.5285	0.0661	0.0349
25	0.4955	0.4663	0.0583	0.0272
26	0.4372	0.4114	0.0514	0.0212
27	0.3857	0.3630	0.0454	0.0165
28	0.3403	0.3203	0.0401	0.0128
29	0.3003	0.2826	0.0353	0.0100
30	0.2649	0.2494	0.0312	0.0078
31	0.2338	0.2200	0.0275	0.0061
32	0.2063	0.1941	0.0243	0.0047
33	0.1820	0.1713	0.0214	0.0037
34	0.1606	0.1511	0.0189	0.0029
35	0.1417	0.1333	0.0167	0.0022
	0.1250			

F. Coupled Forty-Five Group Library

The thirty group neutron library and the fifteen group gamma library, as previously described, are coupled together to form this library. The additional coupling cross-sections, which represent the production of gamma rays by neutron interactions, were obtained as described in MOR note - 10⁸. It should be noted that all gamma production is considered to be isotropic.

Table VII lists the available isotopes and mixtures on this library. In some problems it is desirable to suppress gamma production caused by fission. All fissionable isotopes are given in this library twice, therefore; that with the normal name, i.e., U235, contains the fission produced gammas. That with a star after the normal name, i.e., U235*, does not contain the fission gammas. This is directly opposite to the naming that has been used on the present (old) coupled library.

A number of the light isotopes produce gammas with an energy of 0.478 Mev. This is below the bottom energy of the fifteen group structure. With the cross sections given above, this gamma is not produced. In order to check on the importance of this gamma, a special set of isotopes have been added to the library which include the production of this gamma in the lowest energy group of the library. The use of these isotopes will give the user an idea of the importance of the specific gamma in the solution of their problem. These isotopes are LI6*, LI7*, BE*, B* and B10*. It should be pointed out that these names do not appear in the 30 group neutron or in the 15 and 35 group gamma libraries but only in this coupled library.

The "X** COMPTON" material needed in MORSE-L for the point flux estimator exists on this library and no longer need be supplied on cards by the user.

As with the thirty group neutron library, the library described here is an updated version of a library which has existed for the past two years. The newer library should be used unless the user is trying to compare results with values previously obtained with the older cross sections sets.

G. Other Libraries

Several additional libraries of cross section data exist and have been used with MORSE or ANISN. Since their use was very limited or because they are similar to existing libraries already described, they will not be updated or cared for in any way. They are, however, available if a user would like to use one of them. A list follows with a simple explanation of each. Further information may be obtained from the author.

1) 48 group gamma (1972)

Bottom thirty-five groups identical to present 35 group library. Extends upward with same lethargy spacing to 50.92 Mev. Contains all the cross sections given in Table X except that some of the isotope names are not present.

2) 30 group gamma (1972)

Contains 30 equal lethargy groups between 10 and 0.3 Mev. All elements are included but without full list of isotope names and without mixtures.

3) 152 group gamma (1969)

Left over from R-Division. One-hundred fifty-two unequally spaced groups between 15 Mev and 1 Kev. Compton scattering from free electrons only. Fluorescence and bramsstrahlung included. Only most popular elements on library.

H. Tape Assignments

Table XII indicates the location of tapes containing the previously described cross section libraries. Except where noted, all are "purple" tapes that may be used on the "G" machine, if desired.

Please use the back-up tapes only if you can't find a working tape which may be used. In case of a tape malfunction, please inform the author so that the problem may be corrected.

The author does not maintain any of the libraries as disk files. In general, any (except the sixteen group neutron) library is too large for a practical file. A routine is now being worked on which will create a file containing only desired isotopes for a single user. That user may then keep track of his own library file. This routine will be described in a MOR note shortly.

Table XII

TAPE ASSIGNMENTS FOR CROSS SECTION LIBRARIES

<u>Library</u>	<u>Working Tapes</u>	<u>Back-Up Tape</u>
<u>STANDARD</u>		
16 group neutron	CG615, DH119*	DH121
66 group neutron	CG615	DH121
30 group neutron	CG613, DK208	DK218
15 group gamma	CG613, DK208	DK218
35 group gamma	CG614	DK216
45 group coupled	CG613, DK208	DK218
<u>NON-STANDARD</u>		
48 group gamma	DH124	DK215*
30 group gamma	DH123	DH112*
152 group gamma	CG420*	DH122*
30 group neutron (old)	CG419	DH120
45 group coupled (old)	CG419	DH120

*These tapes are not on "purple" reels and therefore may not be used on the "G" machine during the day.

VI. ADDITIONAL DATA FOR USE WITH MORSE LIBRARIES

A. Energy Absorption

Frequently it is desired to calculate the energy being absorbed, or deposited, in a given material or at a given location. Many radiation transport codes automatically perform this calculation. MORSE-L and ANISN-L do not. In order to evaluate energy absorption, it is necessary to add a detector response function which is equal to the particle energy times the energy absorption cross section. The sum over all energy groups of the product of particle flux, and the response function will then yield the required energy absorption.

For gamma-ray cross sections, the required microscopic energy absorption cross section times energy is defined by:

$$(\overline{E\sigma_{pa}})_g = \frac{\int_{E_{g+1}}^{E_g} \Phi(E) \left\{ E\sigma_{pe}(E) + (E-1.02)\sigma_{pp}(E) + E\sigma_c(E)f_c(E) \right\} dE}{\int_{E_{g+1}}^{E_g} \Phi(E) dE}$$

Where: $\Phi(E)$ is the gamma-ray number flux spectrum
 σ_{pe} is the cross section for the photoelectric process
 σ_{pp} is the cross section for the pair production process
 σ_c is the cross section for the compton process
 f_c is the fraction of energy deposited locally in the compton process

This relationship assumes that the photoelectric process is being treated as pure absorption - i.e., no new gammas are generated by fluorescence.

The value of f_c may be calculated by:

$$f_c = 1 - \frac{\int_{\mu_\ell}^1 \left(\frac{\nu'}{\nu_0} \right)^3 \left(\frac{\nu_0}{\nu'} + \frac{\nu'}{\nu_0} + \mu^2 - 1 \right) d\mu}{\int_{-1}^1 \left(\frac{\nu'}{\nu_0} \right)^2 \left(\frac{\nu_0}{\nu'} + \frac{\nu'}{\nu_0} + \mu^2 - 1 \right) d\mu}$$

Where: $\frac{\nu'}{\nu_0} = \frac{1}{1 + \frac{E}{0.51} (1 - \mu)}$

and $\mu_\ell = -1$ if $E_{\text{lower}} \leq \frac{E}{1 + 2 (E/0.51)}$

otherwise,

$$\mu_\ell = 1 + \frac{0.51}{E} \left(1 - \frac{E}{E_{\text{lower}}} \right)$$

where E_{lower} is the bottom energy of the group structure (in MEV). This relationship assumes that gammas scattered by the Compton process to energies lower than the bottom of the energy group structure are absorbed at the scattering location.

The above relationships have been solved for all available elements using the 15 and 35 group energy structures.

For neutron reactions, cross sections similar to the above are calculated by the CLYDE³ code during the generation of the needed MORSE cross section sets. It should be noted that the total energy deposition, as calculated by CLYDE, includes all gamma rays produced by neutron reactions. In neutron only problems, these gammas are assumed to be absorbed where produced. In coupled neutron-gamma problems, the energy deposition is the CLYDE "local" energy term plus any gamma energy carried by gammas lower than the lowest boundary of the gamma group structure.

All the required energy absorption data needed to go with the 15 and 35 group gamma libraries, the 30 group neutron library and the 45 group coupled library have been generated. Since this data is in microscopic form and since most users desire macroscopic values, a simple computer routine, called MEABS, has been written. This code will read the standard "MIX" block of input to MORSE-L or ANISN-L and then will calculate and punch three detector response functions per mixture. These three will, respectively, when multiplied by number flux, yield results with the following units:

- 1) MEV/cc
- 2) ergs/cc
- 3) Rads

The author will provide this code to any desiring to use it.

B. Prompt Fission Gamma Spectra

The prompt gamma spectra has been measured from the thermal neutron fission of U^{235} and Pu^{239} and from the spontaneous fission of Cf^{252} . Reference 9 and 10 discuss these measurements as well as providing additional reference material.

We have considered four representations of the U^{235} fission gamma spectra:

- 1) The experimental data given in Appendix I of reference 9 (Maienschein 1970). Interpolations between the data points were made by using a four point interpolation scheme.
- 2) A curve fit to the above data as given on page 80 of reference 9. This fit takes the form of:

$$6.6 \gamma \text{ fission}^{-1} \text{ MEV}^{-1} \quad 0.1 \leq E_{\gamma} \leq 0.6 \text{ MEV}$$

$$20.2 e^{-1.78 E_{\gamma}} \quad 0.6 \leq E_{\gamma} \leq 1.5 \text{ MEV}$$

$$7.2 e^{-1.09 E_{\gamma}} \quad 1.5 \leq E_{\gamma} \leq 10.5 \text{ MEV}$$

- 3) A compromise fit to the data of reference 9 and reference 10; again given on page 80 of reference 9.

$$6.3 \gamma \text{ fission}^{-1} \text{ MEV} \quad 0.1 \leq E_{\gamma} \leq 0.6 \text{ MEV}$$

$$17.1 e^{-1.67 E_{\gamma}} \quad 0.6 \leq E_{\gamma} \leq 1.7 \text{ MEV}$$

$$5.87 e^{-1.04 E_{\gamma}} \quad 1.7 \leq E_{\gamma} \leq 10.0 \text{ MEV}$$

- (4) The "old" curve fit to data reported in 1958: (references 11, 12, 13)

$$0.2 e^{1.4 E_\gamma}$$

$$0 \leq E_\gamma \leq 0.3 \text{ MEV}$$

$$26.8 e^{-2.30 E_\gamma}$$

$$0.3 \leq E_\gamma \leq 1.0 \text{ MEV}$$

$$8.0 e^{-1.10 E_\gamma}$$

$$1.0 \leq E_\gamma \leq 10.0 \text{ MEV}$$

Each of the four representations have been integrated in order to obtain the number of fission gammas born in each energy group of the 15 and 35 group libraries. The results are indicated on Tables XIII and XIV with the Table XV indicating the total number of photons and photon energy born within the group energy bounds. The numerical values are also available on punched cards from the author.

It is recommended that the 1970 curve fit be used. This is based upon the following considerations:

- a) The form of the data can easily be used in any of a number of available codes.
- b) Since the measured data is for thermal neutron fission, and since most of our problems involve fast neutron fission, the errors in curve fitting the experimental data are of the same order of magnitude as the errors of assuming that the spectra does not change with neutron energy.
- c) The same arguments as b) but for changes of the fissionable isotope.

The user should be aware that there is at least a 10% experimental uncertainty in the data.

The given data is based upon a single fission. When combined with a neutron fission spectra, the number of neutrons per fission (ν) must be known.

It is the author's understanding that LASL has done further work in measuring the prompt fission spectra. Those interested in more accuracy might attempt to find this newer data.

Table XIII

PROMPT FISSION SPECTRA - 15 GROUP GAMMA LIBRARY

PHOTONS / FISSION				
<u>Group</u>	<u>Experimental</u>	<u>Fit (1970)</u>	<u>Compromise Fit (1970)</u>	<u>Fit (1958)</u>
1	1.420E-03	7.553E-04	9.570E-04	7.684E-04
2	3.167E-03	3.538E-03	4.147E-03	3.656E-03
3	1.452E-02	1.217E-02	1.337E-02	1.274E-02
4	3.032E-02	3.244E-02	3.381E-02	3.432E-02
5	7.790E-02	7.008E-02	6.990E-02	7.479E-02
6	1.170E-01	1.273E-01	1.225E-01	1.368E-01
7	2.086E-01	2.005E-01	1.873E-01	2.168E-01
8	2.756E-01	2.808E-01	2.560E-01	3.052E-01
9	3.585E-01	3.572E-01	3.196E-01	3.897E-01
10	4.246E-01	4.302E-01	4.185E-01	4.594E-01
11	6.157E-01	5.559E-01	5.385E-01	5.072E-01
12	6.322E-01	6.759E-01	6.391E-01	5.472E-01
13	7.485E-01	7.653E-01	7.094E-01	6.618E-01
14	8.217E-01	8.175E-01	7.454E-01	7.635E-01
15	7.716E-01	7.324E-01	6.955E-01	8.282E-01

Table XIV

PROMPT FISSION SPECTRA - 30 GROUP GAMMA LIBRARY

PHOTONS / FISSION

<u>Group</u>	<u>Experimental</u>	<u>Fit (1970)</u>	<u>Compromise Fit (1970)</u>	<u>Fit (1958)</u>
1	2.846E-03	3.178E-04	4.123E-04	3.218E-04
2	7.450E-05	9.239E-04	1.136E-03	9.457E-04
3	2.413E-03	2.338E-03	2.739E-03	2.416E-03
4	6.834E-03	5.232E-03	5.876E-03	5.452E-03
5	1.211E-02	1.050E-02	1.136E-02	1.103E-02
6	1.637E-02	1.916E-02	2.005E-02	2.025E-02
7	3.399E-02	3.209E-02	3.262E-02	3.412E-02
8	5.589E-02	4.988E-02	4.942E-02	5.331E-02
9	6.886E-02	7.256E-02	7.026E-02	7.789E-02
10	8.937E-02	9.954E-02	9.446E-02	1.073E-01
11	1.404E-01	1.297E-01	1.209E-01	1.403E-01
12	1.624E-01	1.614E-01	1.481E-01	1.751E-01
13	1.897E-01	1.929E-01	1.746E-01	2.099E-01
14	2.280E-01	2.225E-01	1.989E-01	2.427E-01
15	2.312E-01	2.487E-01	2.304E-01	2.719E-01
16	2.898E-01	2.823E-01	2.793E-01	2.961E-01
17	3.929E-01	3.358E-01	3.267E-01	3.146E-01
18	3.782E-01	3.862E-01	3.697E-01	3.271E-01
19	3.999E-01	4.306E-01	4.063E-01	3.449E-01
20	4.510E-01	4.672E-01	4.352E-01	3.942E-01
21	4.959E-01	4.946E-01	4.556E-01	4.402E-01
22	5.037E-01	5.126E-01	4.675E-01	4.781E-01
23	5.603E-01	4.988E-01	4.642E-01	5.068E-01
24	4.257E-01	4.362E-01	4.163E-01	5.257E-01
25	4.382E-01	3.848E-01	3.673E-01	5.350E-01
26	3.438E-01	3.395E-01	3.241E-01	5.355E-01
27	3.612E-01	2.996E-01	2.860E-01	5.281E-01
28	2.351E-01	2.643E-01	2.523E-01	5.140E-01
29	1.755E-01	2.332E-01	2.226E-01	3.733E-01
30	1.385E-01	2.058E-01	1.964E-01	2.063E-01
31	1.893E-01	1.816E-01	1.733E-01	1.205E-01
32	1.677E-01	1.602E-01	1.529E-01	7.388E-02
33	1.287E-01	1.413E-01	1.349E-01	4.729E-02
34	1.393E-01	1.247E-01	1.190E-01	3.144E-02
35	1.154E-01	1.100E-01	1.050E-01	2.161E-02

Table XV

INTEGRATED PROMPT FISSION SPECTRA

	<u>Photons/Fission</u>	<u>Photon/Energy Fission</u>
<u>15 Group (0.50-10 Mev)</u>		
Experimental	5.101	6.437 Mev
Fit (1970)	5.062	6.386
Compromise Fit (1970)	4.754	6.039
Fit (1958)	4.942	6.393
<u>30 Group (0.125-10 Mev)</u>		
Experimental	7.570	7.241
Fit (1970)	7.537	7.159
Compromise Fit (1970)	7.116	6.777
Fit (1958)	7.967	7.452

C. Detector Gamma Responses

The gamma response of a number of different LLL detectors has been previously compiled and placed into the GNARM code¹³. This data has now been averaged over the MORSE gamma library energy groups and put into a form so that they may be used as detector responses in MORSE-L and ANISN-L problems.

The average detector response function in energy group g is defined as:

$$\overline{DRF}_g = \frac{\int_{E_{g+1}}^{E_g} \phi(E) DR^E(E) E dE}{\int_{E_{g+1}}^{E_g} \phi(E) dE}$$

where the limits of the group are E_g to E_{g+1} . $\phi(E)$ is the gamma flux spectrum and $DR^E(E)$ is the GNARM gamma sensitivity. Note that this sensitivity is in units of response per gamma energy flux (not number flux) thus the extra energy term in the numerator. The sensitivity is normalized to a value of 1 per gamma MEV at an energy of 1.25 MEV. In keeping with the spectrum averaging used in obtaining the MORSE gamma libraries, the flux spectrum is taken to be unity at all energies (a flat spectrum). The sensitivity was obtained by linear interpolation from the GNARM values.

Results for the average response for the 15 and 35 group libraries are presented in Tables XVI and XVII. These values may be obtained on punched cards from the author.

The data in GNARM was obtained from the following sources. Further description of the detectors and the method of determining the response function may be obtained from them. In a number of cases these reports also give the absolute normalization factors as well as the neutron sensitivity of the detectors.

<u>Detector</u>	<u>Reference</u>
L22	J. Stevens, L. P. Hocker; "First Order Calculation of
L125	the Absolute Gamma-Ray Energy Sensitivity of Evacuated
	Compton Detectors;" EGG-1183-2259; December 1970.

L24C	J. E. Doyle, W. C. Dickinson; "The Cerenkov Response of Lucite and Quartz to Gamma and Fast Neutron Radiation;" UOPB-62-61; July 1962.
L24F	C. M. Ankenbrandt; "Determining Scintillator Response to Photons and Neutrons by Monte Carlo;" UCRL-12150; September 1964.
Solid State	R. W. Kuckuck; "Semiconductor Detectors for Use in the Current Mode;" UCRL-51011; February 1971.
L24NAI L24NIIAL	Private Communication from Monroe Moen.

Table XVI

DETECTOR GAMMA RESPONSES - 15 GROUP LIBRARY

<u>Group</u>	<u>L22</u>	<u>L24C</u>	<u>L24F</u>	<u>Solid State</u>	<u>L24NAI</u>	<u>L24NAIAL</u>	<u>L125</u>
1	6.610	15.95	4.528	3.782	10.66	9.787	3.704
2	5.905	13.04	3.974	3.491	8.456	7.759	3.438
3	5.193	10.65	3.503	3.206	6.723	6.180	3.240
4	4.509	8.689	3.100	2.926	5.390	4.973	3.123
5	3.873	7.082	2.762	2.651	4.328	4.014	3.001
6	3.300	5.661	2.483	2.385	3.487	3.258	2.816
7	2.784	4.383	2.219	2.129	2.798	2.641	2.510
8	2.333	3.305	1.952	1.885	2.270	2.167	2.147
9	1.916	2.422	1.698	1.653	1.851	1.792	1.805
10	1.547	1.742	1.456	1.437	1.511	1.486	1.495
11	1.230	1.223	1.237	1.237	1.233	1.235	1.227
12	0.969	0.838	1.039	1.055	1.003	1.026	0.973
13	0.762	0.553	0.861	0.890	0.817	0.854	0.762
14	0.592	0.342	0.704	0.744	0.666	0.714	0.594
15	0.462	0.204	0.569	0.616	0.536	0.590	0.452

Table XVII

DETECTOR GAMMA RESPONSES - 35 GROUP LIBRARY

<u>Group</u>	<u>L22</u>	<u>L24C</u>	<u>L24F</u>	<u>Solid State</u>	<u>L24NAI</u>	<u>L24NAIAL</u>	<u>L125</u>
1	6.732	16.51	4.633	3.833	11.08	10.18	3.759
2	6.297	14.56	4.262	3.651	9.603	8.812	3.572
3	5.850	12.82	3.934	3.469	8.291	7.607	3.818
4	5.402	11.29	3.635	3.290	7.174	6.590	3.288
5	4.961	9.946	3.359	3.112	6.239	5.741	3.194
6	4.535	8.755	3.113	2.938	5.435	5.012	3.125
7	4.129	7.702	2.895	2.763	4.729	4.374	3.064
8	3.744	6.773	2.694	2.594	4.131	3.837	2.966
9	3.385	5.881	2.527	2.427	3.607	3.366	3.853
10	3.052	5.037	2.358	2.264	3.144	2.951	2.699
11	2.738	4.269	2.196	2.105	2.737	2.586	2.474
12	2.456	3.583	2.026	1.951	2.402	2.285	2.244
13	2.184	2.972	1.862	1.802	2.113	2.026	2.028
14	1.924	2.437	1.704	1.659	1.860	1.799	1.813
15	1.685	1.992	1.551	1.521	1.637	1.600	1.611
16	1.471	1.604	1.404	1.390	1.441	1.424	1.430
17	1.273	1.286	1.267	1.265	1.269	1.268	1.266
18	1.093	1.021	1.139	1.147	1.115	1.128	1.099
19	0.944	0.801	1.018	1.036	0.980	1.005	0.948
20	0.812	0.618	0.906	0.932	0.861	0.895	0.812
21	0.696	0.468	0.801	0.835	0.758	0.800	0.696
22	0.592	0.342	0.705	0.745	0.667	0.715	0.595
23	0.508	0.248	0.618	0.662	0.582	0.635	0.502
24	0.433	0.178	0.539	0.587	0.507	0.563	0.423
25	0.371	0.116	0.467	0.517	0.442	0.489	0.351
26	0.319	0.065	0.405	0.455	0.385	0.443	0.288
27	0.274	0.029	0.350	0.398	0.335	0.394	0.221
28	0.236	0.007	0.301	0.346	0.291	0.351	0.165
29	0.203	0.	0.257	0.300	0.249	0.309	0.099
30	0.174	0.	0.220	0.259	0.208	0.266	0.029
31	0.150	0.	0.188	0.222	0.167	0.225	-0.019
32	0.129	0.	0.161	0.190	0.133	0.189	-0.045
33	0.111	0.	0.137	0.161	0.102	0.156	-0.044
34	0.095	0.	0.117	0.137	0.077	0.129	-0.042
35	0.082	0.	0.099	0.115	0.055	0.103	-0.039

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3. R. J. Doyas, et al; "CLYDE, a Code for the Production of Calculational Constants from Nuclear Data;" UCRL-50400, Volume 5; May 1, 1971.
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5. E. F. Plechaty, et al; "Photon Cross Sections - 1 Kev to 100 Mev;" UCRL-50400, Volume VI; October 22, 1968.
6. J. R. Knight and F. R. Mynatt; "MUG, A Program for Generating Multigroup Photon Cross Sections;" CTC-17; January 20, 1970.
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8. T. P. Wilcox; "Neutron Induced Gamma-Ray Production Cross Sections;" MOR-10, October 25, 1972.
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11. F. C. Maienschein et al; Proc. U.N. International Conf. on Peaceful Uses of Atomic Energy, Second Geneva Conference, 15, Pg. 366 (1958).
12. H. C. Clairborne; "Survey of Methods for Calculating Gamma-Ray Heating;" ORNL-RSIC-8; June 1965, Pgs. 88-92.
13. R. D. Neifert, W. A. Turin; "GNARM Revisited;" DAG 70-34; May 14, 1970.

Appendix A

FORTTRAN Instructions Necessary to Read MORSE Library Tapes

I. To obtain the correct file on a library tape provided by the user:

NNG desired number of neutron groups
NGG desired number of gamma groups
NE desired energy group parameter
NDT unit number on which the library tape is hung

Instructions:

```
      DIMENSION NTIN (4), S(1000)
      REWIND NDT
5     BUFFER IN (NDT,1) (NTIN(1), NTIN(4))
10    IF (UNIT, NDT) 10, 11, 980, 990
11    IF (NTIN(1).EQ.NNG) 12, 15
12    IF (NTIN(2).EQ.NNG) 13, 15
13    IF (NTIN(3).EQ.NE) 14, 15
14    LEN = NISN (4)
      GO TO 25
15    IF (NTIN(1).EQ.-9999) 16, 20
16    WOT 3, 17
17    FORMAT ("UNABLE TO LOCATE PROPER GROUP STRUCTURE ON THIS
      TAPE")
      GO TO ERROR
20    CALL REDEF (NDT)
      GO TO 5
25    BUFFER IN (NDT,1) (S(1), S(LEN))
c      energy limits, velocities and fission spectra are now in S( )
c
980   error procedure for incorrect reads
990   error procedure for incorrect reads
```

Note: The dimension of S() should be equal to or greater than LEN - approximately $2 + 3 \times (NNG + NGG)$

II. To obtain Correct Isotope (or mixture) Data provided by the user:

Correct position in the file (part I, above)
NAME desired isotope name (left adjusted BCD)
L order of Legendre expansion desired
NSO desired flux weighting number

Instructions:

```

        DIMENSION LDUM(17), CRX(IHM, IGM, LMAX)
40  BUFFER IN (NDT,1) (LDUM(1), LDUM(17))
45  IF (UNIT, NDT) 45, 50, 980, 990
50  LENI = LDUM(9)
    IF (LDUM(1).EQ.NAME) 55, 51
51  IF (LDUM(2).EQ.NAME) 55, 52
52  IF (LDUM(3).EQ.NAME) 55, 60
55  IF (LDUM(8).EQ.NSO) 56, 60
    IF (LDUM(7).LT.L) 60, 56
56  DO 58 I = 1, LDUM (7) + 1
    IS = (I-1)*IHM*IGM+1
    BUFFER IN (NDT,1) (CRX(IS), CRX(IS-1 + LENI))
57  IF (UNIT, NDT) 57, 58, 980, 990
58  CONTINUE
    GO TO NEXT
60  IF (LDUM(4).EQ.-9999) 61, 65
    WOT 3, 61
61  FORMAT (" UNABLE TO LOCATE ISOTOPE")
    GO TO ERROR
65  DO 68 I = 1, LDUM (7) + 1
66  IF (UNIT, NDT) 66, 68, 980, 990
68  CONTINUE
    GO TO 40
```

Notes:

- 1) LDUM(1), LDUM(2), LDUM(3) and NAME are normally left adjusted by a subroutine before being checked by this programming.
- 2) NSO/10 (integer arithmetic) must equal NE in Part I.
- 3) If LDUM(7) is greater than L, then only the first L + 1 matrices are stored. The remaining matrices on the library must be skipped over in a manner similar to that indicated at statement 65.

MOR-13

TPW/mm

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