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Neutron Cross-Section Libraries in the AMPX Master Interface Format for Thermal and Fast Reactors

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NEUTRON CROSS-SECTION LIBRARIES IN THE AMPX MASTER INTERFACE FORMAT FOR THERMAL AND FAST REACTORS

M. A. Bjerke, C. C. Webster

Sponsor: CHEMICAL TECHNOLOGY DIVISION
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Waste Management Analysis for Nuclear Fuel Cycles
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ABSTRACT

Neutron cross-section libraries in the AMPX master interface format have been created for three reactor types. Included are an 84-group library for use with light-water reactors, a 27-group library for use with heavy-water CANDU reactors and a 126-group library for use with liquid metal fast breeder reactors. In general, ENDF/B data was used in the creation of these libraries, and the nuclides included in each library should be sufficient for most neutronic analyses of reactors of that type. Each library has been used successfully in fuel depletion calculations.

1. INTRODUCTION

The purpose of this report is to document the creation of three reactor-dependent multigroup neutron cross-section libraries. These libraries should provide the reactor analyst with a set of cross sections sufficient for the analysis of LWRs, CANDUs (or HTGRs) or LMFBRs. All were created between 1977 and 1980 for use in the ORIGEN update effort of the Chemical Technology Division at Oak Ridge National Laboratory and have been used successfully in the analysis of the above three reactor types.[1-4] For each reactor type, two libraries were created. The first contains the fuel nuclides, structural materials, moderators and poisons, while the second contains the major fission products. The point cross sections were obtained from a variety of sources to ensure the completeness of the final library and to include the most recent data at the time of processing. These libraries are available through the Radiation Shielding Information Center (RSIC)*.

Most of the cross sections in these libraries were generated with the NPTXS and XLACS2 modules of the AMPX[5] system, while the NJOY[6] computer code was used for some nuclides of the LMFBR library. All of the libraries were generated from base point cross section sources, usually ENDF/B-IV[7] or ENDF/B-V[8], which were considered the best available at the time of the library's creation. They are very general multigroup libraries in the AMPX master interface format, retaining the resonance parameters necessary for application to a broad design range about the basic reactor type. The compilation of AMPX modules distributed by RSIC as PSR-63/AMPX-II provides the easiest means of accessing and using these libraries. Examples of the further processing and use of these libraries are contained in Refs. 1-4.

In the remainder of this report, Section 2 describes the contents of each library, its intended uses and limitations and the parameters, such as temperature, that were used in its creation. Section 3 describes the procedures and codes which were used.

2. DESCRIPTION OF LIBRARIES

Libraries of cross sections have been created for three different reactor types -- light-water reactors (LWRs), heavy-water CANDU reactors and liquid metal fast breeder reactors (LMFBRs). For each type, there are two libraries, one consisting of the fuel nuclides,

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* Radiation Shielding Information Center, Oak Ridge National Laboratory, P. O. Box X, Oak Ridge, Tennessee 37830, phone 615-574-6176.

moderators, structural materials and poisons, and the other containing only the fission products. The former shall be referred to as the "actinide library".

The ENDF/B-IV compilation is the source of all of the cross sections in each fission product library. These libraries contain all of the 181 fission products in ENDF/B-IV that have point cross sections. For the actinide libraries, several sources of point cross sections are used for each library. This illustrates the changing nature of cross-section evaluation, with new versions of the neutronically important nuclides and first-time versions of the minor actinides continually being introduced. The influence of the recent introduction of the ENDF/B-V compilation is evident in the 126-group LMFBR library, which is composed almost entirely of nuclides from ENDF/B-V.

2.1. LIGHT-WATER REACTORS

These libraries exist in the 84-group energy structure given in Table 1. This group structure, which has been extensively used at Savannah River Laboratory (SRL) for their analysis of heavy-water reactors, contains 30 thermal energy groups with a thermal cutoff of 0.625 eV. The fine structure over a fairly narrow thermal energy range is also appropriate for LWRs, where hydrogen is the principal scattering source.

Table 2 lists the sources of data for the 54 nuclides in the actinide library. The ENDF/B-IV compilation was the source of data for most of the major nuclides, with the exception of natural Sn, ^{238}U and ^{232}Th. Natural Sn was taken from the LENDL[9] evaluation because it did not exist in the ENDF/B-IV compilation. ^{238}U, on the other hand, was derived from ENDF/B-IV but contained adjustments on the capture resonance parameters and background cross sections as supplied by SRL.[10] These adjustments were necessary to achieve agreement between calculations and experiments on heavy-water reactors at SRL. ^{232}Th was taken from an early version of ENDF/B-V supplied by M. Bhat.[11]

All of the minor actinides were processed from a preliminary version of ENDF/B-V. The remaining nuclide in Table 2 is that labeled "pure 1/V absorber". This contains simply a 1/V capture cross section normalized to 1.0 barn at .0253 eV. This standardized cross section can be useful for studying the effects of various neutron spectra on collapsed cross sections and was used to calculate the parameters THERM, RES and FAST in the ORIGEN update effort.

The fission product library for this group structure contains 180 nuclides from ENDF/B-IV -- all of those in Table 3 except for ^{153}Eu. Unlike the fission product libraries for the other two reactor types, however, this library is not complete. It was generated for a special

purpose for the ORIGEN update effort and contains only 1-D cross sections for the total and the charged particle reactions. None of the other 1-D reactions, such as elastic and inelastic scattering, are included, and none of the 2-D transfer arrays. Although this library is an improvement over previous fission product compilations which included only capture cross sections or a limited number of nuclides, it is still quite limited in applicability and should be used with caution.

All of the nuclides in these two libraries were processed with the NPTXS and XLACS2 modules of the AMPX system. With the exception of hydrogen, the point cross sections of every nuclide were Doppler-broadened to 1000 K. For hydrogen a representative coolant temperature of 550 K was used. Also, the hydrogen in the actinide library consists of free-atom data for those groups above the thermal cutoff and water-bound $S_{\alpha,\beta}$ data for the thermal groups. Except for the fission products, which have no scattering data, scattering matrices were calculated to third order. The weighting function used to generate the group constants was a 1/E spectrum tied to a Maxwellian at 5KT eV and to a fission spectrum at 67.4keV. The fission spectrum peaked at 1.27 MeV.

In the ORIGEN update effort, these two libraries have been used successfully on PWRs and BWRs, both with and without plutonium recycle[1], and on alternative fuel cycle PWRs[2]. The latter study includes PWRs based on a thorium fuel cycle and on a high-burnup uranium fuel cycle. At the time of their creation (1978), these libraries were composed of the most recent reliable data. Since that time, the final version of ENDF/B-V has been released, which contains many of the major nuclides in these libraries. This new data, however, has undergone very little testing against thermal reactors and so has not been used to update the 84-group libraries. Comparisons with recent experimental measurements[12] have shown the ENDF/B-IV-based libraries to still accurately predict LWR characteristics.

2.2. HEAVY-WATER CANDU REACTORS

The two libraries for this reactor type exist in a 27-group energy structure which contains 15 groups in the energy range below the thermal cutoff of 3.05 eV. This thermal energy range, broader than that in the 84-group structure, is necessary to account for the increased upscattering which occurs when deuterium is used as the moderator. For this same reason, these libraries could be used for the analysis of a high-temperature gas-cooled reactor (HTGR), in which graphite is the principal moderator. The group structure is listed in Table 4.

At the time of the creation of these libraries (1979), the ENDF/B-V compilation had been officially released. It was thus used for both the major fuel nuclides ($^{235,238}U$, $^{239,240}Pu$) and most of the minor actinides. As with the 84-group actinide library, the SRL adjustments to ^{238}U were used. The remaining moderators, structural materials,

poisons and minor actinides were taken from a 218-group library[13] created from ENDF/B-IV and intended for criticality safety studies. (The only exception to this was natural Zr, which was taken from the LENDL[9] evaluation.) Table 5 is a list of the 59 nuclides included in the actinide library and the source of their point cross sections.

Unlike the 84-group fission product library, this one is a complete AMPX master interface, with all of the data from ENDF/B-IV retained. All of the 181 nuclides listed in Table 3 are included.

As with the libraries in the 84-group structure, the NPTXS and XLACS2 modules of the AMPX system were used for all cross section processing. All of the nuclides taken from ENDF/B-V were broadened to a temperature of 1000 K, scattering matrices were computed to third order, and a Maxwellian-1/E-fission spectrum was used as a weighting function. The join points on the weighting function were 5KT and 67.4keV, and the fission spectrum peak was at 1.27MeV. For those nuclides taken from the 218-group library, all except hydrogen and deuterium were broadened to 293 K. Hydrogen and deuterium, both with water-bound $S_{\alpha,\beta}$ data in the thermal energy range, were broadened to two temperatures, 293 K and 500 K. For all nuclides heavier than ^{27}Al, the weighting function was a Maxwellian-1/E-fission spectrum with join points of 5KT and 100 keV and a fission spectrum peak of 1.27 MeV. For ^{27}Al and all lighter nuclides, the weighting function was a Maxwellian-$1/E\sigma_t$-fission spectrum, with join points at 5KT and 1.4 MeV and a fission spectrum peak at 1.27 MeV. As always, P_3 scattering was computed.

A CANDU reactor using natural uranium and one using slightly enriched uranium have been analyzed in the ORIGEN update effort[3] with these libraries. The calculated values were in good agreement with the discharge concentration values given in the literature and supplied by Atomic Energy of Canada, Limited.

2.3. LIQUID METAL FAST BREEDER REACTORS

The LMFBR libraries use the 126-group energy structure listed in Table 6, a fast reactor structure with only a few groups in the thermal energy range. Unlike the libraries described in Sections 2.1 and 2.2, which were generated entirely with modules of the AMPX system, some nuclides of the actinide library for this group structure were generated with the NJOY[6] computer code.

Due to the growing acceptance of the final released version of ENDF/B-V, this actinide library, which was the last to be created (January 1980), is comprised almost entirely of nuclides from ENDF/B-V. The NPTXS and XLACS2 modules of the AMPX system were used for the processing of all nuclides except ^{233}U and ^{241}Pu. These two nuclides, which contain resonance parameters in the Adler-Adler format that the AMPX system cannot handle, were processed with the NJOY

code. Table 3 lists the 60 nuclides included in the actinide library. Because the ENDF/B-V release contains only a few of the fission products included in Table 7, ENDF/B-IV remained the sole source for fission products. The 126-group fission product library contains all 181 nuclides listed in Table 7 and is, like the 27-group library, a complete library, containing all the cross section data from the ENDF file.

A temperature of 1500 K was used in the Doppler-broadening of all the actinides and fission products and a few of the minor nuclides, while a temperature of 700 K was used for the moderator, structural materials and poisons. Since there is no water in an LMFBR, the hydrogen cross sections contain no water-bound thermal data. P_3 scattering was computed for all nuclides and the pure 1/V absorber, broadened to 700 K, is included. For those nuclides processed with the NJOY code, self-shielding factors were generated at a temperature of 1500 K and a σ_0 value of infinity.

For the analyses described in Refs. 1-3, using the 84- and 27-group libraries, the neutronically important nuclides of the actinide library were used in reactor analysis codes to determine a characteristic neutron spectrum. This spectrum was then used to collapse both the actinide and fission product libraries to a one-group structure for use in the ORIGEN2 code. For the LMFBR analysis, however, this neutron spectrum was calculated using a cross-section library that was derived chiefly from ENDF/B-IV. The ENDF/B-V actinide library documented in this report, which was compiled after much of the reactor analysis work had already been done, was then collapsed to one group and used in ORIGEN2. Despite the rather hybrid nature of this method, the ENDF/B-V 1-group cross sections worked well in the calculation of discharge concentrations.[4]

3. PROCESSING CODES AND PROCEDURES

All of the cross section sets have been created in the AMPX master library format described in Appendix A. This format includes the resonance parameters or self-shielding factors (if any) for a nuclide as well as the infinite dilution values of the group cross sections. As such it is quasi-problem-independent. A single library would be adequate for a broad class of reactors, such as LWRs, and more specific problem-dependent data, such as nuclide concentrations, would be required to create a strictly problem-dependent library.

For those nuclides which were processed through the AMPX system, the NPTXS and XLACS2 modules were used. For a resonance nuclide, NPIXS takes the resonance parameters from an ENDF/B file and reconstructs the resonances by calculating the cross section on an energy grid spanning the resolved and unresolved resonance regions. Background cross sections from file 3 of ENDF/B are added and an output file is generated

which has only the total, elastic scattering, fission and radiative capture cross sections and whose format is identical to an ENDF/B file 3. The data are Doppler-broadened to a user-specified temperature. The data in the unresolved resonance range are evaluated at an input value of the parameter σ_0 that is indicative of the system in which the cross sections will be used. (σ_0 is the total cross section in the surrounding medium.)

The purpose of XLACS2 is to create an AMPX master library which contains all of the s-wave resonance parameters (if any), the flux-averaged group cross sections and the group-to-group scattering matrix. Its input is the ENDF/B file for the nuclide and, if it is a resonance nuclide, the NPTXS-created file of point cross sections. The point elastic scattering cross sections from NPTXS are used to generate the scattering matrix, and all cross sections are group-averaged with an input weighting function.

For nuclides which are analyzed with the NJOY code, an AMPX master library is created which has somewhat different information. The resonances are reconstructed in the resolved and unresolved resonance regions in a manner similar to NPTXS, but are evaluated for various values of the temperature (T) and background cross section (σ_0). Thus, the master library created by NJOY contains group-averaged cross sections for infinite dilution, zero temperature ($\sigma_0=\infty$, T=0), and a set of self-shielding factors for the various T,σ combinations. The self-shielding factors are simply the ratio of the group-averaged cross section at T,σ_0 to that at T=0, $\sigma_0=\infty$. For the two nuclides in the LMFBR actinide library which were processed with NJOY, only one T,σ_0 pair was used, namely 1500 K and infinite dilution. Since the resonance parameters were not saved, these nuclides appear to subsequent codes as nonresonance nuclides.

References 1-4 contain examples of how these master cross-section libraries may be used in the neutronic analysis of a reactor. This is accomplished chiefly with the NITAWL and XSDRNPM modules of the AMPX system. Given the concentrations of the resonance nuclides in a fuel element of the reactor and the resonance parameters stored on the master library, NITAWL employs the Nordheim integral treatment to compute the multigroup cross sections for the particular reactor configuration. Cross sections for the nonresonance nuclides are simply copied over from the master library and a problem-dependent working library is created. This working library is then used in XSDRNPM, an S_n code that accounts for spatial and energy self-shielding effects within the fuel element. XSDRNPM can use the neutron flux it calculates to collapse the working library to a few-group form such as would be needed in a subsequent fuel depletion code. The same flux could be used to collapse the fission product library down to some useful group structure.

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Appendix A

AMPX MASTER INTERFACE FORMAT*

The AMPX multigroup library formats have been designed with a generality paralleling that of the ENDF/B point data libraries. For example:

1. The formats will allow for neutron libraries, gamma libraries, or coupled neutron-gamma libraries.

2. Resonance parameters can be passed.

3. Any number of reaction cross sections is allowed. ENDF/B identifications are retained where possible.

4. Any scattering process can be represented anisotropically to any order.

5. Any process can have a transfer matrix. For example, XLACS produces a transfer matrix for each inelastic level.

6. Temperature dependence is allowed on thermal scattering kernels.

Special arrays designated "magic-word" arrays are used to compact the potentially very lengthy transfer arrays on the interfaces. These formats effectively eliminate zero and impossible elements from the transfer arrays.

Four types of arrays are used in the formats:

1. The aforementioned "magic-word" array,

2. Temperature-dependent 1-D arrays,

3. Temperature-independent 1-D arrays, and

4. The resonance parameter array.

A.1. MAGIC-WORD ARRAY

The structure of a magic-word array is as follows:

1. length of magic-word string which follows,

2. magic word for first nonzero group (note that this is not necessarily the first energy group),

---

*This material was taken from Section 11.3 of Ref. 5.

3. string of transfer elements for this group,

4. magic word for next group,

5. string for next group,

   .
   .
   .
   .

   etc., etc., until the length specified in the first word is satisfied.

A read statement for this array would contain a list:

```
L,(X(I),I=1,L)
```

A magic word is a 9-digit integer consisting of three 3-digit integers (IIIJJJKKK). The number of the group scattered into is KKK. III is the lowest numbered (highest energied) group which scatters to KKK. JJJ is the highest numbered group which scatters to KKK. After the magic word, the transfer string to group KKK is in reverse order:

IIIJJJKKK

σ(JJJ→KKK)

σ(JJJ-1→KKK)

   .
   .
   .
   .

σ(KKK→KKK)

σ(KKK-1→KKK)

   .
   .
   .
   .

σ(III→KKK)

Note that the within-group term does not necessarily fall in the string.

A.2. TEMPERATURE-DEPENDENT 1-D ARRAYS

These arrays are structured as follows:

1. ENDF/B Process ID (MT),
2. Temperature in °K,
3. Average cross section for each energy group,
4. etc., etc.

A.3. TEMPERATURE-INDEPENDENT 1-D ARRAYS

Temperature-independent 1-D arrays are structured:

1. ENDF/B Process ID for first process,
2. Average cross sections for second process for all groups,
3. ENDF/B Process ID for second process,
4. Average cross sections for second process for all groups,
5. etc., etc.

A.4. RESONANCE ARRAY STRUCTURE

The makeup of this data array is as follows:

1. Mass ratio (A) for the isotope.
2. σ_{po}, potential scattering cross section.
3. Average statistical factor, g, in the unresolved region.
4. Number of resolved resonances.
5. s-factor. This factor is used to determine the region over which the Nordheim Integral Treatment is applied for a resolved resonance.
6. Average energy level spacing, $\langle D \rangle$, for the $\ell = 0$ unresolved sequence passed.

7. $\langle \Gamma_n^o \rangle$, average unresolved neutron width.

8. $\langle \Gamma_\gamma \rangle$, average unresolved gamma width.

9. $\langle \Gamma_f \rangle$, average unresolved fission width.

10. Energy of first resolved resonance

11. Γ_n, neutron width of first resolved resonance.

12. Γ_γ, gamma width of first resonance.

13. Γ_f, fission width of first resonance.

14. r-factor (used in Nordheim Treatment).

15. Statistical factor, g.

16. }
17. }
18. } As for 10-15 for second resolved resonance.
19. }
20. }
21. }

. . . . .

10+6(number of resolved resonances) - Energies at which to evaluate the unresolved "averaged" cross sections - low-to-high (eV).

These energies will span the unresolved energy range.

. . . . .

A.5. MASTER INTERFACE SPECIFICATION

The format of the master cross-section interface is specified below. Reference is made to the array definitions made immediately prior to this section.

Record	Length (Words)	Contents
1	110	IDT - Interface identification number NNUC - Number of nuclides in this interface NG - Number of neutron energy groups IFTG - First thermal neutron group MSN - Zero I1 - Number of gamma energy groups I2 - Zero I3 - Zero I4 - Zero I5 - Zero A(1→100) Hollerith information describing the interface
2	50	ID(1→50) Information describing the first nuclide on the interface (See the description of record R1 below for a specification of the ID array.)
NNUC+1	50	As for record 2 for NNUCth nuclide

NNUC+2	2x(NG+1)	Neutron energy group boundaries (high-to-low in eV) followed by corresponding lethargy boundaries.
NNUC+3	2x(I1+1)	Gamma energy group boundaries (high-to-low in eV) followed by corresponding lethargy boundaries.

NOTE: Records 2 through NNUC+1 constitute a Table of Contents for the interface. Record NNUC+2 or NNUC+3 is omitted when NG=0 or I1=0, respectively.

The following set of records is repeated NNUC times, one nuclide after the other.

Record	Length	Contents
R1	50	ID(1-18) - Hollerith information describing the nuclide
		ID(19) - Nuclide identification number
		ID(20) - Number of resolved resonances
		ID(21) - Number of points at which to evaluate unresolved "averaged" cross sections
		ID(22) - Number of one-dimensional neutron arrays (temperature-independent)
		ID(23) - Number of two-dimensional neutron processes; i.e., a process which requires a neutron-neutron transfer array
		ID(24) - Number of temperature-dependent one-dimensional neutron arrays
		ID(25) - Number of one-dimensional gamma arrays
		ID(26) - Number of two-dimensional gamma processes
		ID(27) - Number of neutron-to-gamma production processes
		ID(28) - Not used

Record	Length	Contents
		ID(29) - Mass number (neutron equivalent)
		ID(30) - ZA
		ID(31) - Neutron (XLACS) weighting option
		ID(32) - Identifier of neutron weighting
		ID(33) - Gamma (SMUG) weighting option
		ID(34) - Power per fission (watt-sec/fission)
		ID(35) - Energy released per capture (watt-sec/fission)
		ID(36) - Zero
		ID(37) - Number of processes with Bondarenko factors
		ID(38) - Number of σ_0's
		ID(39) - Number of temperatures
		ID(40) - Maximum number of groups with Bondarenko factors
		ID(41) - Zero
		ID(42) - Identifier of gamma production weighting function
		ID(43) - Zero
		ID(44) - Gamma production (LAPHNGAS) weighting option
		ID(45) - ENDF material number for fast neutron data
		ID(46) - ENDF material number for thermal neutron data
		ID(47) - ENDF material number for gamma data
		ID(48) - ENDF material number for gamma production data

ID(49) - Standard CITATION identification number

ID(50) - Number of records for this nuclide

Record	Length	Contents
R2	ID(38)+ID(39)+2	$(\sigma_0(i),i=1,ID(38))$, $(T(j),j=1,ID(39))$, ELØ,EHI where ELØ and EHI define the range over where the Bondarenko factors apply
R3	6 * ID(37)	$(MT_i,i=1,ID(37))$, $(NF_i,i=1,ID(37))$, $(NL_i,i=1,ID(37))$, $(NX_i,i=1,ID(37))$, $(NY_i,i=1,ID(37))$, $(NZ_i,i=1,ID(37))$, where MT is the MT number of the process, NF is the first group with Bondarenko factors for the process, NL is the last group with Bondarenko factors, and NX, NY, NZ are zeroes, presently.

The following two records are repeated ID(37) times:

R4	NL_i-NF_i+1	$(\sigma_{MT}(i),i=NF,NL)$ Infinite dilution values for the cross section of process MT
R5	$(NL_i-NF_i+1)*ID(38)*ID(39)$	$(((BF(i,j,k),i=1,ID(38)),$ $j=1,ID(39)),k=NF,NL)$
R6	6xID(20)+9+ID(21)	Resonance data array
R7	ID(22)x(NG+1)	Temperature-independent one-dimensional arrays
R8	ID(24)x(NG+2)	Temperature dependent one-dimensional arrays

R9	ID(23)x4	(MTX(I), I = 1, ID(23)), (LX(I), I = 1, ID(23)), (NLX(I), I = 1, ID(23)), (NTX(I), I = 1, ID(23)), where MTX = the ENDF/B process identification (MT), LX = the maximum length of a single matrix for the ith process, NLX = the expansion order of the cross sections for the ith process, NTX = the number of temperatures at which the cross section for the process is evaluated

The following arrays are repeated for each two-dimensional neutron process, through ID(23) processes.

R10(a)	T(i)	The temperature (eV) at which the transfer arrays are given for the MT_i process NOTE: NT = 0 will not require this array
R10(b)	L(i)max	The P_0 array for the MT_i process at T_1, written (X(I), I = 1, LX)
R10(b+1)	L(i)max	The P_1 array for the MT_i process at T_1
	.	
	.	
	.	
	.	
R10(b+NL)	L(i)max	The P_{NL_i} array for the MT_i process at T_1
	.	
	.	
	.	
	.	
R10( )	Li(i)max	The P_0 array for the MT_i process at T_2
R10( )	Li(i)max	The P_{NL_i} array for the MT_i process at T_{NT_i}

****Repeat this pattern until all neutron processes are exhausted.

Gamma Production Arrays

P1	ID(27)x4	(MTY_i, i = 1, ID(27)), (LY_i, i = 1, ID(27)), (NLY_i, i = 1, ID(27)), (NNY_i, i = 1, ID(27)), where MTY_i, LY_i, and NLY_i have meanings as stated in record R9. NNY_i is a data type identifier. An NNY_i of zero identifies the production arrays for the ith process as fractional yield data. A value of one (1) designates that the arrays are in cross section units.

The following records are repeated for ID(27) processes.

P2	L(i)max	P_0 neutron to gamma transfer matrix, written L, (X(I), I = 1, L)
.		.
.		.
.		.
.		.
$P(2+NL_i)$		P_{NL} neutron to gamma transfer matrix
.		
.		
.		
.		

****Repeat until the two-dimensional gamma processes are exhausted.

Gamma Arrays

G1	ID(25)x(I1+1)	One-dimensional gamma interaction arrays
G2	ID(26)x4	(MTZ_i, i = 1, ID(26)), (LZ_i, i = 1, ID(26)), (NLZ_i, i = 1, ID(26)), (NTZ_i, i = 1, ID(26)) [***NT_i will always have all zeroes] These terms are defined as for record R9 except that they apply to gammas.

Repeat the following records for each process.

G3	L(i)	The P_0 array for the MTZ_i gamma process, written (X(I), I = 1, LZ)
	.	
	.	
	.	
$G(3+NL_i)$	L(i)	The P_{NLZ_i} array for the MTZ_i gamma process

****This concludes the data for a nuclide.

TABLE 1. 84-GROUP LWR STRUCTURE

GROUP	UPPER ENERGY BOUNDARY (EV)	GROUP	UPPER ENERGY BOUNDARY (EV)	GROUP	UPPER ENERGY BOUNDARY (EV)
1	1.00000E+07	29	1.23410E+03	57	5.11187E-01
2	7.78801E+06	30	7.48518E+02	58	4.14983E-01
3	6.06531E+06	31	4.53999E+02	59	3.38908E-01
4	4.72367E+06	32	2.75364E+02	60	2.78866E-01
5	3.67879E+06	33	1.67017E+02	61	2.31511E-01
6	2.86505E+06	34	1.30073E+02	62	1.94124E-01
7	2.23130E+06	35	1.01301E+02	63	1.64513E-01
8	1.73774E+06	36	7.88932E+01	64	1.40911E-01
9	1.35335E+06	37	6.14421E+01	65	1.21896E-01
10	1.05399E+06	38	4.78512E+01	66	1.06323E-01
11	8.20850E+05	39	3.72665E+01	67	9.22968E-02
12	6.39279E+05	40	2.90232E+01	68	8.01604E-02
13	4.97871E+05	41	2.26033E+01	69	6.97166E-02
14	3.87742E+05	42	1.76035E+01	70	6.07832E-02
15	3.01974E+05	43	1.37096E+01	71	5.31932E-02
16	2.35177E+05	44	1.06770E+01	72	4.61091E-02
17	1.83156E+05	45	8.31529E+00	73	3.95312E-02
18	1.42642E+05	46	6.47595E+00	74	3.34592E-02
19	1.11090E+05	47	5.04348E+00	75	2.78932E-02
20	8.65169E+04	48	3.92786E+00	76	2.28332E-02
21	6.73794E+04	49	3.05902E+00	77	1.82792E-02
22	4.08677E+04	50	2.38237E+00	78	1.42312E-02
23	2.47875E+04	51	1.85539E+00	79	1.06892E-02
24	1.50344E+04	52	1.43950E+00	80	7.65324E-03
25	9.11882E+03	53	1.12535E+00	81	5.12325E-03
26	5.53084E+03	54	8.33691E-01	82	3.09925E-03
27	3.35463E+03	55	7.84930E-01	83	1.58125E-03
28	2.03468E+03	56	6.32500E-01	84	5.69249E-04

THE BOTTOM OF ENERGY GROUP 84 IS 1.E-5

TABLE 2. NUCLIDES IN 84-GROUP ACTINIDE LIBRARY

NUCLIDE	IDENTIFIER	SOURCE REFERENCE	NUCLIDE	IDENTIFIER	SOURCE REFERENCE
H-1	10010	7	NP-238	932380	8
B-10	50100	7	PU-236	942360	8
B-11	50110	7	PU-238	942380	8
C-12	60120	7	PU-239	942390	7
N-14	70140	7	PU-240	942400	7
O-16	80160	7	PU-241	942410	7
NA	110230	7	PU-242	942420	8
CR	240520	7	PU-243	942430	8
MN	250550	7	AM-241	952410	8
FE	260560	7	AM-242	952420	8
NI	280580	7	AM-242M	952421	8
ZIRCALOY-2	400990	7	AM-243	952430	8
NB-93	410930	7	CM-242	962420	8
MO	420910	7	CM-243	962430	8
NAT. SN	501190	9	CM-244	962440	8
TH-230	902300	8	CM-245	962450	8
TH-232	902320	11	CM-246	962460	8
PA-231	912310	8	CM-247	962470	8
PA-233	912330	7	CM-248	962480	8
U-232	922320	8	BK-249	972490	8
U-233	922330	7	CF-249	982490	8
U-234	922340	8	CF-250	982500	8
U-235	922350	7	CF-251	982510	8
U-236	922360	8	CF-252	982520	8
U-237	922370	8	CF-253	982530	8
U-238	922380	7,10	ES-253	992530	8
NP-237	932370	8	1/V ABS.	999990	-

TABLE 3. NUCLIDES IN FISSION PRODUCT LIBRARY

NUCLIDE	IDENTIFIER	NUCLIDE	IDENTIFIER	NUCLIDE	IDENTIFIER
GE-72	320720	ZR-91	400910	PD-107	461070
GE-73	320730	ZR-92	400920	PD-108	461080
GE-74	320740	ZR-93	400930	PD-110	461100
GE-76	320760	ZR-94	400940	AG-107	471070
AS-75	330750	ZR-95	400950	AG-109	471090
SE-76	340760	ZR-96	400960	AG-111	471110
SE-77	340770	NB-93	410930	CD-108	481080
SE-78	340780	NB-94	410940	CD-110	481100
SE-80	340800	NB-95	410950	CD-111	481110
SE-82	340820	MO-94	420940	CD-112	481120
BR-79	350790	MO-95	420950	CD-113	481130
BR-81	350810	MO-96	420960	CD-114	481140
KR-80	360800	MO-97	420970	CD-115M	481151
KR-82	360820	MO-98	420980	CD-116	481160
KR-83	360830	MO-99	420990	IN-113	491130
KR-84	360840	MO-100	421000	IN-115	491150
KR-85	360850	TC-99	430990	SN-115	501150
KR-86	360860	RU-99	440990	SN-116	501160
RB-85	370850	RU-100	441000	SN-117	501170
RB-86	370860	RU-101	441010	SN-118	501180
RB-87	370870	RU-102	441020	SN-119	501190
SR-86	380860	RU-103	441030	SN-120	501200
SR-87	380870	RU-104	441040	SN-122	501220
SR-88	380880	RU-105	441050	SN-123	501230
SR-89	380890	RU-106	441060	SN-124	501240
SR-90	380900	RH-103	451030	SN-125	501250
Y-89	390890	RH-105	451050	SN-126	501260
Y-90	390900	PD-104	461040	SB-121	511210
Y-91	390910	PD-105	461050	SB-123	511230
ZR-90	400900	PD-106	461060	SB-124	511240

TABLE 3. CONTINUED

NUCLIDE	IDENTIFIER	NUCLIDE	IDENTIFIER	NUCLIDE	IDENTIFIER
SB-125	511250	BA-134	561340	SM-148	621480
SB-126	511260	BA-135	561350	SM-149	621490
TE-122	521220	BA-136	561360	SM-150	621500
TE-123	521230	BA-137	561370	SM-151	621510
TE-124	521240	BA-138	561380	SM-152	621520
TE-125	521250	BA-140	561400	SM-153	621530
TE-126	521260	LA-139	571390	SM-154	621540
TE-127M	521271	LA-140	571400	EU-151	631510
TE-128	521280	CE-140	581400	EU-152	631520
TE-129M	521291	CE-141	581410	EU-153	631530
TE-130	521300	CE-142	581420	EU-154	631540
TE-132	521320	CE-143	581430	EU-155	631550
I-127	531270	CE-144	581440	EU-156	631560
I-129	531290	PR-141	591410	EU-157	631570
I-130	531300	PR-142	591420	GD-154	641540
I-131	531310	PR-143	591430	GD-155	641550
I-135	531350	ND-142	601420	GD-156	641560
XE-128	541280	ND-143	601430	GD-157	641570
XE-129	541290	ND-144	601440	GD-158	641580
XE-130	541300	ND-145	601450	GD-160	641600
XE-131	541310	ND-146	601460	TB-159	651590
XE-132	541320	ND-147	601470	TB-160	651600
XE-133	541330	ND-148	601480	DY-160	661600
XE-134	541340	ND-150	601500	DY-161	661610
XE-135	541350	PM-147	611470	DY-162	661620
XE-136	541360	PM-148	611480	DY-163	661630
CS-133	551330	PM-148M	611481	DY-164	661640
CS-134	551340	PM-149	611490	HO-165	671650
CS-135	551350	PM-151	611510	ER-166	681660
CS-136	551360	SM-147	621470	ER-167	681670
CS-137	551370				

TABLE 4. 27-GROUP CANDU STRUCTURE

GROUP	UPPER ENERGY BOUNDARY (EV)
1	2.00000E+07
2	6.43400E+06
3	3.00000E+06
4	1.85000E+06
5	1.40000E+06
6	9.00000E+05
7	4.00000E+05
8	1.00000E+05
9	1.70000E+04
10	3.00000E+03
11	5.50000E+02
12	1.00000E+02
13	3.00000E+01
14	1.00000E+01
15	3.05000E+00
16	1.77000E+00
17	1.30000E+00
18	1.13000E+00
19	1.00000E+00
20	8.00000E-01
21	4.00000E-01
22	3.25000E-01
23	2.25000E-01
24	1.00000E-01
25	5.00000E-02
26	3.00000E-02
27	1.00000E-02

THE BOTTOM OF ENERGY GROUP 27 IS 1.E-5

TABLE 5. NUCLIDES IN 27-GROUP ACTINIDE LIBRARY

NUCLIDE	IDENTIFIER	SOURCE REFERENCE	NUCLIDE	IDENTIFIER	SOURCE REFERENCE
H-1	10010	7	PU-236	942360	8
H-2	10020	7	PU-237	942370	8
LI-6	30060	7	PU-238	942380	7
LI-7	30070	7	PU-239	942390	8
B-10	50100	7	PU-240	942400	8
B-11	50110	7	PU-241	942410	7
C-12	60120	7	PU-242	942420	7
N-14	70140	7	PU-243	942430	8
O-16	80160	7	PU-244	942440	8
AL-27	130270	7	AM-241	952410	8
CR	240000	7	AM-242	952420	8
FE	260000	7	AM-242M	952421	8
CO-59	270590	7	AM-243	952430	8
NI	280000	7	CM-241	962410	8
NAT. ZR	400000	9	CM-242	962420	8
ZIRCALOY-2	403020	7	CM-243	962430	8
NB-93	410930	7	CM-244	962440	8
TH-230	902300	8	CM-245	962450	8
TH-232	902320	8	CM-246	962460	8
PA-231	912310	8	CM-247	962470	8
PA-233	912330	8	CM-248	962480	8
U-232	922320	8	BK-249	972490	8
U-233	922330	8	CF-249	982490	8
U-234	922340	7	CF-250	982500	8
U-235	922350	8	CF-251	982510	8
U-236	922360	7	CF-252	982520	8
U-237	922370	8	CF-253	982530	8
U-238	922380	8,10	ES-253	992530	8
NP-237	932370	7	1/V ABS.	999	-
NP-238	932380	8			

TABLE 6. NUCLIDES IN 126-GROUP ACTINIDE LIBRARY

NUCLIDE	IDENTIFIER	SOURCE REFERENCE	NUCLIDE	IDENTIFIER	SOURCE REFERENCE
H-1	10010	8	NP-238	932380	8
B-10	50100	8	PU-236	942360	8
B-11	50110	8	PU-237	942370	8
C-12	60120	8	PU-238	942380	8
N-14	70140	8	PU-239	942390	8
N-15	70150	8	PU-240	942400	8
O-16	80160	8	PU-241	942410	8
O-17	80170	8	PU-242	942420	8
NA-23	110230	8	PU-243	942430	8
CR	240000	8	PU-244	942440	8
MN-55	250550	8	AM-241	952410	8
FE	260000	8	AM-242	952420	8
CO-59	270590	8	AM-242M	952421	8
NI	280000	8	AM-243	952430	8
ZIRCALOY-2	403020	7	CM-241	962410	8
NB-93	410930	8	CM-242	962420	8
MO	420000	8	CM-243	962430	8
SN	500000	9	CM-244	962440	8
TH-230	902300	8	CM-245	962450	8
TH-232	902320	8	CM-246	962460	8
PA-231	912310	8	CM-247	962470	8
PA-233	912330	8	CM-248	962480	8
U-232	922320	8	BK-249	972490	8
U-233	922330	8	CF-249	982490	8
U-234	922340	8	CF-250	982500	8
U-235	922350	8	CF-251	982510	8
U-236	922360	8	CF-252	982520	8
U-237	922370	8	CF-253	982530	8
U-238	922380	8	ES-253	992530	8
NP-237	932370	8	1/V ABS.	999990	-

TABLE 7. 126-GROUP LMFBR STRUCTURE

GROUP	UPPER ENERGY BOUNDARY (EV)	GROUP	UPPER ENERGY BOUNDARY (EV)	GROUP	UPPER ENERGY BOUNDARY (EV)
1	1.73330E+07	29	1.35340E+06	57	2.87250E+05
2	1.22140E+07	30	1.28730E+06	58	2.73240E+05
3	1.00000E+07	31	1.22460E+06	59	2.47240E+05
4	8.18730E+06	32	1.16480E+06	60	2.23710E+05
5	6.70320E+06	33	1.10800E+06	61	2.12800E+05
6	6.06530E+06	34	1.00260E+06	62	2.02420E+05
7	5.48810E+06	35	9.61640E+05	63	1.92550E+05
8	4.49330E+06	36	9.07180E+05	64	1.83160E+05
9	3.67880E+06	37	8.62940E+05	65	1.74220E+05
10	3.16640E+06	38	8.20850E+05	66	1.65730E+05
11	3.01190E+06	39	7.80820E+05	67	1.57640E+05
12	2.86500E+06	40	7.42740E+05	68	1.49960E+05
13	2.72530E+06	41	7.06510E+05	69	1.42640E+05
14	2.59240E+06	42	6.72060E+05	70	1.35690E+05
15	2.46600E+06	43	6.39280E+05	71	1.29070E+05
16	2.38520E+06	44	6.08100E+05	72	1.22770E+05
17	2.36530E+06	45	5.78440E+05	73	1.16790E+05
18	2.30690E+06	46	5.50230E+05	74	1.11090E+05
19	2.23130E+06	47	5.23400E+05	75	9.80370E+04
20	2.12250E+06	48	4.97870E+05	76	8.65170E+04
21	2.01900E+06	49	4.50490E+05	77	8.25000E+04
22	1.92050E+06	50	4.07620E+05	78	7.95000E+04
23	1.82680E+06	51	3.68830E+05	79	7.20000E+04
24	1.73770E+06	52	3.33730E+05	80	6.73790E+04
25	1.65300E+06	53	3.01970E+05	81	5.65620E+04
26	1.57240E+06	54	2.98500E+05	82	5.24750E+04
27	1.49570E+06	55	2.97200E+05	83	4.63090E+04
28	1.42270E+06	56	2.94520E+05	84	4.08680E+04

TABLE 7. 126-GROUP LMFBR STRUCTURE

GROUP	UPPER ENERGY BOUNDARY (EV)	GROUP	UPPER ENERGY BOUNDARY (EV)	GROUP	UPPER ENERGY BOUNDARY (EV)
85	3.43070E+04	99	4.30740E+03	113	2.75360E+02
86	3.18280E+04	100	3.70740E+03	114	2.14450E+02
87	2.85000E+04	101	3.35460E+03	115	1.67020E+02
88	2.70000E+04	102	3.03540E+03	116	1.01300E+02
89	2.60580E+04	103	2.74650E+03	117	6.14420E+01
90	2.47880E+04	104	2.61260E+03	118	4.78510E+01
91	2.35790E+04	105	2.48520E+03	119	3.72670E+01
92	2.18750E+04	106	2.24870E+03	120	2.26030E+01
93	1.93050E+04	107	2.03470E+03	121	1.06770E+01
94	1.50340E+04	108	1.58460E+03	122	5.04350E+00
95	1.17090E+04	109	1.23410E+03	123	2.37240E+00
96	9.11880E+03	110	9.61120E+02	124	1.12540E+00
97	7.10170E+03	111	7.48520E+02	125	4.13990E-01
98	5.53080E+03	112	4.54000E+02	126	1.00000E-01

THE BOTTOM OF ENERGY GROUP 126 IS 1.E-5