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ANSL-V: ENDF/B-V BASED MULTIGROUP CROSS-SECTION LIBRARIES FOR ADVANCED NEUTRON SOURCE (ANS) REACTOR STUDIES

W. E. Ford, III*
J. W. Arwood*
N. M. Greene*
D. L. Moses***
L. M. Petrie*
R. T. Primm, III**
C. O. Slater**
R. M. Westfall*
R. Q. Wright*

Computing and Telecommunications Division*
Engineering Physics and Mathematics Division**
Engineering Technology Division***

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ABSTRACT

Pseudo-problem-independent, multigroup cross-section libraries were generated to support Advanced Neutron Source (ANS) Reactor design studies. The ANS is a proposed reactor which would be fueled with highly enriched uranium and cooled with heavy water. The libraries, designated ANSL-V (Advanced Neutron Source Cross Section Libraries based on ENDF/B-V), are data bases in AMPX master format for subsequent generation of problem-dependent cross-sections for use with codes such as KENO, ANISN, XSDRNPM, VENTURE, DOT, DORT, TORT, and MORSE. Included in ANSL-V are 99-group and 39-group neutron, 39-neutron-group 44-gamma-ray-group secondary gamma-ray production (SGRP), 44-group gamma-ray interaction (GRI), and coupled, 39-neutron group 44-gamma-ray group (CNG) cross-section libraries. The neutron and SGRP libraries were generated primarily from ENDF/B-V data; the GRI library was generated from DLC-99/HUGO data, which is recognized as the ENDF/B-V photon interaction data. Modules from the AMPX and NJOY systems were used to process the multigroup data. Validity of selected data from the fine- and broad-group neutron libraries was satisfactorily tested in performance parameter calculations.

1.0 INTRODUCTION

Research and development for the Advanced Neutron Source (ANS) Reactor is being funded by the U.S. Department of Energy Office of Basic Energy Sciences. The reactor is to provide the world's most intense steady-state source of neutrons for a national experimental user facility.[1.1]

Pseudo-problem-independent, comprehensive, multigroup cross-section libraries were generated to support design work on the ANS Reactor. The purpose of this report is to document the libraries.

1.1 Background

The ANS Reactor is to produce a thermal neutron flux of 5 to 10 $\times$ 10^{19} neutrons $m^{-2}s^{-1}$ for beam experiments. The reactor has been described as follows:[1.2]

> *"The ANS Reactor is to be fueled with uranium silicide and is to be moderated and cooled with heavy-water. The reactor core is to have a cylindrical geometry with two annular fuel sections similar to the ORNL High Flux Isotope Reactor (HFIR). Fuel is to be clad with the same aluminum alloy cladding material used on the HFIR fuel.*
>
> *A cold source of liquid deuterium at a temperature of 25 °K is proposed for the ANS reflector. Heat load imposed on the liquid deuterium by radiation from the nearby core is to be minimized by appropriate structural materials and, if necessary, by gamma shielding. Plans are to locate the cold sources in a region where the thermal flux is approximately 2 to 3 $\times$ 10^{19} neutrons $m^{-2}s^{-1}$.*
>
> *A hot source is to provide neutrons of a higher energy than found in the reflector. The source is currently envisioned as being a block of graphite located in the reflector and heated to 2000 °K."*

Specifications for the cross-section libraries described herein were defined to support ANS Reactor studies. For example, materials in the libraries are those expected to be needed for ANS hot and cold source, core, and shielding design; energy group structures for the libraries were defined to conform to anticipated ANS spectra; and criteria for Bondarenko factor data were specified to meet ANS needs.

1.2 ANSL-V Cross-Section Libraries

The collection of cross-section libraries, designated ANSL-V (Advanced Neutron Source Cross Section Libraries based on ENDF/B-V), are data bases in AMPX master format.[1.3] This format is described in Appendix A.

ANSL-V includes the following fine- and broad-group libraries:

1. Fine-Group (99 energy groups) General Purpose Neutron (GPN) Library, denoted hereinafter as the ANSL-V FGGPN Library,

2. Broad-Group (39 energy groups) General Purpose Neutron Library, the ANSL-V BGGPN Library,

3. Secondary Gamma-Ray Production (SGRP) Library containing data in 39-neutron and 44-gamma-ray group structures,

4. Gamma-Ray Interaction (GRI) Library containing data in a 44-gamma-ray group structure,

5. Coupled Neutron-Gamma (CNG) Library containing 39-group neutron data from GPN; 39-neuton, 44-gamma-ray group secondary gamma-ray production data from SGRP; and 44-group gamma data from GRI.

Materials included in the ANSL-V Libraries are listed in Table 1.1. Group structures for the ANSL-V libraries are discussed in Sect. 2. Summaries of the ANSL-V GPN, SGRP, GRI, and CNG Libraries are given in Sect. 3-6, respectively.

1.3 Primary Sources of Data for ANSL-V Libraries

Neutron and secondary-gamma-ray-production data in the ANSL-V Libraries were generated primarily from evaluations in the ENDF/B-V General Purpose Library.[1.4] Where evaluations for specified materials were not available in the ENDF/B-V library, ANSL-V GPN and SGRP data sets were generated from evaluations from other ENDF-formatted libraries. Gamma-ray interaction data sets were generated from evaluations in the ENDF/B-V Photon Interaction Library as distributed in the Radiation Shielding Information Center (RSIC) DLC-99/HUGO Package of Photon Interaction Data.[1.5]

1.4 Use of ANSL-V Libraries

ANSL-V data are to be used for the subsequent generation of problem-dependent fine- and/or broad-group cross sections for a wide range of applications. Included in the applications are core and shield analyses, activation analyses after irradiation of certain elements in the reactor environment, and safety analyses.

The ANSL-V SGRP Library includes data, where available, for the generation of gamma-rays resulting from neutron-induced reactions in the ANS core, structural, and shielding regions. It also includes data for use in calculating transport of photons throughout the ANS.

Problem-dependent cross sections can be derived from ANSL-V data with the AMPX Modular Code System.[1.3] The derived data can be used with the SCALE system[1.6] and with codes such as KENO,[1.7] ANISN,[1.8] XSDRNPM,[1.9] VENTURE,[1.10] DOT,[1.11] DORT,[1.12] TORT,[1.13] MORSE,[1.14] or any computer code which uses data in the traditional multigroup cross-section table format.

Modules of the AMPX system offer a full spectrum of cross-section processing capabilities for ANSL-V data. Examples of such capabilities are energy group collapsing;

interpolating on Bondarenko factors for self-shielding of unresolved resonance data and temperature corrections; resolved resonance processing with the Nordheim integral method; combining of neutron, secondary gamma-ray production, and gamma-ray interaction cross sections into a coupled neutron-gamma library; reformatting; and editing. The AMPX system is available from the Radiation Shielding Information Center (RSIC) as PSR63/AMPX-II.*

1.5 ANSL-V Benchmark Analyses

As first order checks of selected data sets from the ANSL-V GPN Libraries and project data processing procedures and codes, various benchmarks were calculated and the results were compared against experiment. For further analyses, ANSL-V results were compared with calculations made with other selected cross-section libraries. The benchmark calculations and analyses are discussed in Sect. 7. The procedure for using ANSL-V GPN data and the SCALE System to calculate a benchmark is described in Sect. 7.

1.6 Future ANSL-V Work

A task to provide additional first-order data checks by plotting broad-group GPN data which were expected to be "important" to ANS design work and comparing the data with point-wise data was not done; and certain benchmark calculations were not completed. The uncompleted tasks are identified in Sect. 8.

1.7 Restriction on Use of ANSL-V Data

The U.S. Department of Energy has determined that ENDF/B-V will, for the present, be limited to users within the United States and AECL (Chalk River), except for the Standards, Dosimetry, Actinide (special purpose), and Fission Product Files, which are available to everyone. This limitation also applies to information derived from the ENDF/B-V library such as the ANSL-V libraries. Recipients of ANSL-V data are responsible for ensuring that the DOE restrictions are met.

*Radiation Shielding Information Center, Oak Ridge National Laboratory, P.O. Box 2008, Bldg. 6025, Oak Ridge, TN 37831-6362.

Table 1.1. Materials in ANSL-V libraries

Nuclide	ANSL-V Data Sets IDs			Remarks[d]
	GPN[a]	SGRP[b]	GRI[c]	
Ag-107	47107	7845	47	core structural material spectra
Ag-107	471071	7845	47	fuel region spectra
Ag-109	47109	7846	47	core structural material spectra
Ag-109	471091	7846	47	fuel region spectra
Al-27	13027	1313	13	core structural material spectra
Al-27	130271	1313	13	fuel region spectra
Am-241	952411			
Am-242	952420			
Am-242m	952421			
Am-243	952431			
B-10	5010	1305	5	
B-11	5011	7811	5	
Be-9	4009	1304	4	MAT 1064 bound thermal @ 296, 400, 500, 600, 700, 800, 1000, and 1200K
Bk-249	972491			
C-12 (free gas)	6012	1306	6	
Cd	48000	7847	48	
Cd-113	481131		48	
Ce-141	581411			
Cf-249	982491			
Cf-250	982501			
Cf-251	982511			
Cf-252	982521			
Cf-253	982531			
Cm-242	962421			
Cm-243	962431			
Cm-244	962441			
Cm-245	962451			
Cm-246	962461			
Cm-247	962471			
Cm-248	962481			
Co-59	27059	1327	27	
Cr	24000	1324	24	
Cs-133	551331			
Cs-134	551341			
Cs-135	551351			
Cu	29000	1329	29	

Table 1.1. (Cont'd)

Nuclide	ANSL-V Data Sets IDs GPN[a]	SGRP[b]	GRI[c]	Remarks[d]
Es-253	992531			
Eu-151	631511	1357	63	
Eu-152	631521		63	
Eu-153	631531	1359	63	
Eu-154	631541		63	
Eu-155	631551			
Eu-nat	63000	7852	63	Core structural material spectra
Eu-nat	630001	7852	63	fuel region spectra
F-19	90190			
Fe	26000	1326	26	
Graphite	6666	1306	6	MAT 1065 graphite thermal scatt. kernel @ 296, 500, 800, 1200, and 2000K
H-1 (free gas)	1001	1301	1	
H-1-ortho	2222	1301	1	liquid H @ 14K
H-1-para	1111	1301	1	liquid H @ 14K
H-1-water bound	1801	1301	1	MAT 1002 H2O bound thermal @ 296, 350, 400, 450, 500, 600, 800, and 1000K
H-2-D20 bound	1802	1302	1	MAT 1004 D2O bound thermal @ 296, 350, 400, 450, 500, 600, 800, and 1000K
H-2-Para	3333	1302	1	para-deuterium @20K
H-3	1003		1	
He-3	2003		2	
He-4	2004		2	
Hf-nat	72000	8305	72	core structural material spectra
Hf-nat	720001	8305	72	fuel region spectra
Ir			77	
I-135	531351			
K	19000	1150	19	

Table 1.1. (Cont'd)

Nuclide	ANSL-V Data Sets IDs			Remarks
	GPN[a]	SGRP[b]	GRI[c]	
Kr-82	360821			
Kr-83	360831			
Mg	12000	1312	12	
Mn-55	25055	1325	25	
Mo	420001	1321	42	
Mo-97	420971		42	
N-14	7014	1275	7	
N-15	70150	1307	7	
Na-23	11023	1311	11	
Nd-143	601431			
Nd-145	601451			
Nd-147	601471			
Ni	28000	1328	28	
Np-237	932371			
Np-238	932381			
Np-239	932391			
O-16	8016	1276	8	
Pb	820001	1382	82	
Pm-147	611471			
Pm-148	611481			
Pm-148m	611482			
Pm-149	611491			
Pr-143	591431			
Pu-238	942381	7875	94	
Pu-239	942391	1399	94	
Pu-240	942401	1380	94	
Pu-241	942411	1381	94	
Pu-242	942421	1342	94	
Pu-243	942431			
Rh-103	451031			
Rh-105	451051			
Ru-101	441011			
Ru-103	441031			
Si	14000	1314	14	core structural material spectra
Si	140001	1314	14	fuel region spectra
Sm-149	621491		62	
Sm-150	621501			
Sm-151	621511			
Sm-152	621521			
Sm-153	621531		62	

Table 1.1. (Cont'd)

Nuclide	ANSL-V Data Sets IDs			Remarks
	GPN[a]	SGRP[b]	GRI[c]	
Sn	500001			
Ta-181	73181	1285	73	
Ta-182	73182		73	
Tc-99	430991			
Th-232	902321	1390	90	
Ti	22000	1322	22	
U-233	922331	7866	92	
U-234	922341	7867	92	
U-235	922351	1395	92	
U-235		139507	92	includes delayed fission gamma ray data
U-236	922361	7869	92	
U-237	922371			
U-238	922381	1398	92	
U-238		139807	92	includes delayed fission gamma ray data
V	23000	1323	23	
Xe-131	541311			
Xe-133	541331			
Xe-135	541351		54	
Zr	40000	7841	40	
Zr-90	400901		40	
Zr-91	400911		40	
Zr-92	400921		40	
Zr-93	400931		40	
Zr-94	400941		40	
Zr-96	400961		40	
Moderately absorbing fission product	130112			
Weakly absorbing fission product	230112			

[a]Materials and data set identifiers are the same in both the Fine-Group and Broad-Group General Purpose Neutron Libraries. See Table 3.1 for more GPN data set information.

[b]See Table 4.1 for more SGRP data set information.

[c]See Table 5.1 for more GRI data set information.

[d]Unless specified, scattering kernels are supplied at 296, 500, and 900K.

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2.0 ANSL-V NEUTRON AND GAMMA-RAY GROUP STRUCTURES

ENDF/B limitations, experience with group structures used in other cross-section libraries, and specific requirements for the ANS reactor were considered in the selection of ANSL-V group structures.

2.1 ANSL-V 39-Neutron-Group Structure

The ANSL-V 39-neutron-group structure is listed in Table 2.1. Top and bottom energies, 20 MeV and 1.0×10^{-5} eV, respectively, are consistent with the top and bottom energies of ENDF/B-V data.

Groups 1-17 and 28-30 are the same as groups in the CSRL[2.1-2.3] broad-group library structure, and are included in both the CSRL and CSRL-V[2.4-2.5] fine-group structures.* Groups 19-27 reflect expected ANS hot source design needs and groups 31-39 are for expected cold source design needs.[2.6]

For other than special versions of hydrogen, deuterium, and helium-4, the upper energy of the ANSL-V thermal range (energy below which upscatter matrices are included) is 3.0 eV. For the para and ortho versions of the aforementioned materials, the upper energy of the thermal range is 1.0 eV consistent with the upper cutoff energy of the para and ortho data.

2.2 ANSL-V 99-Neutron-Group Structure

The ANSL-V 99-neutron-group structure is the 39-group structure with all groups down to 1.77 eV divided into five equal lethargy units. Included in Table 2.1 is a comparison of the ANSL-V fine- and broad-neutron-group structures and group structures of the 227-group CSRL-V, 218-group CSRL, 27-group SCALE[1.6], and Hansen-Roach[2.7] cross-section libraries currently used at ORNL.

2.3 ANSL-V 44-Gamma-Ray-Group Structure

The 44-group ANSL-V gamma-ray energy structure is listed in Table 2.2. Group structures of the 18-group SCALE, 44-group CSRL-V, 36-group VITAMIN-C[2.8], and 38-group VITAMIN-E[2.9] gamma-ray interaction libraries in use at ORNL are compared with the ANSL-V structure in the table.

*CSRL is a 218-neutron-group ENDF/B-IV based cross-section library. The library and a 27-group subset library have been used extensively in criticality safety, reactor, and shielding studies. The libraries are incorporated in the SCALE System. CSRL-V is a 227-group library analogous to CSRL but based on ENDF/B-V.

Table 2.1. ANSL-V 99-Group and 39-Group Neutron Energy Structures

ANSL-V 99-Gp Group	ANSL-V 39-Gp Group	227-Gp CSRL-V Group	218-Gp CSRL Group	27-Group SCALE Group	16-Group Hansen-Roach Group	Upper Energy (eV)
1	1	1	1	1		2.0000+7
2						1.5941+7
3						1.2706+7
4						1.0127+7
5						8.0721+6
6	2	9	2	2		6.4340+6
7						5.5234+6
8						4.7417+6
9						4.0706+6
10						3.4946+6
11	3	12	5	2		3.0000+6
12						2.7235+6
13						2.4725+6
14						2.2447+6
15						2.0378+6
16	4	15	8	4		1.8500+6
17						1.7497+6
18						1.6548+6
19						1.5651+6
20						1.4803+6
21	5	17	10	5	3	1.4000+6
22						1.2816+6
23						1.1732+6
24						1.0740+6
25						9.8315+5
26	6	25	18	6	4	9.0000+5
27						7.6525+5
28						6.5068+5
29						5.5326+5
30						4.7043+5
31	7	39	32	7	5	4.0000+5
32						3.0314+5
33						2.2974+5
34						1.7411+5
35						1.3195+5
36	8	45	38	8	6	1.0000+5
37						7.0160+4
38						4.9224+4

Table 2.1. (cont'd)

ANSL-V 99-Gp Group	ANSL-V 39-Gp Group	227-Gp CSRL-V Group	218-Gp CSRL Group	27-Group SCALE Group	16-Group Hansen-Roach Group	Upper Energy (eV)
39						3.4536+4
40						2.4230+4
41	9	56	49	9	7	1.7000+4
42						1.2017+4
43						8.4941+3
44						6.0042+3
45						4.2441+3
46	10	63	56	10	8	3.0000+3
47						2.1368+3
48						1.5220+3
49						1.0841+3
50						7.7217+2
51	11	74	67	11	9	5.5000+2
52						3.9110+2
53						2.7811+2
54						1.9776+2
55						1.4063+2
56	12	86	79	12	10	1.0000+2
57						7.8600+1
58						6.1780+1
59						4.8559+1
60						3.8168+1
61	13	116	109	13	11	3.0000+1
62						2.4082+1
63						1.9332+1
64						1.5518+1
65						1.2457+1
66	14	132	125	14	12	1.0000+1
67						7.8600
68						6.1780
69						4.8559
70						3.8168
71	15	149	141	15	13	3.0000[a]
72						2.6996
73						2.4292
74						2.1859
75						1.9670

Table 2.1. (cont'd)

ANSL-V 99-Gp Group	ANSL-V 39-Gp Group	227-Gp CSRL-V Group	218-Gp CSRL Group	27-Group SCALE Group	16-Group Hansen-Roach Group	Upper Energy (eV)
76	16	163	156	16		1.7700
77	17	170	163	17		1.3000
78	18	190	183	19	14	1.0000[d]
79	19					7.6500-1
80	20					6.2500-1
81	21					4.7900-1
82	22					3.9700-1
83	23					3.3000-1
84	24					2.7000-1
85	25					2.1500-1
86	26					1.6200-1
87	27					1.0400-1
88	28	221	214	25		5.0000-2
89	29	223	216	26		3.0000-2
90	30	225	218[b]	27[b]		1.0000-2
91	31					4.4500-3
92	32					3.2500-3
93	33					2.6000-3
94	34					2.1500-3
95	35					1.8000-3
96	36					1.4500-3
97	37					1.1500-3
98	38					8.5000-4
99[c]	39[c]					5.5000-4

[a]For other than special hydrogen, deuterium, and He-4 GPN data sets, this is the top energy of the ANSL-V thermal energy range. This is also the top energy of the CSRL-V thermal energy range.

[b]Last energy group of structure. Bottom energy of the structure is 1.0000-5 eV.

[c]Bottom energy of structure is 1.0000-5 eV.

[d]Top energy of the ANSL-V thermal energy range for ortho and para data sets for hydrogen, deuterium, and He-4.

Table 2.2. ANSL-V 44-Group Gamma-Ray Energy Structure

ANSL-V Group	SCALE Group	CSRL-V Group	VITAMIN-C Group	VITAMIN-E Group	Energy[a] (eV)
1		1		1	20.00+6
2		2	1	2	14.00+6
3		3	2	3	12.00+6
4	1	4	3	4	10.00+6
5	2	5	4	5	8.00+6
6		6	5	6	7.50+6
7		7	6	7	7.00+6
8	3	8	7	8	6.50+6
9		9	8	9	6.00+6
10		10	9	10	5.50+6
11	4	11	10	11	5.00+6
12		12	11	12	4.50+6
13	5	13	12	13	4.00+6
14		14	13	14	3.50+6
15	6	15	14	15	3.00+6
16	7	16	15	16	2.50+6
17		17			2.35+6
18		18			2.15+6
19	8	19	16	17	2.00+6
20		20			1.80+6
21	9	21	17	18	1.66+6
22		22			1.57+6
23		23	18	19	1.50+6
24		24			1.44+6
25	10	25	19	20	1.33+6
26		26			1.20+6
27	11	27	20	21	1.00+6
28	12	28	21	22	8.00+5
29		29	22	23	7.00+5
30	13	30	23	24	6.00+5
31		31	24	25	5.12+5
32		32	25	26	5.10+5
33		33	26	27	4.50+5
34	14	34	27	28	4.00+5
35	15	35	28	29	3.00+5
36	16	36	29	30	2.00+5
37		37	30	31	1.50+5
38	17	38	31	32	1.00+5
39		39	32	33	7.50+4

Table 2.2. (cont'd)

ANSL-V Group	SCALE Group	CSRL-V Group	VITAMIN-C Group	VITAMIN-E Group	Energy[a] (eV)
40		40		34	7.00+4
41	18[b,c]	41	33	35	6.00+4
42		42	34	36	4.50+4
43		43	35	37	3.00+4
44[b]		44[b]	36[b]	38[b]	2.00+4

[a]Upper energy of the group.
[b]Bottom energy of the last group is 1.00+4 eV.
[c]Top energy of Group 18 is 5.00+4 eV.

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2.5 W. E. Ford, III, B. R. Diggs, L. M. Petrie, C. C. Webster, and R. M. Westfall, "CSRL-V: Processed ENDF/B-V 227-Neutron-Group and Pointwise Cross-Section Libraries for Criticality Safety, Reactor and Shielding Studies," ORNL/CSD/TM-160 (NUREG/CR-2306)(June 1982).

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2.8 R. W. Roussin, C. R. Weisbin, J. E. White, N. M. Greene, R. Q. Wright, and J. B. Wright, "The CTR Processed Multigroup Cross-Section Library for Neutronics Studies," ORNL-RSIC-37 (July 1980). Available from RSIC as DLC-41/ VITAMIN-C.

2.9 R. W. Roussin, "VITAMIN-E: A Coupled 174 Neutron, 38 Gamma-ray Multigroup Cross-Section Library for Deriving Application-Dependent Working Libraries for Radiation Transport Calculations," (December 1987). Available from RSIC as DLC-113/VITAMIN-E.

3.0 ANSL-V GENERAL PURPOSE NEUTRON LIBRARIES

Data sets in the ANSL-V Fine-Group (99-Group) and Broad-Group (39-Group) General Purpose Neutron (GPN) Libraries are listed in Table 3.1. Materials and data set identifiers are the same in both libraries. GPN multigroup data were generated with AMPX and NJOY[3.1] modules.

3.1 Sources of Evaluated Data

ENDF-formatted evaluations used to generate ANSL-V GPN data are listed in Table 3.1. Where evaluations were available in the ENDF/B-V General Purpose Library for specified ANS materials, those evaluations were used, except as noted below.

ENDF/B libraries are maintained and distributed by the National Nuclear Data Center (NNDC), which is located at the Brookhaven National Laboratory. The libraries are based on nuclear data evaluations provided by members of the Cross Section Evaluation Working Group (CSEWG). NNDC's policy is that CSEWG can issue modifications for ENDF/-V evaluations as needed. Thus, at the beginning of the ANSL-V project (June 1986), ENDF/B-V evaluations/modifications listed in the table were verified with NNDC to be the "latest" ENDF/B-V data.

Where ENDF/B-V General Purpose Library data were not available, multigroup data were processed, when available, from the LENDL Library.[3.2] Where neither ENDF/B-V General Purpose Library nor LENDL evaluations were available, evaluations from the ENDF/B-V Actinide (special purpose) and Fission Product Libraries were used. Finally, where evaluations were not available from the aforementioned sources, evaluations from the Japanese Evaluated Nuclear Data File (JENDL2) were used.[3.3-3.4]

Thermal bound data for H-1, H-2, Be-9, and graphite were processed from ENDF/B thermal scattering law data and included with the appropriately processed epithermal data.[3.5] ENDF/B evaluations used for the thermal bound calculations are identified in Table 3.1. Thermal data for ortho-hydrogen, and para-hydrogen and para-deuterium were created using the technique described in Sect. 3.5.

In order to address special ANS Reactor concerns, evaluated data for H-1, H-2, He-4, Cd-113, Al-27, Cs-134, Nd 147, Eu-152, Eu-154, Eu-155, natural Eu, and Np-239 were modified before processing. Additional one-dimensional array data were added to ANSL-V U-235 and U-238 GPN data sets to accommodate delayed fission gamma-ray data in corresponding ANSL-V SGRP data sets. These modifications are discussed in Sect. 3.5.

Users of ANSL-V data should be cognizant of the scope and limitations of ENDF-formatted libraries and evaluations in the libraries used to prepare the ANSL-V data. For example, it should be noted that the ENDF/B-V Actinide and Fission Product Libraries do not contain the full complement of cross-section data. Users are reminded that File 1 of an ENDF evaluation contains the evaluator's (or evaluators') summary of the scope and limitations of the evaluation.

Table 3.1. ANSL-V General Purpose Neutron Library Materials[a]

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wtng. Fctn.[i]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
H-1	1001	1301	1	0	N	A	1 2 27 101 251-253 1007 1099 3002 3099 3102 9002 9901	2 1007	Free gas
H-1	1801	1301	1	0	N	A	1 2 27 101 102 251-253 1077 1099 3002 3099 3102 9002 9901	2 1007	MAT 1002 H_2O bound thermal @ 296,350, 400,450,500,600, 800, and 1000 °K
H-1	2222	1301[c]	1	0	N	A	1 2 27 101 102 251-253 1077 1099 3002 3099 3102 9002 9901	2 1007	Ortho-hydrogen
H-1	1111	1301[c]	1	0	N	A	1 2 27 101 102 251-253 1077 1099 3002 3099 3102 9002 9901	2 1007	Para-hydrogen
H-2	1802	1302	2	0	N	A	1 2 16 27 101 102 251-253 1077 1099 3002 3099 3102 9002 9901	2 16 1007	MAT 1004 D_2O bound thermal @ 296,350 400,450,500,600, 800, and 1000 °K
H-2	3333	1302[c]	2	0	N	A	1 2 16 27 101 102 251-253 1077 1099 3002 3099 3102 9002 9901	2 16 1007	Para-deuterium
H-3	1003	1169	2	0	N	A	1 2 16 27 101 251-253 1077 1099 3002 3099 9002 9901	2 16 1007	
He-3	2003	1146	1	0	N	A	1 2 27 101 103 104 251-253 1077 1099 3002 3099 9002 9901	2 1007	
He-4	2004	1270[c]	0	0	N	A	1 2 27 101 251-253 1077 1099 3002 3099 9002 9901	2 1007	w/special scatt. data

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[b]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Be-9	4009	1304	2	0	N	A	1 2 6-9 27 46-49 101-105 107 251-153 740 741 1007 1009 3002 3099 3102 9002	2 46-49 1007	Mat 1064 bound thermal @296,400,500,600,700, 800,1000, and 1200 °K.
B-10	5010	1305	1	0	N	A	1 2 4 27 51-85 101-104 107 113 251-253 700-702 780 781 1007 1099 3002 3099 3102 9002 9901	2 51-85 1007	
B-11	5011	1160	1	0	N	A	1 2 4 16 27 51-53 91 101-103 105 107 251 253 1007 1099 3002 3099 3102 9002 9901	2 16 51 52 53 91 1007	
Graphite	6666	1306	2	0	N	A	1 2 3 4 27 51-68 91 101-104 107 203 204 207 251-253 1007 1099 3002 3099 3102 9002 9901	2 51-68 91 1007	MAT 1065 graphite thermal scatt. kernel @ 296,500, 800,1200, and 2000 °K.
N-14	7014	1275	2	0	N	A	1 2 4 16 27 51-82 101-105 107 108 251-253 700-704 720-723 740 741 780-790 1007 1099 3002 3099 3102 9002 9901	2 16 51-82 1007	
N-15	7015	1307	1	0	N	A	1 2 4 16 22 27 28 51-57 91 101-105 107 251-253 700 701 718-722 738 740-742 758 780-785 798 799 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-57 91 1007	
O-16	8016	1276	2	0	N	A	1 2 4 27 51-89 101-104 107 251-253 780-783 1007 1099 3002 3099 3102 9002 9901	2 51-89 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[b]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Na-23	11023	1311	3	4	N	B	1-4 16 27 51-68 91 101-103 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-68 91 1007	
Mg	12000	1312	1	0	N	B	1-4 16 22 27 28 51-91 101-103 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-91 1007	
Al-27	13027	1313[c]	1	0	N	B	1 2 4 16 27 51-89 90 101-105 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-89 90 1007	
Al-27	130271	1313[c]	1	0	N	C	1 2 4 16 27 51-90 101-105 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-90 1007	
Si	14000	1314	3	0	N	B	1-4 16 22 27 28 51-72 91 101-104 107 251-253 700-714 718 719 780-791 798 799 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-72 91 1007	
Si	140001	1314	3	0	N	C	1-4 16 22 27 28 51-72 91 101-104 107 251-253 700-714 718 719 780-791 798 799 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-72 91 1007	
K	19000	1150	1	0	N	B	1 2 4 16 22 27 28 51-67 91 101 102 103 107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-67 91 1007	
Ti	22000	1322	1	0	N	B	1 2 4 16 17 22 27 28 51-62 91 101-107 111 112 203 207 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-62 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[e]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
V	23000	1323	1	0	N	B	1 2 4 16 22 27 28 51-57 91 101-105 107 203 207 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-57 91 1007	
Cr	24000	1324	2	81	N	B	1 4 16 17 22 27 28 51-91 101-107 203-207 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-91 1007	
Mn-55	25055	1325	2	43	N	B	1 2 4 16 17 22 27 28 51-62 91 101 102 103 104 106 107 251 525 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-62 91 1007	
Fe	26000	1326	3	57	N	B	1 2 3 4 16 22 27 28 51-91 101-107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-91 1007	
Co-59	27059	1327	3	72	N	B	1 2 4 16 22 27 28 51-61 91 101-107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-61 91 1007	
Ni	28000	1328	2	129	N	B	1 2 4 16 22 27 28 51-76 91 101-104 107 111 203 204 207 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-76 91 1007	
Kr-82	360821	1332	1	1	N	C	1-4 16 27 51-59 91 101-107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-59 91 1007	Nuclide in fission product chain

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[b]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Kr-83	360831	1333	1	2	N	C	1 2 3 4 16 17 27 51-56 91 101-107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 51-56 91 1007	Nucl. in fis. prod. chain
Zr	40000	1340	2	114	N	B	1 2 4 16 27 51-69 91 101-103 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-69 91 1007	
Zr-90	400901	1385	2	34	N	C	1 2 4 16 27 51-55 91 101-102 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51 52 53 54 55 91 1007	
Zr-91	400911	1386	2	35	N	C	1 2 4 16 27 51-61 91 101-103 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-61 91 1007	
Zr-92	400921	1387	2	15	N	C	1 2 4 16 27 51-56 91 101-103 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-56 91 1007	
Zr-93	400931	9232	1	0	N	C	1 2 4 27 51-60 91 101 102 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 51-60 91 1007	Nucl. in fis. prod. chain
Zr-94	400941	1388	2	22	N	C	1 2 4 16 27 51-58 91 101-103 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-58 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[i]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Zr-96	400961	1389	2	8	N	C	1 2 4 16 27 51 52 53 54 91 101 102 107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 51 52 53 54 91 1007	
Mo-97	420971	9284	1	30	N	C	1 2 4 27 51-69 91 101 102 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 51-69 91 1007	Nucl. in fis. prod. chain
Tc-99	430991	1308	2	42	Y	C	1 2 4 16 27 51-61 91 101 102 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 51-61 91 1007	Nucl. in fis. prod. chain
Ru-101	441011	9330	1	7	N	C	1 2 4 27 51-69 91 101 102 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 51-69 91 1007	Nucl. in fis. prod. chain
Ru-103	441031	9332	1	0	N	C	1 2 4 27 51-64 91 101 102 251-253 1007 1099 3002 3099 3102 9002	2 51-64 91 1007	
Ag-107	47107	1407	2	73	N	B	1 2 4 16 17 22 27 28 51-63 91 101-107 203-207 251 252 253 1007 1009 3002 3099 3102 9002 9901	2 16 17 22 28 51-63 91 1007	
Ag-107	471071	1407	2	73	N	C	1 2 4 16 17 22 27 28 51-63 91 101-107 203-207 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-63 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[i]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Ag-109	47109	1409	2	83	N	B	1 2 4 16 17 22 27 28 51-70 91 101-107 203-207 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-70 91	
Ag-109	471091	1409	2	83	N	C	1 2 4 16 17 22 27 28 51-70 91 101-107 203-207 251 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-70 91 1007	Nucl. in fis. prod. chain
Cd	48000	1281	1	0	N	B	1-4 16 27 51-54 91 101 103 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-54 91 1007	
Cd-113	481131	1318	1	12	N	C	1 2 4 16 27 51 52 53 91 101 102 103 107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 51 52 53 91 1007	
Cs-135	551351	9665	1	0	N	C	1 2 4 27 51 52 53 91 101 102 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 51 52 53 91 1007	Nucl. in fis. prod. chain
Ce-141	581411	9725	1	0	N	C	1 2 4 27 51 52 53 54 91 101 102 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 51 52 53 54 91 1007	
Pr-143	591431	9745	1	0	N	C	1 2 4 27 51-56 91 101 102 251 252 253 1007 3022 3099 3102 9002 9901	2 51-56 91 1007	Nucl. in fis. prod. chain

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[b]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Pm-147	611471	9783	1	13	N	C	1 2 4 16 17 22 27 28 51-55 91 101-107 251-253 1007 1009 3002 3099 3102 9002 9901	2 16 17 22 28 51-55 91 1007	Nucl. in fis.
Sm-149	621491	1319	1	29	N	C	1 2 4 16 17 27 51-60 91 101-103 107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 51-60 91 1007	Nucl. in fis. prod. chain
Sm-153	621531	9812	1	0	N	C	1 2 4 27 51-62 91 101 102 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 51-62 91 1007	Nucl. in fis. prod. chain
Eu	63000	9463[c]	1	162	Y	B	1 2 4 16 17 22 27 28 51-70 91 101 102 103 104 105 106 107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-70 91 1007	
Eu	630001	9463[c]	1	162	Y	C	1 2 4 16 17 22 27 28 51-70 91 101-107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-70 91 1007	
Eu-151	631511	1357	1	91	Y	C	1 2 4 16 17 22 27 28 51-59 91 101 102 103 104 105 106 107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-59 91 1007	
Eu-152	631521	4292[c]	2	83	Y	C	1 2 4 16 17 22 27 28 51-55 91 101-107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-55 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wtng. Fctn.[e]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Eu-153	631531	1359	1	71	Y	C	1-4 16 17 22 27 28 51-61 91 101-107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-61 91 1007	Nucl. in fis. prod. chain
Eu-154	631541	4293[c]	1	59	Y	C	1 2 4 16 17 22 27 28 51-55 91 101 102 103 104 105 106 107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-55 91 1007	Nucl. in fis. prod. chain
Hf	72000	1372	1	235	Y	B	1 2 4 16 27 51-62 91 101 102 103 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 51-62 91 1007	
Hf	720001	1372	1	235	Y	C	1 2 4 16 27 51-62 91 101 102 103 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 51-62 91 1007	
Ta-181	73181	1285	2	75	Y	B	1 2 4 16 17 27 51-60 91 101 102 103 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 51-60 91 1007	
Ta-182	73182	1127	1	9	Y	B	1 2 4 16 17 27 51-58 91 101 102 107 251 252 253 1007 1099 3002 3099 3102 9002 9901	2 16 17 51-58 91 1007	
U-233	922331	1393	2	158	Y	C	1 2 4 16 17 18 27 51-64 91 101 102 251 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-64 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[e]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
U-234	922341	1394	3	118	Y	C	1 2 4 16 17 18 27 51-56 91 101 102 251 252 253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-56 91 1007	
U-235	922351	1395	3	129	Y	C	1 2 4 16 17 18 27 51-66 91 101 102 251 252 253 452 1007 1018 1099 3002 3099 3102 9002 9901	2 16 17 51-66 91 1007	
U-236	922361	1396	3	190	Y	C	1 2 4 16 17 18 27 51-56 91 101 102 251 252 253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-56 91 1007	
U-238	922381	1398	3	163	Y	C	1 2 4 16 17 18 27 38 51-77 91 101 102 251 252 253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-77 91 1007	
C-12	6012	1306	2	0	N	A	1-4 27 51-68 91 101-104 107 203 204 207 251-253 1007 1099 3002 3099 3102 9002 9901	2 51-68 91 1007	
N-15	70150	1307	1	0	N	A	1 2 4 16 22 27 28 51-57 91 101-105 107 251-253 700 701 718-722 738 740-742 758 780-785 798 799 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-57 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[i]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
F-19	90190	1309	3	0	N	D	1-4 16 22 27 28 51-71 91 101-105 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 22 28 51-71 91 1007	
Cu	29000	1329	1	46	N	B	1-4 16 17 22 27 28 51-61 91 101-104 106 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-61 91 1007	
Mo	420001	1321	1	25	Y	B	1 2 4 16 17 27 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 91 1007	
Sn	500001	8850	0	0	N	C	1 2 4 16 17 27 91 101-102 251-253 285 1007 1099 3002 3099 3102 9002 9901	2 16 17 91 1007	
Pb	820001	1382	2	0	N	B	1-4 16 17 27 51-85 91 101-105 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 51-85 91 1007	
Rh-103	451031	1310	1	59	N	C	1 2 4 16 27 51-64 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-64 91 1007	
Rh-105	451051	9355	2	0	N	C	1 2 4 27 51-53 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 51-53 91 1007	
I-135	531351	9618	1	0	N	C	1 2 4 27 91 101-102 251-253 1007 1099 3002 3099 9002 9901	2 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[i]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Xe-131	541311	1351	1	39	N	C	1 2 4 16 17 27 51-56 91 101-105 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 51-56 91 1007	
Xe-133	541331	9643	1	0	N	C	1 2 4 27 51 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 51 91 1007	
Xe-135	541351	1294	1	0	N	C	1 2 4 27 51 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9001	2 51 91 1007	
Cs-133	551331	1355	1	123	N	C	1 2 4 16 27 51-55 91 101-103 107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 51-55 91 1007	
Cs-134	551341	4663[e]	1	6	N	C	1 2 4 27 51-55 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 51-55 91 1007	
Nd-143	601431	9764	1	17	N	C	1 2 4 16 17 22 27 28 51-58 91 101-107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-58 91 1007	
Nd-145	601451	9766	1	79	N	C	1 2 4 16 17 22 27 28 51-64 91 101-107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-64 91 1007	
Nd-147	601471	4768[e]	1	7	N	C	1 2 4 27 51-53 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 51-53 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wtng. Fctn.[i]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Pm-148	611481	9784	1	0	N	C	1 2 4 27 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 91 1007	
Pm-148m	611482	9785	1	1	N	C	1 2 4 27 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 91 1007	
Pm-149	611491	9786	1	0	N	C	1 2 4 27 51-56 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 51-56 91 1007	
Sm-150	621501	9809	1	11	N	C	1 2 4 27 51-66 91 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 51-66 91 1007	
Sm-151	621511	9810	1	7	N	C	1 2 4 16 17 22 27 28 51-56 91 101-107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-61 91 1007	
Sm-152	621521	9811	1	57	N	C	1 2 4 16 17 22 27 28 51-64 91 101-107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-64 91 1007	
Eu-155	631551	4832[e]	1	7	N	C	1 2 4 16 17 22 27 28 51-59 91 101-107 251-253 1007 1099 3002 3099 3102 9002 9901	2 16 17 22 28 51-59 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[i]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Th-232	902321	1390	2	241	Y	C	1 2 4 16 17 18 27 51-65 91 101-102 251-253 452 455-456 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-65 91 1007	
U-237	922371	8237	1	27	Y	C	1 2 4 16 17 18 27 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 91 1007	
Np-237	932371	1337	2	168	Y	C	1 2 4 16 17 18 27 51-61 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-61 91 1007	
Np-238	932381	8338	1	94	N	C	1 2 18 27 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901 9452	2 1007 9452	
Np-239	932391	2932	1	0	N	C	1 2 4 16 17 18 27 51-58 91 101-102 251 452 1007 1018 1099 3002 3018 3099 3102 9002 9901 9452	2 16 17 51-58 91 1007 9452	
Pu-238	942381	1338	3	14	Y	C	1 2 4 16 17 18 27 51-65 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-65 91 1007	
Pu-239	942391	1399	2	127	Y	C	1 2 4 16 17 18 27 37 51-68 91 101-102 251-253 452 455-456 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-68 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[i]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Pu-240	942401	1380	3	200	Y	C	1 2 4 16 17 18 27 51-62 91 101-102 251-253 452 455-456 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-62 91 1007	
Pu-241	942411	2945	0	91	Y	C	1 2 4 16 17 18 27 37 51-61 91 101-102 251 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-61 91 1007	
Pu-242	942421	1342	2	67	Y	C	1 2 4 16 17 18 27 51-69 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-69 91 1007	
Pu-243	942431	8443	1	40	Y	C	1 2 4 16 17 18 27 37 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 91 1007	
Am-241	952411	1361	2	65	Y	C	1 2 4 16 17 18 27 51-68 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-68 91 1007	
Am-242	952420	8542	1	81	Y	C	1 2 18 27 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 1007	
Am-242m	952421	1369	1	6	Y	C	1 2 4 16 17 18 27 37 51-63 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 71 51-63 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[e]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Am-243	952431	1363	2	219	Y	C	1 2 4 16 17 18 27 37 51-67 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-67 91 1007	
Cm-242	962421	8642	1	20	Y	C	1 2 4 16 17 18 27 51-53 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-53 91 1007	
Cm-243	962431	1343	1	15	Y	C	1 2 4 16 17 18 27 37 51-65 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-65 91 1007	
Cm-244	962441	1344	2	37	Y	C	1 2 4 16 17 18 27 51-53 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-53 91 1007	
Cm-245	962451	1345	2	38	N	C	1 2 4 16 17 18 27 37 91 101-102 251-253 452 455-456 1007 1018 1099 3002 3018 3099 3102 9002 9901 9452	2 16 17 91 1007 9452	
Cm-246	962461	1346	1	10	Y	C	1 2 4 16 17 18 27 37 51-61 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-61 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[i]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Cm-247	962471	8647	1	34	Y	C	1 2 4 16 17 18 27 37 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 91 1007	
Cm-248	962481	8648	1	46	Y	C	1 2 4 16 17 18 27 37 51-68 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 51-57 91 1007	
Bk-249	972491	8749	1	91	Y	C	1 2 4 16 17 18 27 37 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 91 1007	
Cf-249	982491	8849	1	52	Y	C	1 2 4 16 17 18 27 37 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 91 1007	
Cf-250	982501	8850	1	20	Y	C	1 2 4 16 17 18 27 37 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 91 1007	
Cf-251	982511	8851	1	20	Y	C	1 2 4 16 17 18 27 37 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 91 1007	

Table 3.1. (Cont'd)

Nuclide	ANSL-V Data Set ID	ENDF/B-V Evaluation MAT	MOD	No. Re.[g] Res.	Unr. Res.[h]	Wting. Fctn.[b]	MT Nos. for 1-d Processes In Data Set[f]	MT Nos. for 2-d Processes In Data Set[f]	Remarks
Cf-252	982521	8852	1	20	Y	C	1 2 4 16 17 18 27 37 91 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 16 17 91 1007	
Cf-253	982531	8853	1	119	Y	C	1 2 18 27 101-102 251-253 452 1007 1018 1099 3002 3018 3099 3102 9002 9901	2 1007	
Es-253	992531	8953	1	27	Y	C	1 2 27 101-102 251-253 1007 1099 3002 3099 3102 9002 9901	2 1007	
MAFP	130112	-	-	0	N	C	1 2 4 16 17 22 27 28 101-107 251-253 1007 1021 1023 1099 9901	1	
WAFP	230112	-	-	0	N	C	1 2 4 16 17 27 101-107 251-253 1007 1021 1023 1099 9901	1	

a. Information in the table is applicable to both the Fine-Group and Broad-Group General Purpose Neutron Libraries.

b. The following weighting functions were used in the generation of the multigroup data:

Weight Fctn. Designation	Weight Function
A	$10^{-5} \le E_n \le 0.1265$ eV, Maxwellian spectrum with a temperature of 300 K; 0.1265 eV $< E_n \le 1.4$ MeV, $1/(E*Sigma_T)$ spectrum; $1.4 < E_n = 20$ MeV, fission spectrum with a temperature of 1.27×10^6 eV. The "$Sigma_T$" is the microscopic total cross sections for the respective materials.

Table 3.1. (Cont'd)

B	$10^{-5} \le E_n \le 0.1265$ eV, Maxwellian spectrum with a temperature of 300K; 0.1265 eV $< E_n \le 0.75$ MeV, 1/(E*Sigma$_T$) spectrum; 0.75 $< E_n \le 20$ MeV, fission spectrum with a temperature of 1.27×10^6 eV. The "Sigma$_T$" is the microscopic total cross sections for the respective materials.
C	$10^{-5} \le E_n \le 0.1265$ eV, Maxwellian spectrum with a temperature of 300K; 0.1265 eV $< E_n \le 0.1$ MeV, 1/(E*Sigma$_T$) spectrum; 0.1 $< E_n \le 20$ MeV, fission spectrum with a temperature of 1.27×10^6 eV. The "Sigma$_T$" is the macroscopic total cross sections for a homogenized representation of the fuel region.
D	$10^{-5} \le E_n \le 0.1265$ eV, Maxwellian spectrum with a temperature of 300 K; 0.1265 eV $< E_n \le 0.75$ Mev, 1/E spectrum; 0.75 $< E_n \le 20$ MeV, fission spectrum with a temperature of 1.27×10^6 eV.

c. Evaluation modified in house for the ANSL-V project.

d. ENDF B-V evaluation unless otherwise noted.

e. Potential scattering cross sections for GPN Bondarenko factors are listed in Table 3.2.

f. MT numbers are defined in Appendix B.

g. Number of resolved resonances in processed data set.

h. Are unresolved resonance data included in the ENDF evaluation? If "yes," Brondarenko factor data are included in the ANSL-V data set.

3.2 Calculational Procedure

Materials were processed into the ANSL-V 99-neutron-group structure and then collapsed into the 39-group structure. The standard procedure for generating data sets for the GPN Libraries is depicted in Fig. 3.1.

The processing procedure was to: (1) process an ENDF-formatted evaluation with XLACS-77,* (2) perform first-order data checks by running RADE on the resulting data set, using VASELINE to plot selected cross sections, and evaluating the findings, (3) execute FRESH to adjust thermal scattering matrices for the heavier materials, (4) execute COMET to either force the averaged values to agree with the matrix sums or to force the matrix sums to agree with the average values, (5) execute RIGEL[3.6]-UNRESR-TABU sequences to generate Bondarenko factors for unresolved resonance data where applicable, (6) execute UNITAB to combine the averaged XLACS-77 data with the Bondarenko factor data, (7) run RADE on the resulting data set, (8) use AJAX to combine all fine-group data sets into a single library, (9) execute MALOCS to collapse the fine-group data sets into the broad-group structure, and, finally, (10) use AIM to convert the resulting ANSL-V GPN Libraries, which at this point in the procedure are in binary AMPX master format, into BCD format. Steps in this procedure are summarized in Sects. 3.2.1 through 3.2.7.

3.2.1 XLACS-77 Processing

Using the specified ENDF-formatted evaluations as input, XLACS-77 cases were executed to generate AMPX master data sets for GPN materials. For each material, the code was used to process multigroup neutron data for all neutron-induced reactions except unresolved resonance data. Although XLACS-77 was derived from XLACS-II, which has been in production use for many years, it is noted that the ANSL-V project involved the first production use of XLACS-77.

3.2.2 RADE-VASELINE First-Order Data Check

Each XLACS-77 produced AMPX master data set was checked with the RADE module. One of the more important checks made by the module is that of summing elements of transfer matrices and comparing these sums against averaged values for a process. By definition,

$$\overline{\sigma_g} = \sum_{g'} \sigma_o\ (g \rightarrow g') \qquad (3\text{-}1)$$

This says that the averaged cross section in a group should equal the sum of all the transfers out of the group of the zeroth moment of a Legendre fit to the cross section. In XLACS-77, the calculation of $\bar{\sigma}_g$ and $\sigma_0(g \rightarrow g')$ are kept very independent by not force-normalizing the scattering matrix elements to equal the averaged value for the process. This allows a code like RADE to be used to determine both processing inadequacies and problems in the basic data which are to be averaged.

*XLACS-77 has not been documented.

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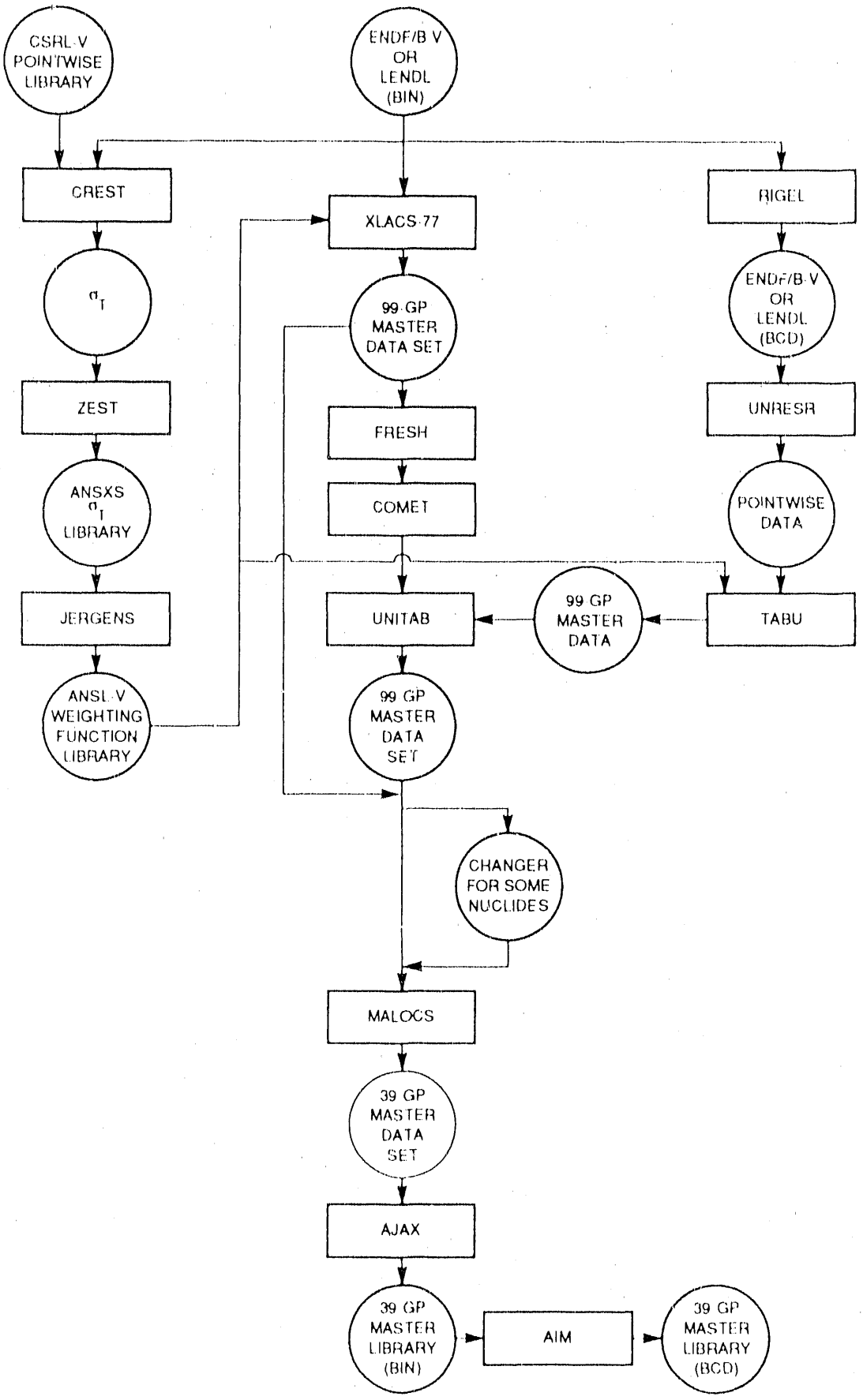

Fig. 3.1 Calculational Sequence for ANSL-V GPN Library.

Bear in mind that the basic equations used to generate the terms are quite different, viz.,

$$\overline{\sigma_g} = \frac{\int_g dE\, \sigma(E)\phi(E)}{\int dE\ \phi(E)} \tag{3-2}$$

while,

$$\sigma_o(g \rightarrow g') = \frac{\int_{-1}^{1} d\mu \int_g dE\ \sigma_s(E)\ \phi\ (E) \int_{g'} dE'\, f(E \rightarrow E', \mu)}{\int_g dE\ \phi(E)} \tag{3-3}$$

Obviously Eq. 3-3 is considerably more complicated than Eq. 3-2, such that if a check indicates that Eq. 3-1 is satisfied to a specified (small) tolerance, then one can place a high degree of confidence in the processing procedures. When the discrepancies were more than 1% for the ANSL-V processing, the processing procedures and/or the ENDF/B data were examined.

The VASELINE module was used to make plots of selected multigroup data from each XLACS-77 produced data set. Where point data were available for a nuclide, the multigroup values for the total, elastic scattering, and (n,gamma) cross sections were plotted with the point data for the reactions. Information from the RADE and VASELINE checks resulted in the FRESH and COMET corrections described below.

3.2.3 FRESH to Adjust Thermal Scattering Matrices

A problem arises due to the manner in which XLACS-77 generates thermal scattering matrices for heavier nuclides. The free gas model is employed in the code in a treatment which assumes a constant free atom value at the upper end of the thermal region and the treatment itself generates a 1/v variation which is the consequence of Doppler broadening a flat cross section. For many nuclides this is reasonable, and the only problem one faces is that of picking the free atom scattering value.

In ENDF/B files, the scattering radius for each nuclide is specified in its evaluation. XLACS-77 uses this radius for determining the free atom scattering value. For nuclides with thermal resonances, this is not correct as the cross section is not a smooth function. For heavy nuclides, the treatment of scattering generally does not make much difference since the scattering serves only to shift the energy of a neutron slightly. (This fact is the basis of the Narrow Resonance Moderators and Infinite Mass Absorbers, NRIM, resonance treatment where heavy nuclide scattering is totally ignored.) For processing ANSL-V data, a less severe approximation was desired.

Consequently, XLACS-77 was changed to always retain the point-averaged scattering values in the thermal energy range, even when these data are overridden by values generated by the smooth free gas treatment. This was done in order to allow one to examine plots of point versus group averaged thermal values and to readjust the thermal matrices to make the variation follow the resonance behavior when it appeared to be more reasonable.

This led to the creation of the FRESH (Fix RESonance Hangups) AMPX module. FRESH takes an XLACS-77 generated AMPX master data set, sums the thermal matrices

and renormalizes them to make the sums agree with the resonance values. Appropriate adjustments are also made to the total cross section to account for the changes.

3.2.4 UNRESR-TABU-UNITAB Processing to Add Bondarenko Data

XLACS-77 contains no provisions for generating Bondarenko factors for self shielding in the unresolved resonance region. This processing is provided by a separate AMPX module called TABU and requires data such as can be produced by the UNRESR module of the NJOY system. For the ENDF/B-V evaluations which contain unresolved resonance data (see Table 3.1), UNRESR was used to process pointwise cross sections from unresolved resonance data. UNRESR produces Bondarenko factor data for a user-specified range of potential scattering cross sections and temperatures. Potential scattering cross sections specified for ANSL-V materials are listed in Table 3.2. Temperatures for Bondarenko factor data are 300, 600, 900, 1200, and 2100 K. The fact that these temperatures are different than those selected for scattering kernels poses no problem since Bondarenko factors are chosen by interpolation between supplied data values whereas scattering kernel data are selected from the temperture closest to that of the problem conditions.

UNRESR-produced cross sections were placed into the ANSL-V 99-group format with the TABU module. XLACS-77 and TABU-produced data for materials were combined into a consistent data set in AMPX master (binary) format with the UNITAB module. AJAX was used to collect data sets into a single library.

3.2.5. COMET Normalization

With an AMPX master data set as input, COMET can be triggered either to force averaged cross section values to agree with matrix sums or to force the matrix sums to agree with the averaged values. This normalization can be quite important because when an imbalanced cross section set is used in a transport code, the difference between the averaged value and the sum of the transfer elements is treated like an absorption cross section. It is easy to envision that nuclides such as graphite, which have very small absorption values, can have balancing "errors" on the same order as the absorption value.

Because of the above considerations, nuclides in the ANSL-V GPN Library were processed with COMET and the matrix sums for each of the 99 energy groups for the following processes were forced to agree with the group averaged value:

MT=2 (elastic scattering)
MT=4 (inelastic scattering)
MT=6 through 9 (special n,2n for Be-9)
MT=16 (n,2n)
MT=17 (n,3n)
MT=22 (n,n')
MT=28 (n,n'p)
MT=46 through 49 (special n,2n for Be-9)

In normalizing MT=2, COMET also normalizes thermal matrices designated by MT=1007 and the MT=4 normalization forces the sum of matrices identified by MT=51 through MT=91 (up to 41 matrices) to agree with the averaged MT=4 values. There was one exception, Cd-113, described in Sect. 3.5 below.

Table 3.2. Potential scattering cross sections for GPN Bondarenko factors.

Nuclide	Potential Scattering Cross Sections (Barns)
Mo	50,10^2,10^3,10^4,10^{10}
Tc-99	10^2,10^3,10^4,10^{10}
Eu	50,10^2,10^3,10^4,10^{10}
Eu-151	50,10^2,10^3,10^4,10^{10}
Eu-152	50,10^2,10^3,10^4,10^{10}
Eu-153	50,10^2,10^3,10^4,10^{10}
Eu-154	50,10^2,10^3,10^4,10^{10}
Hf	1,10,50,10^2,10^3,10^4,10^{10}
Ta-181	10^2,10^3,10^4,10^{10}
Ta-182	1,10,50,10^2,10^3,10^4,10^{10}
Th-232	1,10,50,10^2,10^3,10^4,10^{10}
U-233	1,10,50,10^2,10^3,10^4,10^{10}
U-234	1,10,50,10^2,10^3,10^4,10^{10}
U-235	1,10,50,10^2,10^3,10^4,10^{10}
U-236	1,10,50,10^2,10^3,10^4,10^{10}
U-237	1,10,50,10^2,10^3,10^4,10^{10}
U-238	1,10,50,10^2,10^3,10^4,10^{10}
Np-237	1,10,50,10^2,10^3,10^4,10^{10}
Pu-238	1,10,50,10^2,10^3,10^4,10^{10}
Pu-239	1,10,50,10^2,10^3,10^4,10^{10}
Pu-240	1,10,50,10^2,10^3,10^4,10^{10}
Pu-241	1,10,50,10_2,10^3,10^4,10^{10}
Pu-242	1,10,50,10^2,10^3,10^4,10^{10}
Pu-243	10^2, 10^3, 10^4, 10^{10}
Am-241	50,10^2,10^3,10^4,10^{10}
Am-242	10^2, 10^3, 10^4, 10^{10}
Am-242m	10^2, 10^3, 10^4, 10^{10}
Am-243	50,10^2,10^3,10^4,10^{10}
Cm-242	1,10,50,10^2,10^3,10^4,10^{10}
Cm-243	10^2, 10^3, 10^4, 10^{10}
Cm-244	1,10,50,10^2,10^3,10^4,10^{10}
Cm-246	1,10,50,10^2,10^3,10^4,10^{10}
Cm-247	1,10,50,10^2,10^3,10^4,10^{10}
Cm-248	1,10,50,10^2,10^3,10^4,10^{10}
Bk-249	50,10^2,10^3,10^4,10^{10}
Cf-249	50,10^2,10^3,10^4,10^{10}
Cf-250	1,10,50,10^2,10^3,10^4,10^{10}
Cf-251	1,10,50,10^2,10^3,10^4,10^{10}
Cf-252	1,10,50,10^2,10^3,10^4,10^{10}
Cf-253	50,10^2,10^3,10^4,10^{10}
Es-253	10^2, 10^3, 10^4, 10^{10}

3.2.6 RADE Review

After all XLACS-77, FRESH, UNRESR-TABU-UNITAB, COMET, and AJAX runs were made, data sets in the GPN Library were again checked with RADE and the results were reviewed.

3.2.7 MALOCS Collapsing

MALOCS was used to collapse the 99-group data sets into the 39-group structure. Weighting functions for the collapse were the multigroup representations of the weighting spectra used to generate the respective XLACS-77 produced data sets. (The weighting function for each material is included in the respective AMPX master data set as the MT=1099 one-dimensional array.)

3.2.8 CHANGER Processing

After data had been generated for 71 nuclides, ANS Project management requested that the 0.588 eV energy boundary be moved to 0.625 eV in order that calculations with the ANSL-V libraries be easily comparible to those for other reactors. Since only one group boundary was to be changed and since the cross-section generation project had limited financial resources, N. M. Greene wrote a computer program named CHANGER to accomplish the group boundary revision. Only data related to the two groups separated by the 0.588 eV boundary were affected.

3.2.9 AIM Conversion

Finally, AIM was used to convert the binary libraries to BCD format.

3.3 Order of Scattering

The order of scattering for epithermal and fast elastic scattering and discrete level inelastic scattering is P_5 for all materials. For thermal scattering, it is P_3.

3.4 Weighting Functions

Weighting functions used in the XLACS-77 generation of GPN multigroup data from ENDF/B-formatted data are identified in Table 3.1. The fission-1/(E*Sigma$_f$)-Maxwellian weighting function is not one of the "built-in" XLACS-77 weighting functions. Consequently, appropriate ANS representative fuel region and material-dependent weighting functions, Weighting Functions C and B, respectively, were generated for use with XLACS-77 and TABU.

Steps to generate weighting functions involved using: (1) CREST to access microscopic total cross-section sets from the appropriate library, (2) ZEST to collect the cross-section sets into a single library, and (3) JERGENS to produce the various fission-1/(E*Sigma$_f$)-Maxwellian weighting functions. Sources of "Sigma$_f$" data which were used as input to the JERGENS module are identified in Table 3.3. For non-resonance materials, the sources were the appropriate ENDF/B-V, LENDL, and JENDL2 evaluations. For resonance materials, the source was the previously prepared

Table 3.3. Sources of sigma-T data for JERGENS input

Nuclide	ID	MAT	Source	Nuclide	ID	MAT	Source
H-1	1001	1301	ENDF/B-V	Si	14000	1314	ENDF/B-V
H-1	1801	1301	ENDF/B-V	K	19000	1150	ENDF/B-V
H-1	2222	1301	ENDF/B-V	Ti	22000	1322	ENDF/B-V
H-1	1111	1301	ENDF/B-V	V	23000	1323	ENDF/B-V
H-2	1802	1302	ENDF/B-V	Cr	24000	1324	CSRL-V
H-2	3333	1302	ENDF/B-V	Mn-55	25055	1325	CSRL-V
H-3	1003	1169	ENDF/B-V	Fe	26000	1326	CSRL-V
He-3	2003	1146	ENDF/B-V	Co-59	27059	1327	CSRL-V
He-4	2004	1270	ENDF/B-V	Ni	28000	1328	CSRL-V
Be-9	4009	1304	ENDF/B-V	Cu	29000	1329	CSRL-V
B-10	5010	1305	ENDF/B-V	Zr	40000	1340	CSRL-V
B-11	5011	1160	ENDF/B-V	Mo	420001	1321	CSRL-V
Graphite	6666	1306	ENDF/B-V	Ag-107	47107	1407	CSRL-V
C-12	6012	1306	ENDF/B-V	Ag-109	47109	1409	CSRL-V
N-14	7014	1275	ENDF/B-V	Cd	48000	1281	ENDF/B-V
N-15	70150	1307	ENDF/B-V				
O-16	8016	1276	ENDF/B-V	Eu	63000	9463	CSRL-V
F-19	90190	1309	ENDF/B-V	Hf	72000	1372	CSRL-V
Na-23	11023	1311	CSRL-V	Ta-181	73181	1285	CSRL-V
Mg	12000	1312	ENDF/B-V	Ta-182	73182	1127	CSRL-V
Al-27	13027	1313	CSRL-V	Pb	820001	1382	ENDF/B-V

CSRL-V Pointwise Cross Section Library.[3.7-3.8] It should be noted that data in the CSRL-V Library are based on the infinite dilute approximation.

Materials which were included in the representative ANS unit cell are listed in Table 3.4. The fuel region weighting function, Weighting Function "C," was generated for a homogenized representation of the cell. A plot of the spectrum is given in Fig. 3.2.

XLACS-77 also generates 1/E weighted data for each nuclide with fission and/or Maxwellian spectra at the appropriate extremities of the 1/E spectra. These 1/E weighted values are carried with the cross sections weighted over the spectrum selected for the set and are used by NITAWL-2 in its resolved resonance calculations.

3.5 Special ANSL-V GPN Data Sets

Certain data sets in the ANSL-V GPN Library were processed with nonstandard procedures. These data sets are described below:

Table 3.4. Representative ANS reactor unit cell

Region	Thickness (cm)	Constituents	Atom Density (atoms/bn * cm)
Heavy water	0.1270	D	6.55509E-2 3.27754E-2
Clad wall	0.0254	Al	6.02419E-2
Fuel matrix	0.0463	U-234	1.04683E-4
		U-235	9.78342E-3
		U-236	4.19709E-5
		U-238	5.66460E-4
		Si	6.99769E-3
		Al	3.44059E-2
Filler	0.0299	Al	4.36902E-2
Clad wall	0.0254	Al	6.02419E-2

Ortho- and Para-Hydrogen and Deuterium

A special procedure was developed for generating data to represent the kinematics of very low temperature scattering for the ortho- and para-forms of hydrogen and deuterium. The first step in the procedure was the generation of a thermal scattering kernel for a material using the MYDOL code. Since MYDOL has not been documented, a brief description of the theory included in the code is given here.

MYDOL is primarily based on the scattering model of Young and Koppel.[3.9] This model takes into account the spin corrections, rotations, and vibrations of the molecules. The vibrations are assumed to be harmonic and vibration-rotation coupling is ignored. The model assumes free translations for the molecules which is expected to be valid for neutron energies above 7 meV, the Debye temperature for hydrogen.

For neutron energies below the Debye temperature, the free translation part of the Young and Koppel scattering kernel was replaced in MYDOL by a diffusive model suggested by Egelstaff and Schofield.[3.10] Also, since coherent scattering contributes significantly to the deuterium scattering kernel, the convolution approximation for the dynamics of liquids has been implemented in the code.[3.11]

Output data from MYDOL were written in the special tabular format for ENDF/B-V File 6 data and merged into the appropriate ENDF/B-V evaluation. The last step in

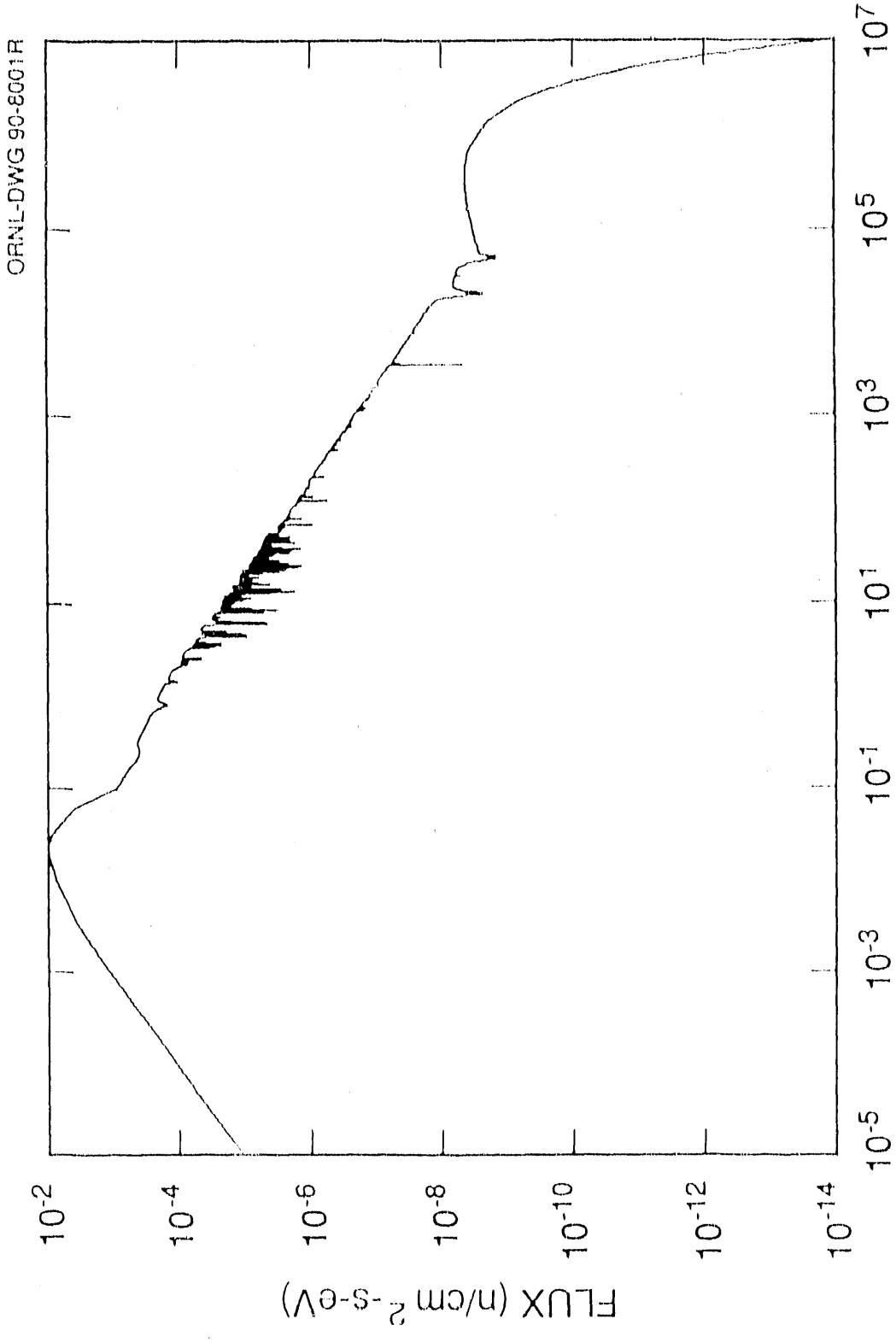

Fig. 3-2. ANSL-V Weight Function "C"

the process was to use the modified ENDF/B evaluation as input to XLACS-77 and to generate multigroup constants for the material of interest.

The upper energy for data generated with MYDOL is 1.00 eV. Consequently, this energy was the upper thermal energy boundary for the processed ortho- and para-forms of hydrogen and deuterium, giving 22 thermal groups in the ANSL-V 99-group structure for these nuclides.

He-4 Scattering Data

The ENDF/B-V evaluation for He-4, MAT 1270, does not include scattering data. To compensate for the lack of data, Cohen and Feynman's equation for the He-4 scattering kernel was developed into a computerized model.[3.12] The model was used to calculate scattering data, which were normalized to measured He-4 scattering cross sections, put into ENDF/B-V File 6 format, and included in the MAT 1270 evaluation. Finally, the modified evaluation was used in XLACS-77 to generate He-4 processed cross sections.

Cd-113

As is always the case, there was one exception to the renormalization procedure for the group and transfer matrices described in Sect. 3.2.5 above. That was for Cd-113. In this case, it was decided to force the average inelastic (MT=4) cross section to be the sum of the MT=51 through MT=91 matrix terms. This decision was made in order to accommodate certain data representations in the Cd-113 ENDF/B-V evaluation (MAT 1318) which present a problem to a processing code. To explain this action, one must first review a philosophy of ENDF.

ENDF evaluations require a "summed" process to contain the union of all energy meshes for the individual component parts in its energy mesh. At each of these energy points, the "summed" value is forced to be the sum of the individual values. However, it is not true that the "summed" process necessarily maintains this integrity internal to the energy mesh because of the ENDF interpolation schemes, e.g., cross section linear in log of energy or log of cross section linear in log of energy. Sums of processes which have cross sections linear in energy or log of energy are still linear functions in the selected independent variable, but the sum of the other types of functions are functions different from the component functions, i.e., the sum of log-log functions is not necessarily a log-log function.

An examination of the data for the ENDF/B-V evaluation for Cd-113 revealed that the MT=4 data specified linear-linear interpolation, whereas the MT=51, 52, 53, and 91 data all used log-log specifications. Since the summation of four log-log functions is clearly not linear and since one would suspect more attention would be given toward specifying the shape of the component functions of MT=4 data, it was decided to give precedence to the component functions.

None of the other ANSL-V nuclides showing any significant imbalance for inelastic processes was found to have used such obviously improper interpolation schemes, and

it was decided to normalize the sum of all the transfer matrices for the individual levels to the average of the total inelastic cross section for all other materials.

Al-27

As a result of general concerns about the quality of the ENDF/B-V evaluation for Al (MAT 1313) and a comparison of the JENDL-2 Al evaluation with MAT 1313 data, "minor" revisions were made to the in-house MAT 1313. The revised evaluation is identified herein as MAT 4313.

Specifically, the revisions were as follows:

Resolved resonance parameters from Mughaghab[3,13] were used to define the total, elastic scattering, and capture cross sections for the energy range 3-192 kev. For five small levels in the range 90-104 kev and a small level at 203.4 kev, no resonance parameters were given by Mughaghab. For these levels, "reasonable choices" were made which result in values of the capture area which are in agreement with values from Mughadghab. The negative level and the scattering radius were taken from the JENDL-2 evaluation for Al-27. The capture and total cross sections were revised in the range 1-3 kev and the elastic scattering and total cross sections were revised from 192 to 198 kev.

In order to minimize project costs, the previously generated Weight Function B for MAT 1313 was used in the XLACS-77 processing of MAT 4313. The revised evaluation was not used for any other aspect of the ANSL-V project; i.e., it was not used to generate Weight Function C.

Cs-134, Nd-147, Eu-152, Eu-154, and Eu-155

Since the fission product evaluations for Cs-134 (MAT 9663), Nd-147 (MAT 9768), Eu-152 (MAT 1292), Eu-154 (MAT 1293), and Eu-155 (MAT 9832) have been issued, more acceptable values for thermal cross sections and/or resonance integrals for the materials have been identified and documented. The evaluations were modified in-house with the documented data prior to processing for ANSL-V. The modified evaluations were assigned the following MAT numbers for in-house use: Cs-134 (MAT 4663), Nd-147 (MAT 4768), Eu-152 (MAT 4292), Eu-154 (MAT 4293), and Eu-155 (MAT 4832).

Natural Eu

The ENDF/B-V natural Eu evaluation, MAT 1463, was generated at BNL by combining data for Eu-151 and Eu-153. In the evaluation, the lower energy limits for the Eu-151 and Eu-153 unresolved resonance ranges are 98.81 and 97.22 eV, respectively.

The algorithm that UNRESR uses to generate an energy mesh does not allow for multi-isotope resonance nuclides with variable unresolved energy ranges. This potential problem was circumvented by setting the MAT 1463 resolved resonance upper limits and unresolved resonance lower limits of both Eu-151 and Eu-153 to

98.0 eV. For the purposes of this project, the modified evaluation was designated MAT 9463.

Lumped Fission Products

The methodology used to generate the two lumped fission product nuclides is described in Ref. 3.14. The ORIGEN[3.15] code was used to calculate fission product concentrations for a 2-week, 209,000 Mwd/tonne U fuel cycle. Those nuclides with absorption rates from $1(10^{15})$ to $1(10^{16})$ absorptions/m s were lumped into a moderately absorbing fission product. Those with absorption rates less than $1(10^{15})$ were lumped into a weakly absorbing fission product. The ICE[3.16] code was used to mix the fission product nuclides with concentrations weighted according to fission yield. The WORM program was used to convert the AMPX working library output from ICE to an AMPX master format. The CHANGER program was used to modify the energy boundaries to the 99 group structure. The MAD program was used to define the nuclide identification number and title.

3.6 Data Set Identification

ANSL-V GPN data sets are identified with a ZA notation, e.g., the data set for U^{235} is identified as "92235". Where additional data sets are generated for a material, the secondary data sets are identified with a suffix, e.g., 922351 for the second U^{235} data set.

The title card for each data set includes the following information: ANSL-V GPN LIBRARY, Material (Evaluation, Modification), comments. For instance, the title card for U^{235} as processed from ENDF/B-V MAT 1395 MOD 3 reads "ANSL-V GPN LIBRARY U235 (1395,3)."

3.7 NITAWL2 Processing of ANSL-V GPN Data

In earlier versions of NITAWL, several special sets of cross sections were carried with resonance nuclides. These were identified as follows:

MT	Value
1021	The 1/E averaged value of the (neutron,gamma) cross sections over the energy regions for which the Nordheim calculations would be made.
1022	The 1/E averaged value of the (neutron,fission) cross sections calculated as described above.
1023	The 1/E averaged value of the elastic scattering cross sections over the full resonance range.

In this case, the "total values" in MT=1, MT=18, MT=27, MT=101, and MT=102, corresponding to total, fission, absorption, capture, and (neutron,gamma), respectively, contained the weighted values for the cross sections for the energy range outside of the ranges over which the Nordheim calculation is made.

This was used with a procedure which added the "shielded" value to the "total" values to define a final cross-section value. The special cross sections, MT=1021 and MT=1022, were only used when no resonance calculation was requesed, thereby making a set of cross sections with infinite dilution values.

Elastic scattering was treated slightly differently in that MT=1023 contained the total infinite dilution value, while MT=2 contained everything except the "bodies" of the resonances to be calculated by the Nordheim treatment.

The primary disadvantage of the treatment just described is that it requires the preprocessing code to know enough about the Nordheim treatment to be able to know how to split the averages into the pieces described.

The formulation of the Nordheim treatment in NITAWL-2 references its shielded values to infinite dilution values, i.e.,

$$\sigma_{final} = \sigma_{infinite\ dilute} + \Delta\sigma \tag{3-4}$$

The new procedure, thereby, relieves the basic processing code of having to know anything at all about the Nordheim treatment. It also means that one can introduce additional resonance data to the calculation, if it is necessary. An example of when this might be required is the case of including $\ell=1$ resonance parameters in the Nordheim calculation, something that has not been done previously with AMPX-produced libraries. The new scheme does put an additional burden on the Nordheim treatment; viz., it now must determine infinite dilution averages, but this is a simple and straightforward process, compared with that of determining self-shielded values.

In the ANSL-V GPN Libraries, cross sections with MT numbers given by "3000 + MT" are used to store the reference infinite dilution values. For example, 3102 contains the infinite dilution values for MT=102. The values in MT=102 are those determined by weighting the cross sections over the flux used in the basic processing of the data and will not be used by the Nordheim treatment unless the MT in the "3000" range is not present.

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4.0 ANSL-V SECONDARY GAMMA-RAY PRODUCTION LIBRARY

The ANSL-V SGRP Library includes secondary gamma-ray production cross sections and/or multiplicities for the generation of gamma-rays resulting from neutron-induced reactions. The content of the library is given in Table 4.1. Data in the SGRP Library are in the ANSL-V 39-neutron 44-gamma-ray group structures listed in Tables 2.1 and 2.2, respectively.

4.1 Sources of Data for Generation of SGRP Library

The following guidelines were used in processing data for the SGRP Library:

1. Where appropriate data were available, multigroup cross sections and/or multiplicities were processed from the ENDF/B-V evaluations/modifications used to process the ANSL-V GPN data sets.

2. Of the ENDF/B-V materials which did not contain photon production data, the LENDL-84[4.1] library contains photon production data for nine of the materials, viz., B-11, Zr, Cd, Ag-107, Ag-109, Hf, U-233, U-234, U-236, and Pu-238. Thus, the LENDL-84 data were used to process appropriate SGRP Library data sets for the nine materials.

Since the ENDF/B-V evaluators chose not to include the LENDL-84 photon production data for the nine materials listed above in their evaluations, ANSL-V users should recognize a possible undefinable risk associated with these hybrid sets.

4.2 SGRP Library Calculational Scheme

With ENDF/B-V and LENDL evaluations as input, the AMPX module LAPHNGAS was used to process secondary gamma-ray multiplicities and/or cross sections in the ANSL-V 39-neutron and 44-gamma-ray group structures. LAPHNGAS was derived from D. J. Dudziak, et al., LAPH and LAPNANO codes and placed in the AMPX system in 1976.[4.2-4.4] The module has been used extensively at ORNL for the generation of secondary gamma-ray data.

LAPHNGAS retrieves photon production data from Files 12-15 of ENDF/B-IV and/or -V formatted evaluations and, depending on the photon production data available in evaluations and selections made by the LAPHNGAS user, calculates either multigroup secondary gamma-ray production cross sections (SGRPXS) and/or multigroup secondary gamma-ray yields (multiplicities). For resonance materials, LAPHNGAS was triggered to generate multigroup fission (ENDF MT=18) and capture (MT=102) multiplicities where the evaluations include such data.

Table 4.1. ANSL-V secondary gamma-ray production library materials.

Nuclide	SGRP Lib. ID	ENDF/B-V MAT	ENDF/B-V MOD	LENDL MAT	Processed MT's Yields	Processed MT's SGRPXS	Remarks
H-1	1301	1301	1			102	
H-2	1302	1302	2			102	
Be-9	1304	1304	2			102,741	
B-10	1305	1305	1			4,102,103,781	
B-11	7811			7811		16,51-53,91, 102,103,107	
C-12	1306	1306	2			51,102	
N-14	1275	1275	2			4,102-105,107	
N-15	1307	1307	1			4,16,103-105, 107	
O-16	1276	1276	2			4,22,102,103, 107	
Na-23	1311	1311	3		102	3,51	
Mg	1312	1312	1		102	3	
Al-27	1313	1313	1			4,28,102,103	
Si	1314	1314	3			4,22,28,102, 103,107	
K	1150	1150	1		102	51-67,91,102, 103,107	
Ti	1322	1322	1			3,102	
V	1323	1323	1			3,102	
Cr	1324	1324	2		102	3,4	
Mn-55	1325	1325	2		102	3	
Fe	1326	1326	3		102	3,51-61	
Co-59	1327	1327	3		102	3	
Ni	1328	1328	2		102	3	
Cu	1329	1329	1		102	3	
Zr	7841			7841		3,102	
Mo	1321	1321	1		102	3	
Cd	7847			7847		3,102	
Ag-107	7845			7845		3,102	
Ag-109	7846			7846		3,102	
Eu-151	1357	1357	1		102	3,51-59	
Eu-153	1359	1359	1		102	3,51-61	
Hf-nat	8305			8305			
Ta-181	1285	1285	2			3,102	
Pb	1382	1382	2			3,102	
Th-232	1390	1390	2		18,102	3	
U-233	7866			7866		3,18,102	
U-234	7867			7867		3,18,102	
U-235	1395	1395	3		18,102	3,4	
U-235	139507	1395	3		18,102, 181	3,4	Includes delayed fission gamma-ray data
U-236	7869			7869		3,18,102	
U-238	1398	1398	3		18,102	3	

Table 4.1. (Cont'd)

Nuclide	SGRP Lib. ID	ENDF/B-V MAT	ENDF/B-V MOD	LENDL MAT	Processed MT's Yields	Processed MT's SGRPXS	Remarks
U-238	139807	1398	3		18,102, 181	3	Includes delayed fission gamma-ray data
Pu-238	7875			7875		3,18,102	
Pu-239	1399	1399	2		18,102	3,4	
Pu-240	1380	1380	3		18,102	3,4	
Pu-241	1381	1381	2		18,102	3	
Pu-242	1342	1342	2		18,102	3	

4.3 Weighting Function

The default LAPHNGAS fission-1/E-Maxwellian weighting function was used to weight the neutron reaction cross sections (ENDF MF=3), photon production multiplicities (ENDF MF=12), and photon production cross sections (ENDF MF=13). In the weighting function, the fission and 1/E spectra were joined at 6.74×10^4 eV; 1/E and Maxwellian spectra were joined at 0.1265 eV. The mean neutron temperature of the Maxwellian distribution was 300 K. The temperature of the fission spectrum was 1.273 MeV.

4.4 Order of Scattering

Of the materials included in the library, only C-12 and O-16 ENDF/B-V evaluations contain anisotropic photon production data. Consequently, LAPHNGAS was triggered to produce P_6 scattering matrices for these materials and P_0 scattering data for all other materials.

4.5 Data Set Identification

Data sets produced by LAPHNGAS are identified with the MAT number of the ENDF/B formatted data which are processed. For the special cases where more than one SGRP data set is produced for a material, the AJAX module was used to assign unique identifiers to duplicate data sets.

The title for each data set includes the following information:

ANSL-V SGRP LIBRARY, Material (Evaluation, Modification), comments

For example, the title for U-235 as processed from the ENDF/B-V MAT 1395 MOD 3 reads "ANSL-V SGRP LIBRARY U235 (1395,3)."

4.6 Delayed Fission Photon Production Data

In earlier work, Hermann and Morrison used data from the ENDF/B-IV Fission Production Library with the ORIGEN code to determine delayed gamma-ray multiplicities, i.e., the integral of gamma-ray decay, for the sum of fission products of fissile isotopes over a 10^{13} second duration after a fission.[4.5] Included in their computed multiplicities were results for the thermal and fast neutron fission of U-235 and the fast fission of U-238, each set of multiplicities being given in the VITAMIN-C 36-group gamma-ray structure.[4.6]

A two-step procedure was used to include the Hermann-Morrison data in appropriate ANSL-V SGRP data sets. First, for ANSL-V groups which are contained within the VITAMIN-C groups, a linear ratio of group energy widths was used to apportion Hermann's and Morrison's multiplicities among the ANSL-V groups. Second, an AMPX AIM-UNITAB sequence introduced by R. W. Roussin in earlier work was used to place the delayed fission data into a copy of the previously-generated ANS SGRP U-235 and U-238 data sets.[4.7] Specifically, the data were placed in a separate one-dimensional array, the MT 181 array.

For U-235, the MT 181 array contains the adjusted Hermann-Morrison U-235 fast fission multiplicities in all neutron groups above 3.00 eV and the thermal fission multiplicities in all groups with an upper boundary less than or equal to 3.00 eV. For U-238, the MT 181 array contains the adjusted Hermann-Morrison U-238 fast fission multiplicities for all neutron groups. Note from Table 4.1 that the U-235 and U-238 SGRP data sets without the delayed fission data have identification numbers of 1395 and 1398, respectively. The U-235 and U-238 data sets with the delayed fission data have ID numbers of 139507 and 139807, respectively.

Special ANSL-V GPN U-235 and U-238 data sets, identified as Data Sets 9223507 and 9223807, respectively, are required for use with the SGRP data sets. (See Sect. 7.1.) In essence, the special data sets are the respective GPN data sets with infinite dilute fission cross sections placed in a separate one-dimensional MT 181 array. However, when the ANSL-V GPN, SGRP, and GRI libraries are used in a subsequent AMPX procedure to generate problem-dependent coupled cross sections, shielded fission cross sections are combined with the delayed fission multiplicities to give delayed fission secondary gamma-ray production cross sections.

4.7 Data Set Checks

As a first order check, each SGRP data set was checked with RADE, the AMPX module which checks cross-section data for consistency. No discrepancies were noted in the checking process.

4.8 Using ANSL-V SGRP Data Sets to Generate Coupled Neutron-Gamma Data Sets

To form problem-dependent coupled neutron-gamma cross sections, normally the following steps should be performed:

a. Given the ANSL-V GPN library, AJAX should be used to select the data sets of interest which contain unresolved Bondarenko factor data and the BONAMI module should be used to generate appropriate unresolved resonance data.

b. Either the CHOX or the UNITAB module is then used to combine processed ANSL-V GPN, SGRP, and GRI data sets of interest into a single AMPX master library.

c. NITAWL-2 should be used to perform the resolved resonance cross-section processing, combine the appropriate neutron reaction cross sections with secondary gamma-ray multiplicities to generate secondary gamma-ray production cross sections, and to produce the coupled data in a format which can be read by certain transport codes.

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5.0 ANSL-V GAMMA-RAY INTERACTION LIBRARY

The ANSL-V Gamma-Ray Interaction (GRI) Library includes cross sections for use in calculating the transport of photons. The table of contents for the library is given in Table 5.1. Gamma-ray group structure for the library is listed in Table 2.2.

5.1 Calculational Procedure

The SMUG module[5.1] of the AMPX cross-section processing system was used to produce data sets in the GRI Library. SMUG calculates multigroup photon cross sections with transfer coefficients represented by a Legendre approximation of specified order - P_6 for the GRI Library. Scattering moments were computed from the Klein-Nishina equation. Using DLC-99/HUGO photon interaction data[5.2] as input, SMUG used flat weighting to generate multigroup photo-electric and pair production cross sections.

SMUG was triggered to place gamma-ray energy absorption coefficients (i.e., kerma factor data, E_{gamma}* σ_{abs}, with units of barns*eV) in the MT 527 array of each GRI data set. Thus, when problem-dependent cross sections are subsequently prepared from the ANSL-V data, the kerma factor data are available to be inserted in the σ_{abs} table position of ANISN-formatted cross sections.

5.2 GRI Data Set Identification

GRI data sets are identified with the atomic number of the element; e.g., the data set for uranium has an ID number of "92." Each data set contains one- and two-dimensional cross section data for the processes listed in Table 5.2. Note that the processes are identified with the "MT" specifications of ENDF/B-V.[5.3]

The GRI Library is in AMPX master format. Service modules of the AMPX system can be used to prepare problem-dependent cross sections from master-formatted data and to perform a spectrum of data management operations, including data set listing, merging, plotting, conversion to formats required for transport codes, and other procedures.

5.3 GRI Library Data Checks

As a first order check, each ANSL-V GRI data set was checked with the RADE module. No discrepancies were noted in the checking process.

Table 5.1. ANSL-V Gamma-Ray Interaction Library Materials

Material	Data Set Identification	Material	Data Set Identification
H	1	Co	27
He	2	Ni	28
Be	4	Cu	29
B	5	Zr	40
C	6	Mo	42
N	7	Ag	47
O	8	Cd	48
Na	11	Xe	54
Mg	12	Sm	62
Al	13	Eu	63
Si	14	Hf	72
K	19	Ta	73
Ti	22	Ir	77
V	23	Pb	82
Cr	24	Th	90
Mn	25	U	92
Fe	26	Pu	94

Table 5.2. Reaction Data in ANSL-V GRI Data Sets

MT Number	One- or Two-Dimensional Data	Reaction
501	1	Total photon interaction cross section
502	1	Photon coherent scattering
504	2	Photon incoherent scattering
516	1	Pair production, nuclear and electron field (i.e., pair plus triplet production)
516	2	Pair production, nuclear and electron field
527	1	Gamma-ray energy absorption coefficients, i.e., kerma factors (same as MT 1527)
602	1	Photoelectric
1527	1	Gamma-ray energy absorption coefficients, i.e., kerma factors (same as MT 527)

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6.0 ANSL-V COUPLED NEUTRON-GAMMA LIBRARY

The ANSL-V Coupled Neutron-Gamma (CNG) Library includes cross sections for use in calculating the transport of both neutrons and photons. A table of contents for this library is given in Table 6.1.

6.1 Calculational Procedure

The UNITAB module of the AMPX cross-section processing system was used to produce the data sets in the CNG Libraray. UNITAB was used to combine cross-sections from the 39-Group General Purpose Neutron (GPN) Library, the 39/44-Group Secondary Gamma Ray Production (SGRP) Library, and the 44-Group Gamma Ray Interaction (GRI) Library.

6.2 CNG Data Set Identification

CNG data sets use the same identification numbers as those used for the GPN Libraries. The identification numbers for the CNG are given in Table 6.1. The user should note that the same material may appear more than once, since different weighting functions may be used for the General Purpose Neutron (GPN) Library.

6.3 CNG Library Data Checks

The RADE and DIAL modules can be used to check and selectively edit the contents of the CNG Library.

Table 6.1 ANSL-V Coupled Neutron-Gamma Library Materials

Material	CNG/GPN	SGRP	GRI	Material	CNG/GPN	SGRP	GRI
H-1	1001	1301	1	Mo	42001	1321	42
H-1	2222	1301	1	Ag-107	47107	7845	47
H-1	1111	1301	1	Ag-107	471071	7845	47
H-1	1801	1301	1	Ag-109	47109	7846	47
H-2	3333	1302	1	Ag-109	471091	7846	47
H-2	1802	1302	1	Cd	48000	7847	48
Be-9	4009	1304	4	Eu	63000	7852	63
B-10	5010	1305	5	Eu	630001	7852	63
B-11	5011	7811	5	Eu-151	631511	1357	63
C-12	6012	1306	6	Eu-153	631531	1359	63
C-12	6666	1306	6	Hf	72000	8305	72
N-14	7014	1275	7	Hf	720001	8305	72
N-15	70150	1307	7	Ta-181	73181	1285	73
O-16	8016	1276	8	Pb	820001	1382	82
Na-23	11023	1311	11	Th-232	902321	1390	90
Mg	12000	1312	12	U-233	922331	7866	92
Al-27	13027	1313	13	U-234	922341	7867	92
Al-27	130271	1313	13	U-235	922351	139507	92
Si	14000	1314	14	U-235(a)	922352	1395	92
Si	140001	1314	14	U-236	922361	7869	92
K	19000	1150	19	U-238	922381	139807	92
Ti	22000	1322	22	U-238(b)	922382	1398	92
V	23000	1323	23	Pu-238	942381	7875	94
Cr	24000	1324	24	Pu-239	942391	1399	94
Mn-55	25055	1325	25	Pu-240	942401	1380	94
Fe	26000	1326	26	Pu-241	94241	1381	94
Co-59	27059	1327	27	Pu-242	942421	1342	94
Ni	28000	1328	28				
Cu	29000	1329	29				
Zr	40000	7841	40				

(a) GPN identification is 922351.
(b) GPN identification is 922381.

7.0 BENCHMARK CALCULATIONS

7.1 Introduction

Criticality calculations have been performed for a variety of critical experiments in order to qualify the ANSL-V cross-section libraries[7.1] for use in analyzing the reactivity of the possible ANS Reactor core configurations following a severe accident. Most of the experiments used highly-enriched uranium (93.65 to 93.9 weight-percent ^{235}U) either as a metal or in light- or heavy-water solution. The reflected criticals had light or heavy water surrounding uranium metal spheres; heavy water surrounding uranium solutions moderated by heavy water; or light water surrounding uranium solutions moderated by light water. Validity of selected data from the ANSL-V Fine-Group General Purpose Neutron Library was satisfactorily tested in performance parameter calculations for the BAPL-1,[7.2] TRX-1, [7.2] and ZEEP-1[7.3] thermal reactor benchmarks. BAPL-1 is a H_2O moderated, uranium oxide lattice; TRX-1 is a H_2O moderated, 1.31 weight percent enriched uranium metal lattice; ZEEP-1 is a D_2O moderated, natural uranium lattice.

Problem-dependent cross sections were processed using a combination of code modules from the AMPX[7.4] and SCALE[7.5] systems. The automated cross-section generator portion of the Criticality Safety Analysis Sequence of SCALE was used in all cases to generate the cross sections. In a few cases the automated sequence also included the eigenvalue calculation, such as a few 1-D spherical transport calculations with the XSDRNPM code or the Monte Carlo calculations with the KENO code. Two-dimensional transport calculations with the DORT[7.6] code required the production of an ANISN-formatted cross-section library from the SCALE working library using the CONTAC module of AMPX.

7.2 Bare Enriched U-metal Sphere (Godiva)

The Godiva sphere was calculated with both the 39- and 99-group ANSL-V cross-section libraries using the parameters recommended in the CSEWG benchmark compilation[7.7] (i.e., S_{16}, P_3, and 40 mesh intervals) and the following uranium composition (also from the benchmark compilation):

Isotope	Atom density ($m^{-3} \times 10^{-30}$)
^{234}U	4.920-4
^{235}U	4.500-2
^{238}U	2.498-3

These numbers are consistent with values obtained using uranium with a density of 18.74 kg/l and respective isotopic weight percents of 1.02, 93.71, and 5.27 as found in Ref. 7.8. The ^{234}U percent was reported to remain around 1.0 with a few percent variation in the ^{235}U composition. Thus, about 1% ^{234}U was assumed for all other configurations (exactly 1.0 for the reflected metal spheres), since the ^{235}U percent composition varied only about 0.25% among all the experiments.

The critical radius of the Godiva sphere is 87.41 mm, giving an enriched uranium critical mass of 52.4 kg of which 49.1 kg is ^{235}U. The reevaluation of the Godiva experiment[7.8] led to upward adjustments in the critical mass (1%) and the effective uranium density (0.2%). Uncertainties in those parameters certainly affect the accuracy of the multiplication factor (k_{eff}) calculation. It is noted that the mass of ^{235}U in the Godiva experiment is nearly twice the inventory of the ANS Reactor core. Nevertheless, it is an important benchmark for the cross-section libraries used in the analysis of ANS Reactor accident analyses.

Calculated results for Godiva are compared with measured results in Table 7.1. The table shows the calculated k_{eff} along with ratios of spectrum averaged capture gamma production (n,γ) or fission (n,f) cross sections to the spectrum averaged ^{235}U fission cross section. The k_{eff} values calculated with XSDRNPM fall within one standard deviation of the measurement and are consistent with values calculated by others.[7.9] The mean value calculated by KENO is very slightly more than one standard deviation from the measurement, but the two error bands do overlap. The ratios of fission cross sections agree well with the measured values. However, the calculated ratios of the capture gamma cross sections to the ^{235}U fission cross section show greater deviation from the measured values, particularly in the case of ^{59}Co for which the results are 84% below the measured value. The reason for the discrepancy is not understood. Possible reasons include low capture gamma cross sections at high energy or a mistake in the published measured value. The deviation is judged not to be relevant to the current study. The values for ^{55}Mn are within two standard deviations of the measurement and can be considered in good agreement.

7.3 Light- and Heavy-water Reflected Enriched Uranium Spheres

Calculations were performed with XSDRNPM for three H_2O reflected spheres and five D_2O reflected spheres. A KENO Monte Carlo calculation was run for one of the D_2O reflected spheres because of concerns in the XSDRNPM calculations about the adequacy of mesh spacing at the interface between the metal sphere and the reflector. Calculations for one of the configurations showed virtually no difference in the calculated k_{eff} for a somewhat coarse and a very fine mesh at the interface. The heavy water experimental configuration specifications were empirically adjusted for the small holes in the centers of the spheres, while the light water configuration specifications were not. Therefore, the H_2O reflected configurations were modeled with small void spheres at the center.

The results are shown in Table 7.2 along with the Godiva result (0.0 mm reflector thickness) for comparison. The calculated values are very close to 1.0 (all within 0.7%). Note that the H_2O reflected critical masses are about the same for the thick and thin reflectors. The reason for this is that the H_2O reflector is effectively infinite in all cases. The thickness for the infinite reflector should be about 1.5 times the migration length.[7.10] For light water, this thickness is about 89 mm ($1.5\sqrt{(L^2 + \tau)}$ where L = 27.6 mm and τ = 2770 mm^2). Albedo data for slab geometry for several H_2O reflector thicknesses shows that the thermal-neutron albedo is the same for thicknesses greater than 200 mm.[7.11] It is not possible to extrapolate the data to the thinner 83-mm reflector thickness to determine if that reflector thickness is also nearly infinite as the measured and calculated results indicate. It is noted that the D_2O reflector thickness approximately equal to the thickness of the thin H_2O reflector results in about the same critical mass as

Table 7.1. Results for GODIVA Calculations

1-D Transport (XSDRNPM)

Parameter	Measured	39-group	99-group	Monte Carlo (KENO)
k_{eff}	1.00±0.003	0.9990(-0.1)[a]	0.9979(-0.21)	0.9965±.002
k_{∞}		2.3459	2.3552	
$\frac{^{59}Co(n,\gamma)}{^{235}U(n,f)}$	0.038±0.003	0.00604(-84.1)	0.00589(-84.5)	
$\frac{^{55}Mn(n,\gamma)}{^{235}U(n,f)}$	0.0027±0.0002	0.00314(16.4)	0.003(10.9)	
$\frac{^{233}U(n,f)}{^{235}U(n,f)}$	1.59±0.003	1.5535(-2.3)	1.5544(-2.2)	
$\frac{^{238}U(n,f)}{^{235}U(n,f)}$	0.1647±0.0018	0.1711(3.9)	0.1706(3.6)	

[a]Percent deviation from measurement.
NOTE: The critical mass is 52.4 kg (^{235}U mass is 49.1 kg).

does the H_2O reflector. Fitting the D_2O thermal-neutron albedo data of Ref. 7.11 to a polynomial allows one to extrapolate to a value of about 0.78 for an 83-mm-thick reflector. This compares to an albedo of 0.811 for the infinite H_2O reflector. Thus, the almost infinite 83-mm-thick H_2O reflector should have about the same albedo as the same thickness D_2O reflector and consequently about the same critical mass as shown. Since the D_2O albedo is 0.981 for an infinite reflector, the critical mass can be much smaller than that for an H_2O reflector.

7.4 Enriched Uranyl Nitrate-Light Water Solution Bare Spheres

Five ORNL critical spheres consisting of enriched uranyl nitrate in water in one of either of two spheres of radius 345.98 or 610.108 mm were analyzed. The first and fifth spheres contain the critical concentrations for unpoisoned solutions within the two spheres. In the other three, criticality is maintained by counter-balancing increases in the uranium concentration with increases in the natural boron concentration

Table 7.2. k_{eff} results for the light- and heavy-water reflected U-metal spheres

Light-Water Reflected Spheres				
Inner radius (mm)	Outer Radius (mm)	Critical Mass (kg ^{235}U)	Reflector Thickness (mm)	XSDRNPM k_{eff}
10.54	68.59	23.4	305	1.0029(1.0699)[a]
5.8	68.33	23.2	305	1.0031(1.0698)
5.8	68.63	23.5	82.6	0.9972(1.6050)

Heavy-Water Reflected Spheres				
Outer Radius	Critical Mass	Reflector Thickness	k_{eff}	
(mm)	(kg ^{235}U)	(mm)	XSDRNPM	KENO
87.410[b]	49.1	0.0	0.9990(2.3459)[a]	0.9965±0.0020
68.414	23.3	83.3	1.0064(2.2186)	1.0069±0.0023
65.556	20.5	116.6	1.0055(2.1991)	
63.916	19.0	139.7	1.0056(2.1755)	
61.710	17.1	173.7	1.0002(2.1538)	
56.893	13.4	388.6	1.0066(1.7552)	

[a] k_∞ shown in parentheses.
[b] GODIVA bare sphere.

in the solution. The atomic densities of nuclides in the solutions are shown in Table 7.3. These data were obtained from Ref. 7.12 which in turn obtained or derived these data from data in Ref. 7.13. Reference 7.12 also reported that the measured eigenvalues have an uncertainty less than 0.25%. The spheres were calculated using the SCALE automated Criticality Safety Analysis Sequences CSAS1X (BONAMI-NITAWL2-XSDRNPM) with the 39- and 99-group libraries and CSAS25 (BONAMI-NITAWL2-KENO) with the 39-group library. The results are shown in Table 7.4. The KENO result for ORNL-10 was obtained using spatially-dependent average weights based on the XSDRNPM fluxes. Based on 47,500 histories, the tabulated result is an improvement over the 0.9917±0.0018 obtained with 50,000 histories and without biasing. The XSDRNPM results are consistent with the Los Alamos National Laboratory ENDF/B-V calculated results and are better for spheres 3 and 10. They are also within the 0.25% uncertainty of the experiments. The KENO results are in good agreement with the measurements as their 1σ error bands overlap and the average calculated values lie within the ±2σ bands of the measurements.

Table 7.3 Atomic densities for the ORNL uranyl nitrate/water solution critical spheres.

	Atomic Densities ($m^{-3} \times 10^{-30}$)				
Sphere	ORNL-1	ORNL-2	ORNL-3	ORNL-4	ORNL-10
radius (mm)	345.948	345.948	345.948	345.958	610.108
nuclide					
^{234}U	5.3800-7[a]	6.3100-7	7.1600-7	7.6200-7	4.0900-7
^{235}U	4.8066-5	5.6206-5	6.3944-5	6.7959-5	3.6185-5
^{236}U	1.3800-7	1.6300-7	1.8400-7	1.9700-7	2.2000-7
^{238}U	2.8070-6	3.2810-6	3.7340-6	3.9670-6	1.9850-6
N	1.8690-4	2.1290-4	2.3920-4	2.5480-4	1.1160-4
H	6.6228-2	6.6148-2	6.6070-2	6.6028-2	6.6394-2
O	3.3736-2	3.3800-2	3.3865-2	3.3902-2	3.3592-2
^{10}B	0.0	1.0286-6	2.0571-6	2.5318-6	0.0
^{11}B	0.0	4.1714-6	8.3429-6	1.0268-5	0.0

[a]Read as 5.3800×10^{-7}.

Table 7.4 k_{eff} Results for ORNL Uranyl Nitrate/Water Solution Unreflected Critical Spheres.

Sphere	$H/^{235}U$ Atomic Ratio	Radius (mm)	XSDRNPM 39-Group	XSDRNPM 99-Group	KENO 39-Group
ORNL-1	1378	345.98	1.0025(1.2153)[a]	1.0012(1.2152)[a]	1.0046±0.0026
ORNL-2	1177	345.98	1.0023(1.2107)		
ORNL-3	1033	345.98	0.9993(1.2039)		
ORNL-4	971.6	345.98	1.0007(1.2043)		
ORNL-10	1835	610.108	1.0013(1.0745)		1.00031±0.0013

[a]k_∞ shown in parentheses.

7.5 Enriched Uranyl Fluoride-Heavy Water Solution Criticals

Six D_2O-reflected spheres and five bare cylinders filled with uranyl fluoride in heavy water solution were calculated using XSDRNPM and/or DORT mainly with the 39-group library, but selectively with the 99-group library. (In addition to the actual geometry specifications for the bare cylindrical assemblies, the experimenter also presented idealized bare spherical geometries for those assemblies.)[7.14] Sketches of the spherical and cylindrical geometries are shown in Figs. 7.1 and 7.2, and the experimental specifications are shown in Table 7.5. The atomic densities within the solutions are shown in Table 7.6. Most of the data were derived directly or indirectly from Ref. 7.14. Data on the glory hole dimensions and the fractions of H_2O in D_2O in the solutions and in the reflector were obtained from Ref. 7.15.

The reflected sphere calculations used a 3.18-mm-thick SS outer container wall rather than the 1.27-mm-thick wall shown in Fig. 7.1. The thickness was assumed to be the same as that for the cylinders, since it was not explicitly given. The wall thickness in Fig. 7.1 was later derived from data given in Ref. 7.14. The thicker wall (and slightly thinner reflectors) used in the calculations are not thought to have a significant impact on k_{eff}.

The calculated values of k_{eff} are shown in Table 7.7. The results, in general, deviate more from unity than did those for the other experiments. Some values would lie several standard deviations away from unity if the same uncertainty quoted for the ORNL spheres were used here. The author quotes less than 1% uncertainty in the critical masses and 2% in the $D/^{235}U$ ratios. These uncertainties were not translated into uncertainties in k_{eff}.

It is seen from the table that the 99-group results seem to be lower than the 39-group results by 0.004 to 0.005. It is also seen that DORT using the 2-D cylindrical mockups gave higher values than did XSDRNPM using the idealized bare sphere mockups. Therefore, one can conclude that the idealized bare spheres and the bare cylinders are not equivalent.

DORT-calculated ratios of spectrum-averaged microscopic cross sections are shown in Table 7.8. The measurements included results obtained from fission chambers and gamma product counting with significant differences between the two results in some cases. Comparisons in the table are made with the fission chamber results. The agreement is excellent for the ratio of the fission cross sections of ^{233}U and ^{235}U. The agreement is fair for the cadmium ratios for ^{233}U and ^{235}U except that for ^{233}U at $D/^{235}U=2081$. Part of the discrepancy may be due to the incomplete shielding of the foils by the cadmium. Uncertainty in the measured values may be responsible for other disagreement. The worst disagreement for ^{233}U may be due to a bad experimental value since the ratios of the ^{233}U and ^{235}U fission cross sections are in excellent agreement and the ^{235}U cadmium ratio is in line with those at other fuel concentrations.

Table 7.9 shows three-group flux fractions and microscopic fission cross sections. The energy ranges for the author's three groups are: (1) $E > 1.4$ MeV, (2) $0.6\ eV < E < 1.4$ MeV, and (3) $E < 0.6$ eV. The author's spectral fractions for $D/^{235}U=2110$ are 0.130, 0.754, and 0.116 based on ^{235}U fission cross sections of 1.3, 20, and 500 b, respectively, ^{238}U fission cross sections of 0.5, 0, and 0 b, respectively, and the measured values of the ^{238}U to ^{235}U fission cross-section ratio and the ^{235}U cadmium ratio. Notice that $D/^{235}U=2110$ rather than 2081 originally specified for the assembly. Whether this is an error or not, the slight difference in the ratio should not significantly alter the spectrum. The author's spectrum could not be reproduced using three sets of values for the

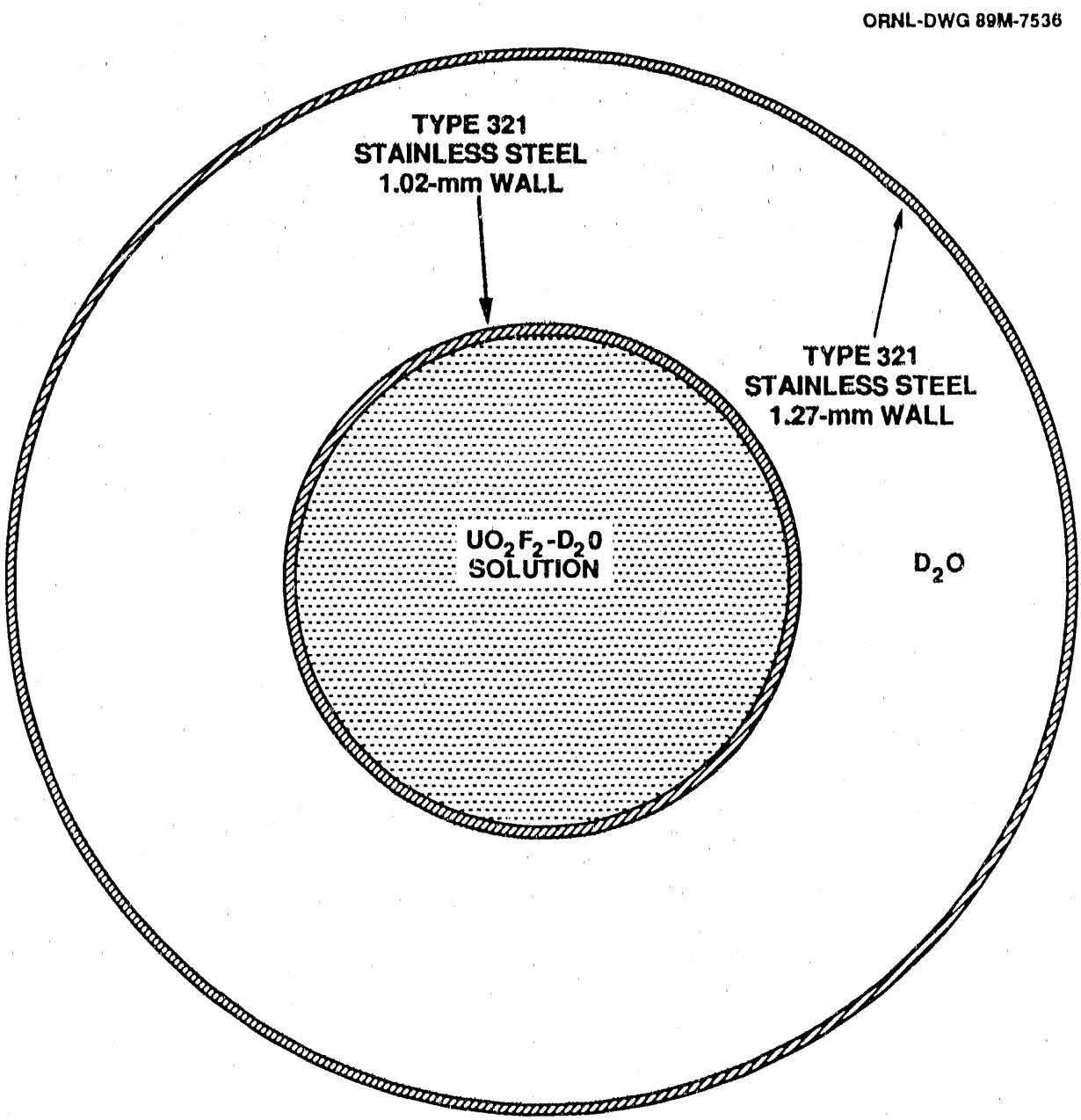

Fig. 7.1 Enriched Uranyl Fluoride Critical, Spherical Configuration.

ORNL-DWG 89M-7537

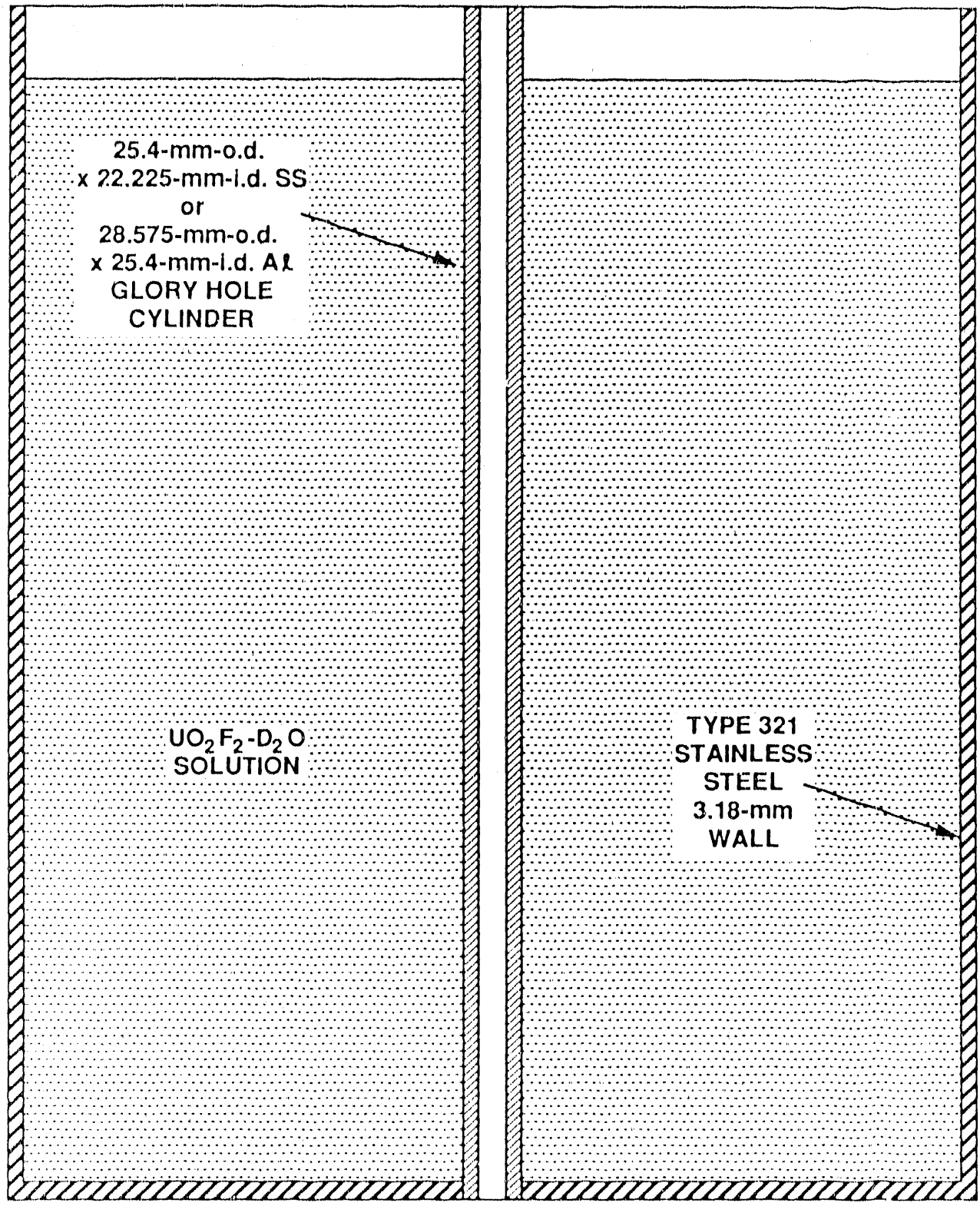

Fig. 7.2 Enriched Uranyl Fluoride Bare, Critical, Cylindrical Configuration.

Table 7.5. Specifications for the Uranyl Fluoride/Heavy Water Solution Critical Experiments

Spheres

Sphere No.	Critical radius (mm)	D/^{235}U atomic ratio	^{235}U density (kg/l)	Critical mass (kg ^{235}U)	D_2O reflector Thickness (mm)
1	171.45	34.2	0.679	14.2	271.78
2	184.75	53.7	0.443	11.6	259.08
3	196.85	81.2	0.302	9.6	246.38
4	209.55	135.3	0.185	7.0	233.68
5	222.25	243.0	0.104	4.8	220.98
6	234.95	431.0	0.0595	3.2	208.28

Cylinders

Cylinder No.	Solution Inner radius (mm)	Solution outer radius (mm)	Critical height (mm)	D/^{235}U Atomic ratio	^{235}U Density (kg/l)	Critical radius of ideal bare sphere (mm)	Bare sphere critical mass (kg ^{235}U)	Cylinder critical mass (kg ^{235}U)
1	12.7	316.0	714.5	230	0.1094	361.387	21.6	24.5
2	12.7	316.0	787.4	419	0.0610	369.724	12.9	15.0
3	12.7	381.0	610.9	856	0.0301	389.227	7.4	8.4
4	14.2875	381.0	608.3	856	0.0301	389.227	7.4	8.3
5	14.2875	381.0	847.5	2081	0.0124	433.018	4.2	4.8

Table 7.6. Atom Densities for the Uranyl Fluoride/Heavy Water Solution Critical Configurations

$D/^{235}U$ Atomic ratio	Atom densities ($m^{-3} \times 10^{-30}$)						
	H[a]	D	O	F	^{234}U	^{235}U	^{238}U
Spheres							
34.2	1.7852-4[b]	5.9328-2	3.3467-2	3.7135-3	1.9032-5	1.7400-3	9.7776-5
53.7	1.8288-4	6.0778-2	3.2903-2	2.4228-3	1.2417-5	1.1352-3	6.3792-5
81.2	1.8852-4	6.2651-2	3.3071-2	1.6517-3	8.4649-6	7.7389-4	4.3488-5
135.3	1.9243-4	6.3949-2	3.3083-2	1.0118-3	5.1854-6	4.7407-4	2.6640-5
243.0	1.9428-4	6.4566-2	3.2949-2	5.6879-4	2.9151-6	2.6650-4	1.4976-5
431.0	1.9715-4	6.5518-2	3.3183-2	3.2541-4	1.6678-6	1.5247-4	8.5680-6
Cylinders							
230	6.4479-4	3.2838-2	5.9832-4	3.0664-6	2.8034-4	2.8034-4	1.5754-5
419	6.5496-4	3.3082-2	3.3362-4	1.7098-6	1.5632-4	1.5632-4	8.7840-6
856	6.6025-4	3.3177-2	1.6462-4	8.4369-7	7.7133-5	7.7133-5	4.3344-6
2081	6.5464-2	3.3130-2	6.6817-5	3.4756-7	3.4756-7	3.1776-5	1.7856-6

[a] H_2O is ~0.3 mole % of D_2O in spheres and ~ 1.0 mole % in cylinders.
[b] Read as 1.7852×10^{-4}.

Table 7.7. k_{eff} Results for Uranyl Fluoride/Heavy Water Solution Criticals

	XSDRNPM	
$D/^{235}U$ Atomic ratio	39-Group	99-Group
34.2	1.0083(1.6212)[a]	1.0044(1.6210)[a]
53.7	1.0017(1.6244)	
81.2	1.0089(1.6382)	
135.3	1.0105(1.6608)	
243.0	1.0044(1.6868)	
431.0	0.9963(1.7049)	

Bare Cylinders

	XSDRNPM[b]		
$D/^{235}U$ Atomic ratio	39-Group	99-Group	DORT 39-Group
230	0.9994(1.8203)[a]	0.9945(1.8169)[a]	1.0050(1.7995)[c]
419	1.0007(1.8932)		1.0075(1.8636)
856	1.0089(1.9604)		1.0105(1.9223)
856	1.0089(1.9604)		1.0100(1.9266)
2081	1.0188(2.0022)		1.0234(1.9421)

[a] k_∞ shown in parentheses.
[b] As idealized bare spheres.
[c] Estimated k_∞ for 2-D cases.

Table 7.8. Ratios of Spectrum Averaged Microscopic Cross Sections for the Uranyl Fluoride/Heavy Water Bare Cylindrical Critical Assemblies.

	D/^{235}U Atomic Ratios			
Cross-section ratio	230	856	856	2081
$\bar{\sigma}_f(^{233}U)/\bar{\sigma}_f(^{235}U)$				
Measured (M)	1.78	1.29	1.29	1.11
Calculated (C)	1.74	1.27	1.24	1.08
C/M	0.98	0.98	0.96	0.97
$\bar{\sigma}_f(^{235}U)/\bar{\sigma}_f(^{238}U)$				
Measured	384	1123	1123	3180
Calculated	523	1827	1928	4619
C/M	1.36	1.63	1.72	1.45
$\bar{\sigma}_f(^{235}U)/\bar{\sigma}_f^{Cd}(^{233}U)^a$				
Measured	2.01	4.79	4.79	12.4
Calculated	2.37	6.15	6.64	14.1
C/M	1.18	1.28	1.39	1.14
$\bar{\sigma}_f(^{233}U)/\bar{\sigma}_f^{Cd}(^{233}U)^a$				
Measured	1.43	2.51	2.51	11.5
Calculated	1.52	2.84	3.01	5.54
C/M	1.07	1.13	1.20	0.48

[a]Cadmium ratio (i.e., ratio of the spectrum-averaged fission cross section without a cadmium cover over the foil to that with a cadmium cover. The cadmium is assumed to essentially reduce to zero the fraction of fissions in the foil due to neutrons with energies less than 0.6 eV).

Table 7.9. Three-Group Flux Fractions and Fission Cross Sections for the Uranyl Fluoride/Heavy Water Bare Cylindrical Critical Assemblies

Case	$D/^{235}U$ Atomic ratio	Group[a]	Flux fraction	σ_f(bares) ^{233}U	^{235}U	^{238}U
1	230	1	0.0936	1.83	1.24	0.521
		2	0.8421	34.3	12.7	1.92-3[b]
		3	0.0643	237	228	2.30-6
2	419	1	0.0847	1.83	1.24	0.521
		2	0.8067	38.4	14.0	1.81-3
		3	0.1086	272	270	2.66-6
3	856	1	0.0739	1.83	1.24	0.521
		2	0.7447	41.9	15.3	1.71-3
		3	0.1814	318	325	3.13-6
4	856	1	0.0745	1.83	1.24	0.521
		2	0.7347	41.9	15.2	1.72-3
		3	0.1908	325	333	3.21-6
5	2081	1	0.0588	1.83	1.24	0.522
		2	0.6117	45.0	16.3	1.62-3
		3	0.3295	381	399	3.77-6

[a]Groups: (1) E > 1.4 MeV; (2) 0.6 eV < E ≤ 1.4 MeV; and (3) E ≤ 0.6 eV.
[b]Read as 1.92×10^{-3}.

Table 7.10. Comparison of Unfolded and Calculated Three-Group Flux Fractions for the Uranyl Fluoride/Heavy Water Bare Cylindrical Critical Assembly with a $D/^{235}U$ Atomic Ratio of 2110

		Unfolded flux fractions			
Group	Calculated flux fractions	$r_1 = 3128$[a] $r_2 = 11.95$[a]	$r_1 = 3180$ $r_2 = 12.4$	$r_1 = 3128$ $r_2 = 10.6$	Author's values
1	0.0588	0.0967	0.0974	0.0897	0.130
2	0.6117	0.6263	0.6180	0.6561	0.754
3	0.3295	0.2771	0.2847	0.2542	0.116

[a]Definitions:

$$r_1 = \frac{\bar{\sigma}_f\,(^{235}U)}{\bar{\sigma}_f\,(^{238}U)} = \frac{\sigma_{f1}\,(^{235}U)\phi_1 + \sigma_{f2}\,(^{235}U)\phi_2 + \sigma_{f3}\,(^{235}U)\phi_3}{\sigma_{f1}\,(^{238}U)\phi_1}$$

$$r_2 = CR(^{235}U) = \frac{\sigma_{f1}\,(^{235}U)\phi_1 + \sigma_{f2}\,(^{235}U)\phi_2 + \sigma_{f3}\,(^{235}U)\phi_3}{\sigma_{f1}\,(^{235}U)\phi_1 + \sigma_{f2}\,(^{235}U)\phi_2}$$

Where the group fluxes, ϕ_i, satisfy

$$\phi_1 + \phi_2 + \phi_3 = 1.0.$$

Where the σ_{fi} are the group fission cross sections and CR is the cadmium ratio. The three equations are used to unfold the flux fractions.

measured results. As shown in Table 7.10, the unfolded spectra using those data and the author's cross sections more nearly resemble the calculated spectra than the author's unfolded spectra even though the calculated average fission cross sections for groups 2 and 3 are more than 20% higher (see Table 7.8).

7.6 Enriched Uranyl Fluoride-Light Water Solution Spheres

Five supplemental ORNL reflected and bare critical spheres consisting of enriched uranyl fluoride in water were analyzed. These experiments, labeled L7 to L11, supplemented the ORNL uranyl nitrate spheres in that a wider range of $H/^{235}U$ ratios was introduced and reflected spheres were included. The specifications for the experiments[7.16] are shown in Table 7.11. All cases were calculated with both the ANSL-V 39- and 99-group libraries using the SCALE System Criticality Safety Analysis Sequence CSASN to process the cross sections and XSDRNPM to perform the k-calculation. The calculated results are shown in Table 7.12. The k_{eff} values compare favorably with calculated ENDF/B-V values reported in Ref. 7.16 (i.e., 1.0056, 1.0058, 1.0100, 1.0056, and 1.0053, respectively). In addition, some of the calculated reaction-rate ratios also compare favorably with those presented in Ref. 7.16.

7.7 Establishing A Bias for the Calculated k_{eff}

Since for a critical system, $k_{eff}=1.0$, deviations of the calculated values from unity indicate some bias in the calculational methods and/or data. This assumes, of course, that the experimental geometry is critical. The calculated k_{eff} values for the lumped and solution critical experiments have been separated and a mean and standard deviation have been computed for each. For the lumped criticals, the average of nine values is 1.0039 with a standard deviation of 0.0027, while for the solution criticals the average of twenty (20) values is 1.0052 with a standard deviation of 0.0055. Now for the solution criticals, one value is around 1.02, a significant deviation from 1.0. For twenty measurements, Price[7.17] data on Chauvenet's criterion for rejecting data indicates that a value may be rejected if its deviation is more than 2.23 standard deviations from the mean. The above value is 2.47 standard deviations from the mean and is therefore rejected. The remaining nineteen values then give an average k_{eff} of 1.0045 with a standard deviation of 0.0046. Since the average calculated values are above 1.0, the more conservative deviations would be those about 1.0. The standard deviations about unity are 0.0050 for the lumped criticals and 0.0064 for the solution criticals.

To assure subcriticality of the ANSR core configurations, the calculated k_{eff} values should be below established limits. Since in a normal distribution 99% of the values fall within $\pm 3\sigma$ about the mean, the established limits are set at $k_{avg} - 3\sigma - 0.02$ (where 0.02 has been subtracted for extra shutdown margin). Thus, using the more conservative deviations about unity, one arrives at a bias k_{eff} of 0.965 for the lumped criticals and 0.9653 for the solution criticals. A single value of 0.965 can be used for all configurations, since these two values are so close. Any configuration with k_{eff} greater than 0.965 would require some mechanism to introduce negative reactivity or to prevent the core material from reconfiguring in that fashion.

Table 7.11. Specifications for the ORNL Uranyl Fluoride-Light Water Critical Spheres.

	Sphere Designation				
	L7	L8	L9	L10	L11
Radius (mm)					
Al shell	115.176	279.132	346.327	118.442	279.132
Thickness (mm)	1.6	2.0	3.2	1.6	2.0
H_2O-reflected?	yes	no	no	yes	yes
kg ^{235}U	2.08	2.13	3.25	1.39	1.86
$H/^{235}U$	76.1	1112	1393	126.5	1270
k_{eff} (exp)	1.0000	1.0004	1.0000	1.0000	0.9999
	Atom Densities ($m^{-3} \times 10^{-30}$)				
Solution					
H	6.3368-2[a]	6.6614-2	6.4729-2	6.4729-2	6.6436-2
O	3.3471-2	3.3436-2	3.3434-2	3.3463-2	3.3330-2
F	1.7873-3	1.2855-4	1.0270-4	1.0983-3	1.2338-4
^{234}U	8.7577-6	7.3270-7	5.8540-7	5.3816-6	5.5020-7
^{235}U	8.3269-4	5.9905-5	4.7856-5	5.1169-4	5.2312-5
^{236}U	4.2895-6	0.0	0.0	2.6359-6	2.6950-7
^{238}U	4.7899-5	3.6380-6	2.9063-6	2.9434-5	3.0091-6
Al Shell					
Al	6.0250-2	6.0250-2	6.0250-2	6.0250-2	6.0250-2
H_2O Reflector					
H	6.6688-2	6.6688-2	6.6688-2	6.6688-2	6.6688-2
O	3.3344-2	3.3344-2	3.3344-2	3.3344-2	3.3344-2

[a]Read as 6.3368×10^{-2}.

Table 7.12. Calculated Results for the ORNL Uranyl Fluoride-Light Water Critical Spheres

	Sphere Designations				
	L7	L8	L9	L10	L11
99 Groups					
k_{eff}	1.0076	1.0090	1.0056	1.0057	1.0054
K_∞	1.0550	1.3256	1.2177	1.0512	1.0194
Core leakage	0.4695	0.2390	0.1744	0.4661	0.2038
ρ^{25}	0.5184	7.051-2[a]	6.025-2	0.3354	6.107-2
ρ^{28}	33.86	4.983	4.267	22.56	4.325
δ^{25}	0.1980	2.499-2	2.085-2	0.1251	2.161-2
δ^{28}	3.180-3	3.539-4	2.965-4	1.998-3	2.988-4
C*	0.1448	2.795-2	2.466-2	0.1029	2.493-2
$\phi epi/\phi tot$[b]	0.8951	0.5438	0.5029	0.8489	0.5061
39 Groups					
k_{eff}	1.0090	1.0103	1.0068	1.0069	1.0069
k_∞	1.0547	1.3257	1.2178	1.0508	1.0201
Core leakage	0.4691	0.2381	0.1735	0.4657	0.2028
ρ^{25}	0.5104	6.976-2	5.962-2	0.3307	6.043-2
ρ^{28}	33.89	4.980	4.265	22.55	4.323
δ^{25}	0.1951	2.476-2	2.111-2	0.1235	2.141-2
δ^{28}	3.165-3	3.500-4	2.928-4	1.988-3	2.494-2
C*	0.1453	2.794-2	2.466-2	0.1030	2.494-2
$\phi epi/\phi tot$[b]	0.8944	0.5428	0.5018	0.8482	0.5052

[a]Read as 7.051×10^{-2}.
[b]Ratio of the epithermal to total neutron flux. Thermal cutoff is 0.588 eV.

Note:

ρ^{25} is the ratio of epithermal to thermal capture in ^{235}U.
ρ^{28} is the ratio of epithermal to thermal capture in ^{238}U.
δ^{25} is the ratio of epithermal to thermal fission in ^{235}U.
δ^{28} is the ratio of the ^{238}U fission rate to the ^{235}U fission rate (microscopic cross sections).
C* is the ratio of the ^{238}U capture rate to the ^{235}U capture rate (microscopic cross sections).

7.8 BAPL-1, TRX-1, and ZEEP-1

Using the 99-group data as input, benchmark calculations were done using the SCALE[7.5] sequence: AJAX-CSASN-XSDRNPM-XSDRNPM. The first XSDRNPM case was a discrete ordinates calculation (S_8,P_3) to cell weight cross sections. In the second XSDRNPM case, the cell-weighted cross sections and the "B-3" option with a "height" which corresponded to the total buckling of each lattice were used to calculate benchmark performance parameters.

Results of the benchmark calculations are given in Table 7.13. Results for the low-enriched uranium benchmarks, TRX-1 and BAPL-1, are in good agreement with those of other data testers as reported in Ref. 7.18. Results for the natural-uranium, heavy-water benchmark, ZEEP-1, are in reasonable agreement with that of Craig as reported in Ref. 7.3. Calculated k-eff for BAPL-1 is 0.12% low; DELTA28 (U-238 fissions/U-235 fissions) is 3.33% low but is within one standard deviation. Calculated k-eff for TRX-1 is 0.64% low. CR (U-238 capture/U-238 fissions) is 1.25% high. DELTA28 is slightly less than one standard deviation relative to the measured parameter. RHO28 (epithermal/thermal U-238 capture) significantly exceeds the benchmark value. Calculated k-eff for ZEEP-1 is 0.09% low; DELTA28 is 0.89% high. In Ref. 7.19, it is shown that for selected changes in U-238 resolved resonance parameters (e.g., 2% increase in the S-wave capture width) a difference of 1.00% in CR will correspond to a difference of about 0.33% in k-eff for BAPL-1 and TRX-1 (hereinafter called the one-third rule). It was shown that the one-third rule holds for ZEEP-1 by increasing U-238 capture cross sections by a constant 2.0% between 1.86 and 275 eV and then comparing CR and k-eff with the unmodified case.

7.9 Summary

Calculations have been performed for a series of critical experimental configurations to qualify the ANSL-V cross-section libraries for use in the analysis of the ANS Reactor. The experiments lay at the extremes of possible core configurations following a core-disruptive accident. In general, the libraries performed well in the calculations, giving k_{eff} values near 1.0 and in some cases reproducing well the measured spectral parameters. Thus, the libraries are judged to be suitable for performing the criticality calculations at moderately low temperatures. High-temperature applications would require benchmarking against a high-temperature experiment. For those cases calculated, the 99-group library gave lower k_{eff} values than did the 39-group library (about 0.001 for GODIVA and the uranyl nitrate- and fluoride-light water solutions and 0.004 to 0.005 for the uranyl fluoride-heavy water solutions).

Table 7.13. TRX-1, BAPL-1, and ZEEP-1 Benchmark Results

			Calculated % Difference[b] or Parameter Value	
	Parameter	Measured[a]	Craig[c]	ANSL-V
BAPL-1	k-eff	1.0000	-0.02	-0.12[d]
	DELTA25[e]	0.084±0.002 (±2.4%)	0.71	0.36[d]
	DELTA28[e]	0.078±0.004 (±5.1%)	-1.67	-3.33[d]
	RHO28[e]	1.39±0.01 (±0.72%)	2.95	3.67[d]
	CR[e]	f	0.815	0.817[g]
	LCF[h]		0.873	0.874[g]
	NFL[i]		0.127	0.126[g]
TRX-1	k-eff	1.0000	-0.42[d]	-0.64[d]
	DELTA25[e]	0.0987±0.0010 (±1.0%)	2.03[d]	1.11[d]
	DELTA28[e]	0.0946±0.0041 (±4.3%)	4.55[d]	4.12[d]
	RHO28[e]	1.320±0.021 (±1.6%)	3.41[d]	5.61[d]
	CR[e]	0.797±0.008 (±1.0%)	0.25[d]	1.25[d]
	LCF[h]		0.842[g]	0.843[g]
	NFL[i]		0.158[g]	0.157[g]
ZEEP-1	k-eff	1.0000±0.002 (±0.2%)	0.36[d]	-0.09[d]
	DELTA28[e]	0.0675±0.0014 (±2.1%)	1.04[d]	0.89[2]
	RHO28	f	0.282[d]	0.290[d]
	CR[e]	j	0.838[g]	0.842[g]
	RCR[e]	1.260±0.005 (±0.4%)	1.67[d]	2.12[d]
	LCF[h]		0.810[g]	0.809[g]
	NFL[i]		0.190[g]	0.191[g]

Table 7.13. (Cont'd)

Parameter	Measured[a]	Calculated % Difference[b] or Parameter Value	
		Craig[c]	ANSL-V

[a]BAPL-1 and TRX-1 values are given in Ref. 7.1; ZEEP-1 values are given in Ref. 7.2.
[b]Percent difference - ((calculated-measured)/measured) x 100.
[c]D. S. Craig reported benchmark testing for 14 criticals, including the three benchmarks of interest in this study.[7.2]
[d]Value expressed in percent difference.
[e]Reaction rate ratio definitions:
DELTA25 - ratio of epithermal-to-thermal U-235 fissions
DELTA28 - ratio of U-238 fissions to U-235 fissions
RHO28 - ratio of epithermal-to-thermal U-238 captures
CR - U-238 captures/U-235 fissions
RCR - [U-238 captures/U-235 fissions (in the cell)]/ (same ratio in a Maxwellian at the temperature of the moderator. (The calculated CR in a Maxwellian spectrum at 20°C is 0.6541.)
[f]Not reported.
[g]Calculated parameter.
[h]LCF is the "leakage correction factor" which is defined as k-eff/k-inf.
[i]NFL is the "net fractional leakage" and is defined as (k-inf - k-eff)/k-inf. NFL is also numerically equal to (1 - LCF).
[j]Not reported (see RCR in footnote e).

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8.0 COMMENTARY

8.1 Neptunium-239 Dataset

The only available, reasonably complete Np-239 evaluation was found to be in the Japanese Evaluated Nuclear Data Library (JENDL2) -- MAT 2932.[3.3] The library is available in ENDF/B-IV format from RSIC as DLC-122 and is documented in Ref. 3.4.

In reviewing MAT 2932, it was noted that the thermal capture cross section was 37 barns and was based on very old data given in Ref. 3.14. The more recent evaluation of Mughabghab (Ref. 3.13) expresses the cross section as follows:

$$\sigma_c(E) = 77.0 * \sqrt{E_0} / \sqrt{E}$$

where E is the energy in eV and E_0 = 0.0253. For the ANSL-V project, the Mat 2932 capture cross section was revised below 4.0 eV per Mughabghab's expression. The total cross section was modified to be in agreement with the sum of the elastic and the revised capture cross section and, finally, the format of the evaluation was converted to ENDF/B-V. For the purposes of the ANS project, the modified evaluation was designated MAT 2932, Mod. 1.

8.2 First-Order Data Checks

Specified reaction cross sections for selected high-priority ANSL-V 39-group materials were to be plotted and, where possible, compared with plots of appropriate ENDF/B-V and/or CSRL-V pointwise data. Specified plots are listed in Table 8.1.

In identifying GPN data to be plotted, probable use of materials, location in the reactor, probable neutron energy spectrum incident on it, and resulting reactions with the largest cross sections for that material were considered. Energy regions to be plotted were selected to overlap at least two ANS broad groups and contain some resonance structure or threshold behavior. An energy region bracketing the thermal value at 0.0253 eV was selected for each non-threshold reaction. Future versions of the library should include these plots.

8.3 NITAWL-2

A plan for validating NITAWL-2 should be developed, tests of the module conducted, and the tests documented. The module only exists in SCALE-4. The module should be included in the AMPX package which is distributed by RSIC.

Table 8.1 ANSL-V GPN Data to be Plotted

Nuclide	ANSL-V GPN ID No.	ENDF/B MF	ENDF/B MT	Range to be plotted Energy	Range to be plotted Groups	Remarks
H-1	1001	3	2	0.003 eV -0.0 eV 100 keV - 1.5 MeV	32-27	Free Gas
H-1	1801	3	2	0.003 eV - 0.1 eV 100 keV - 1.5 MeV	32-27 7-4	H_2O bound @ 296 K kernel
H-1	2222	3	2	0.003 eV - 0.1 eV 100 keV - 1.5 MeV	32-27	Ortho
H-1	1111	3	2	0.003 eV - 0.1 eV 100 keV - 1.5 MeV	32-27 7-4	Para
H-2	1802	3	2	0.003 eV - 0.1 eV 100 keV - 1.5 MeV	32-27 7-4	D_2O bound @ 296 K kernel
H-2	3333	3	2	0.003 eV - 0.1 eV 100 keV - 1.5 MeV	32-27 7-4	Para
He-3	2003	3 3	2 103	0.003 eV - 0.1 eV 2 keV - 100 keV	32-27 10-8	
He-4	2004	3	2	0.003 eV - 0.1 eV 90 keV - 4 MeV	32-27 8-2	w/special scat. data
B-10	5010	3 3	2 107	0.003 eV - 0.1 eV 2 keV - 100 keV	32-27 10-8	
B-11	5011	3	2	0.003 eV - 0.1 eV 100 keV - 900 keV	32-27 8-6	
Graphite	6666	3	2	0.003 eV - 0.1 eV 10 keV - 100 keV 900 keV - 3 MeV	32-27 9-8 6-3	296 K kernel
O-16	8016	3	2	0.003 eV - 0.1 eV 900 keV - 3 MeV	32-27 6-3	
Mg	12000	3	2	0.003 eV - 0.1 eV 17 keV - 400 keV	32-27 8-7	
Al-27	13027	3 3 3	2 4 107	0.003 eV - 0.1 eV 550 eV - 100 keV 900 keV - 6.434 MeV 3 MeV - 20 MeV	32-27 10-8 5-2 2-1	
Si	14000	3 3 3	2 103 107	0.003 eV - 0.1 eV 17 keV - 400 keV 3 MeV - 20 MeV 3 MeV - 20 MeV	32-27 8-7 2-1 2-2	

Table 8.1 (Cont'd)

Nuclide	ANSL-V GPN ID No.	ENDF/B MF	ENDF/B MT	Range to be plotted Energy	Range to be plotted Groups	Remarks
Cr	24000	3	2	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
				550 eV - 17 keV	10-9	Data set 1324
		3	4	400 keV - 1.4 MeV	6-5	
Fe	26000	3	2	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
				100 eV - 100 keV	11-8	Data set 1326
		3	102	100 eV - 100 keV	11-8	Plot vs CSRL-V
						Data set 1326
		3	4	900 keV - 6.434 MeV	5-2	
Zr	40000	3	2	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
				100 eV - 100 keV	11-8	Data set 1340
				0.1 eV - 1 eV	27-18	Plot vs CSRL-V
						Data set 1340
U-233	922331	3	18	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
				0.1 eV - 10 eV	27-14	Data set 1393
		3	102	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
				0.1 eV - 10 eV	27-14	Data set 1393
U-234	922341	3	102	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
				1 eV - 30 eV	17-13	Data set 1394
		3	2	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
				1 eV - 30 eV	17-13	Data set 1394
U-235	922351	3	18	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
				1 eV - 30 eV	17-13	Data set 1395
		3	102	1 eV - 30 eV	17-13	Plot vs CSRL-V
						Data set 1395
U-236	922361	3	2	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
						Data set 1396
		3	102	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
						Data set 1396
U-238	922381	3	3	0.003 eV - 0.1 eV	32-27	Plot vs CSRL-V
				100 keV - 550 eV	11-11	Data set 1398
				3 keV - 17 keV	9-9	
		3	102	0.003 eV - 0.1 eV	32-27	
				100 keV - 550 eV	11-11	
				3 keV - 17 keV	9-9	

8.4 MYDOL Modifications

A coherent scattering model should be added to MYDOL for hydrogen. Although coherent scattering is not as important for hydrogen as it is for deuterium, it would increase the accuracy of the calculated cross sections at lower neutron energies. It is assumed that the coherent scattering model for deuterium could be used for hydrogen. The task would essentially be to find the static structure factor for hydrogen in the literature and modify the code.

The diffusive scattering model by Egelstaff should be replaced by the more sophisticated model by Utsuro. [8.1]

With the current version of MYDOL, cross sections for deuterium have been improved slightly but calculated and measured data still disagree at energies around 1 meV. This is due to the large coherent scattering contribution. This discrepancy should be addressed by implementing the convolution approximation in MYDOL for the dynamics of liquids which accounts for intermolecular interferences.

8.5 Liquid He-4

A code to generate the scattering kernel for liquid helium based on the model by Dietrich was initiated by M. W. Waddell but never completed.

8.6 New Graphite Thermal Scattering Law Data

It is noted that RSIC has recently received a pre-release copy of an ENDF/B evaluation for thermal scattering law data for graphite, i.e., a modification to MAT 1065. At this time, there is no known use of the data within the Oak Ridge community. It is recommended that an ANSL-V graphite data set be generated with the new evaluation and compared with ANSL-V data set 6666.

8.7 Neutron Kerma Factors

Neutron heating rates in various parts of the ANS facility will, in general, be less than gamma heating rates. However, two components, the cold source and the pressure vessel, will have such high heat loads that small increments in the heating rate may be important. Moreover, since the heating rate is largely controlled by the fast flux which has a strong spatial variation in the reflector, information regarding the neutron heat flux may influence the position of the cold source with respect to the reactor core. Likewise, heat deposition location in the reflector due to the moderation may influence cooling flow patterns.

Project management has expressed an interest in including neutron kerma factors in the ANSL-V libraries for the following nuclides to enable the effects noted above to be studied:

Para-hydrogen (20K)
Ortho-hydrogen (20K)
Cold (20K) deuterium
Aluminum (20, 150, 300, 600, 900, 1200K)
Magnesium (20, 150, 300K)
Normal (300K) deuterium
Oxygen (300K)

8.8 Unresolved Resonance Data Processing

Upon completion of the ANS libraries, a review of the ENDF data files showed that ENDF/B-V has unresolved resonance data for natural zirconium, Zr-91, Rh-103, Cd-113, Sm-149, Np-238, and Cm-245. However, no Bondarenko factors for these nuclides are included in the ANS libraries. Future work should include the processing of unresolved resonance parameters for these nuclides.

8.9 Cost

The ANS cross section libraries project was instigated in June 1986, temporarily halted from October 1987 to October 1989, restarted in October 1989, and completed in March of 1990. Costs per fiscal year were $20,609, $173,453, and $44,317 for fiscal years 1986, 1987, and 1990 respectively.

REFERENCES

8.1. M. Utsuro, Z. Phys. B27, P. 111 (1977).

APPENDIX A
AMPX LIBRARY FORMATS

1. Header information - written on the front of the library to specify the number of neutron and/or gamma groups, the number of nuclides, etc., contained in the library. (Record Type 1)

2. Energy structure information - contains the group boundaries. (Record Type 2)

3. Nuclide directory information - 50 words which give a title for the nuclide along with other parameters which specify the kinds of information included for the nuclide, such as number of records in the library for the nuclide and how much neutron and gamma data are given. (Record Type 3)

4. Resonance parameter data - provisions for resolved resonance parameters in the single level Breit-Wigner (SLBW) or multi-level Breit Wigner (MLBW) formats. Average parameters for an unresolved calculation may also be included. (Record Type 4)

5. Bondarenko data - four record types are used for this information:

   a. A record which gives the values of σ_0 and T at which the factors are tabulated, along with cutoff energies for the Bondarenko calculation. (Record Type 5)

   b. A directory record containing information about the specific processes for which the Bondarenko factor data applies, such as the process, the energy groups for which data are given, etc. (Record Type 6)

   c. A record containing infinite dilution values for a process (Record Type 7)

   d. A record containing the Bondarenko factors for a process (Record Type 8)

6. A record containing average cross sections by process (Record Type 9)

7. Three record types are used to present transfer matrices:

   a. A directory record which specifies the processes, orders of anisotrophy, lengths, units, etc. (Record Type 10)

   b. A record to specify temperatures when the matrices are temperature dependent. (Record Type 11)

   c. A "magic-word" record to store a transfer matrix. (Record Type 12)

The structure of the various record types is now discussed.

Record Type 1 (Header Record)

The header record is the first record on a master and a working library and always contains 110 words:

1. IDTAPE - An identification number for the library.

1. Header information - written on the front of the library to specify the number of neutron and/or gamma groups, the number of nuclides, etc., contained in the library. (Record Type 1)

2. Energy structure information - contains the group boundaries. (Record Type 2)

3. Nuclide directory information - 50 words which give a title for the nuclide along with other parameters which specify the kinds of information included for the nuclide, such as number of records in the library for the nuclide and how much neutron and gamma data are given. (Record Type 3)

4. Resonance parameter data - provisions for resolved resonance parameters in the single level Breit-Wigner (SLBW) or multi-level Breit Wigner (MLBW) formats. Average parameters for an unresolved calculation may also be included. (Record Type 4)

5. Bondarenko data - four record types are used for this information:

   a. A record which gives the values of σ_0 and T at which the factors are tabulated, along with cutoff energies for the Bondarenko calculation. (Record Type 5)

   b. A directory record containing information about the specific processes for which the Bondarenko factor data applies, such as the process, the energy groups for which data are given, etc. (Record Type 6)

   c. A record containing infinite dilution values for a process (Record Type 7)

   d. A record containing the Bondarenko factors for a process (Record Type 8)

6. A record containing average cross sections by process (Record Type 9)

7. Three record types are used to present transfer matrices:

   a. A directory record which specifies the processes, orders of anisotrophy, lengths, units, etc. (Record Type 10)

   b. A record to specify temperatures when the matrices are temperature dependent. (Record Type 11)

   c. A "magic-word" record to store a transfer matrix. (Record Type 12)

The structure of the various record types is now discussed.

Record Type 1 (Header Record)

The header record is the first record on a master and a working library and always contains 110 words:

1. IDTAPE - An identification number for the library.

2. NNUC - The number of sets of data on the library.

3. IGM - The number of neutron energy groups on the library.

4. IFTG - The first thermal neutron group on the library, i.e., the first group which receives an upscatter source.

5. MSN - Zero.

6. IPM - The number of gamma-ray energy groups on the library.

7. I1 - Zero.

8. I2 - (0/1 = no/yes) A trigger which specifies that this library was produced by weighting a working library in the XSDRNPM module.

9. I3 - Zero.

10. I4 - Zero.

11-110. (TITLE(I),I = 1,100) - 100 words of text describing the cross-section library.

Record Type 2 (Energy Boundaries)

This record is on both a master and a working library and specifies the energy boundaries in eV of the neutron groups and/or gamma groups followed by the corresponding lethargy boundaries. The energy boundaries are arranged in ascending order followed by the lethargy boundaries in descending order. The "lethargy zero" is taken at 10 MeV. The structure is:

(EB(I),I=1,IGP),(UB(I),I=1,IGP)

where IGP is the number of groups plus one.

Record Type 3 (Cross-Section Set Directory Record)

Each set of data on a master or working library has a 50-word directory record which specifies certain parameters needed to determine dimensions required to process the data and to describe the make-up of the set of data. The following table describes this data:

Word(s)	Master library	Working library
1-18	18 words of text describing the set	18 words of text describing the set
19	Identifier of the set	Identifier of the set
20	Number of 6-parameter sets of resolved resonance data	Identifier of the set from which this set derived
21	Number of energies at which to evaluate unresolved values	Zone number in which the nuclide occurred
22	Number of 1-D neutron processes (temperature independent)	Number of zones in problem which produced this set
23	Number of 2-D neutron processes	Length of P_o total scattering matrix
24	Zero	Order of expansion of total scattering matrix
25	Number of 1-D gamma processes	Sequence of this set in all zone-weighted sets
26	Number of 2-D gamma processes	Number of zone-weighted sets for this nuclide
27	Number of 2-D neutron-to-gamma processes	Maximum length of any P_o array in the total transfer matrix
28	(Maximum order of scattering) *32768 + (total number of separate 2-D arrays for this set)	Number of 1-D neutron processes
29	A - neutron equivalent mass number	A - neutron equivalent mass number
30	ZA - 1000*Z + A	ZA - 1000*Z + A
31	Zero	Zero
32	Zero	Zero
33	Zero	Zero
34	Power per fission in watt-sec/fission	Power per fission in watt-sec/fission
35	Energy release per capture in watt-sec/capture	Energy release per capture in watt-sec/capture
36	Maximum length of any 2-D record in the set	Zero
37	Number of sets of Bondarenko data	Zero
38	Number of σ_o's in Bondarenko data	Zero
39	Number of T's in Bondarenko data	Zero
40	Maximum number of groups in Bondarenko data	Zero
41	Zero	Number of 1-D gamma processes
42	Zero	Zero

Word(s)	Master library	Working library
43	σ_p - potential scattering cross section	Zero
44	Zero	Zero
45	ENDF MAT for fast neutron data	ENDF MAT for fast neutron data
46	ENDF MAT for thermal neutron data	ENDF MAT for thermal neutron data
47	ENDF MAT for gamma data	ENDF MAT for gamma data
48	ENDF MAT for gamma production data	ENDF MAT for gamma production data
49	Nuclide symbol (Text)	Nuclide symbol (Text)
50	Number of records in this set	Number of records in this set

Note that the 50-word records are made up of integer, character, and floating point words. Words 1 to 18 and 49 are character data. Words 29, 30, 34, 35, and 43 are floating point. All other words are integers. For both types of libraries, many parameters may have no meaningful interpretation for a particular set of data. This is especially true of the working library; for example, words 20, 21, 22 25, and 26 only have meaning if the working library has been produced by weighting a previous working library. Zero values will be used when a parameter is not applicable.

Record Type 4 (Resonance Parameters)

This record is present only when either the number of six-word resonance sets, NRES, or the number of unresolved energy points, NUNR, is nonzero. Its length is

$$9 + 6*NRES + NUNR \ .$$

All values are in floating point.

The first nine words contain parameters used in both the resolved and unresolved resonance calculations:

1. A, the mass ratio for the isotope or mixture of isotopes,
2. σ_{PO}, the potential scattering cross section,
3. g, the average statistical factor in the unresolved region,
4. NRES, the number of six-parameter resonance sets,
5. S, a factor used in the Nordheim calculation to determine the range over which the calculation will be made,
6. $<D>$, the average level energy spacing for the $\ell=0$ unresolved sequence which will be calculated,
7. $<\Gamma_n^o>$, the average unresolved neutron width,
8. $<\Gamma_\gamma>$, the average unresolved gamma width,
9. $<\Gamma_f>$, the average unresolved fission width.

The next 6*NRES words consist of six-word sets of data used in the Nordheim calculation.

1. Zero
2. NBLK, the number of blocks of resolved data
3. Zero
4. Zero
5. TREF, the reference temperature at which the infinite dilution arrays on this library were calculated.
6. Zero

Folllowing these six words are NBLK six-word groups which are used to specify information concerning blocks of resonance data that apply to, for example, different isotopes, different energy regions, different kinds of data (s-wave or p-wave resonances), etc. These are stacked as follows:

1. AWRI, the mass ratio associated with the block of data,
2. ABUN, the abundance (atom fraction of isotope in natural element) for the block of data,
3. NRE, the number of resonances in the block,
4. ℓ, the value of spin for the resonances in the block,
5. EL, the low-energy cut-off for resonances in the block,
6. EH, the upper energy cut-off for resonances in the block,

.
.
.

After these NBLK groups of six words comes the six-word sets of resonance parameters arranged as follows:

1. E_0, the resonance energy,
2. Γ_n, the neutron width of the resonance,
3. Γ_γ, the gamma width of the resonance,
4. Γ_f, the fission width of the resonance,
5. r, a factor used in the Nordheim treatment for determing the range of calculation,
6. g, the statistical factor for the resonance.

The following chart illustrates the structure just described schematically:

1	2	3	4	5	6
0	NBLK	0	0	T_{ref}	0
$AWRI_1$	$ABUN_1$	NRE_1	ℓ_1	EL_1	EH_1
$AWRI_2$	$ABUN_2$	NRE_2	ℓ_2	EL_2	EH_2
.					
.					
.					
$AWRI_{NBLK}$	$ABUN_{NBLK}$	NRE_{NBLK}	ℓ_{NBLK}	EL_{NBLK}	EH_{NBLK}
E_o	Γ_n	Γ_γ	Γ_f	r	g
.	.	.	.	.	.
.	.	.	.	.	..
.	.	.	.	.	.

After the resolved resonance parameters, the points at which the unresolved cross section should be evaluated are specified. There are NUNR of these points arranged low-to-high in energy:

(EUNR(I), I=1, NUNR)

Record Type 5 (First Record of Bondarenko Block)

This record is used to specify the σ_0 and temperature values at which all Bondarenko factors for the nuclide will be presented. It also specifies the upper and lower energies for which factors can apply in the case where they do not span all energy groups. The number of σ_0 values, NSIGO, is specified in the 38th word in the set directory, and the 39th word specifies the number of temperatures, NT. The record structure is:

(σ_0(I), I=1, NSIGO), (T(I), I=1, NT), ELO, EHI

The σ_0 values can either ascend or descend, while the temperatures are in Kelvin in ascending order.

Record Type 6 (Directory for Bondarenko Data)

This record type is used to specify the proceses which have Bondarenko data in the set. Its length is six times NBOND, the number of Bondarenko processes, specifed in the 37th word in the set directory. The structure is:

(MT(I), I=1, NBOND),

(NF(I), I=1, NBOND),
(NL(I), I=1, NBOND),
(ORDER(I), I=1, NBOND)
(IOFF(I), I=1, NBOND),
(NZ(I), I=1, NBOND)

The parameters have the following interpretation: MT is the identifier of the process; e.g., MT=2 is for elastic scattering as in ENDF/B. NF is the number of the first energy group for which parameters are given. NL is the last group for which parameters are given. ORDER is used to specify the order of the scattering matrix in the event it is to be self-shielded. IOFF is the offset from the "magic-word" in the transfer matrices; e.g., for elastic scattering, the first word after the magic-word is generally the within-group term such that IOFF=1 would specify the shielding for the within-group terms. (This parameter will be clearer in the discussion of AMPX transfer matrices given below.) NZ is presently unused and has a zero value.

Record Type 7 (Infinite Dilution Values for Bondarenko Data)

Each process which has Bondarenko data has one of these records which contains the infinite dilution values for the process. Its structure is

$$(\sigma^{\infty}(I), I=NF, NL)$$

where NF and NL are the first and last groups with data for the process.

Record Type 8 (Bondarenko Factors)

This record is a three-dimensional array and contains the Bondarenko factors for a process. Its structure is

$$(((BF(I,J,K),I=1, NSIGO), J=1, NT), K=NF, NL)$$

Record Type 9 (Temperature-Independent Average Cross Sections)

This record type is used to present average cross sections (sometimes called one-dimensional cross sections) on the library.

Its structure is

MT_1, $(\sigma_1(I), I=1, IGM)$
MT_2, $(\sigma_2(I), I=1, IGM)$
.
.
.
MT_{LAST}, $(\sigma_{LAST}(I), I=1, IGM)$

where the MT's are the process identifiers, and the cross sections, σ, are given for all groups. (Note that the MT's are given as floating point numbers.)

Record Type 10 (Scattering Matrix Directory)

An AMPX master library always provides a directory which identifies the scattering matrices that are given for a nuclide. The structure is

(MT(I), I=1, N2D),
(L(I), I=1, N2D),
(NL(I), I=1, N2D),
(NT(I), I=1, N2D),

where N2D is the number of scattering (two-dimensional) processes, MT is the process identifier, L is the maximum length of any of the scattering matrices for the process, NL is the order of Legendre fit to the scattering matrix, and NT is a parameter whose definition depends on the type of data (whether neutron, gamma production, or gamma) given as follows:

a. For neutron-neutron data, NT is the number of temperatures at which scattering matrices are given.
b. For gamma production data, NT is zero if the data are in yield units and is unity if they are in cross-section units.
c. For gamma-gamma data, NT is zero.

Record Type 11 (Scattering Matrix Temperatures)

This record type is only used on a master library and specifies the temperatures (in eV) of the scattering matrices. It is only used for neutron-neutron data and is given when NT>0 (see Record Type 10). The temperatures are in ascending order as follows:

(T(I), I=1, NT)

Record Type 12 (Scattering Matrix)

This record type is used to store scattering matrix data (sometimes called 2-D data). As will be illustrated, it has provisions for truncating zero and/or impossible elements from the array. It exists in two forms: (1) a self-defining form used for gamma production data on a master library and for all scattering matrices on a working library, and (2) a form that is not self-defining. The only difference is that the self-defining form specifies the length as the first word, while the other does not; i.e.,

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

or

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

The structure of the X-array is as follows:

Magic word for a group,
Terms for scattering to the group,
Magic word for the next group,
Terms for scattering to this group,
Etc., etc.

In some cases, a negative or zero magic-word is used to specify the end of data in the record.

A "magic-word" is used to define:

a. the sink group number, III,
b. the first group number, JJJ, which scatters to this group,
c. the last group number, KKK, which scatters to this group.

The magic-word is then defined as

$$MW = 1000000*JJJ + 1000*KKK + III$$

such that it is composed of three 3-digit integers:

MW : JJJKKKIII

The scattering terms below a "magic-word" are in reverse ordering (following typical practice for transport theory programs); i.e., the scattering term for scattering from the last group is first, etc,:

MW for group III
σ(KKK→III)
σ(KKK-1→III)
.
.
.
σ(JJJ→III)

The scattering matrix record will contain one P_ℓ matrix for a process.

Consider an elastic scattering matrix for hydrogen which will be a full triangular matrix and assume three energy groups. The scattering matrix will look as follows:

1001001
σ(1→1)
1002002
σ(2→2)
σ(1→2)
1003003
σ(3→3)
σ(2→3)
σ(1→3)

Note that the record is a mixture of integer and floating point terms.

AMPX Master Library Format

The overall structure of an AMPX master library is given below:

	Record Type
Header Record	1
Nuclide Directory (one record per nuclide)	3
Neutron Energy Boundaries	2
Gamma Energy Boundaries	2
Records for Nuclide 1	
Records for Nuclide 2	
Etc.	

The structure of the records of a nuclide is:

	Record Type
Nuclide Directory Record	3
Bondarenko Data	
Resonance Parameter Data	4
Temperature-Independent, Group-Averaged Neutron Cross Sections	9
Scattering Matrix Data for Neutrons	
Scattering Matrix Data for Gamma Production	
Group-Averaged Gamma Cross Sections	9
Scattering Matrix Data for Gammas	

The internal structure for Bondarenko data is:

	Record Type
(σ_0(I),I=1,NSIGO),(T(J),J=1,NT),EL,EH	5
Bondarenko Data Directory (MT(I),I=1,NB), (NF(I),I=1,NB), (NL(I),I=1,NB), (ORDER(I),I=1,NB), (IOFF(I),I=1,NB), (NZ(I),I=1,NB)	6
The following records are given in pairs for all NB Bondarenko processes.	
Infinite Dilution Values (σ^{∞}(I),I=NF,NL)	7
Bondarenko Factors (((BF(I,J,K),I=1,NSIGO),J=1,NT),K=NF,NL)	8

The internal structure of the scattering matrix data for neutrons is:

	Record Type
Scattering Matrix Directory (MT(I),I=1,N2D), (L(I),I=1,N2D), (NL(I),I=1,N2D), (NT(I),I=1,N2D)	10
The following structure is repeated N2D times.	
Temperature Values (NT>0) (T(I),I=1,NT)	11
The following record is repeated MAX(1,NT)*(NT+1) times.	
Scattering Matrix (X(I),I=1,L)	12

The internal structure for gamma production scattering is:

	Record Type
Scattering Matrix Directory (MT(I),I=1,N2D), (L(I),I=1,N2D), (NL(I),I=1,N2D), (NT(I),I=1,N2D)	10
For each process, I=1,N2D, the following record type is given NL+1 times corresponding to the P_0, P_1,...P_{NL} matrices.	
Self-Defining Scattering Matrix Length LENGTH<(X(I),I=1,LENGTH)	12

The internal structure for gamma-gamma scattering is:

	Record Type
Scattering Matrix Directory (MT(I),I=1,N2D), (L(I),I=1,N2D), (NL(I),I=1,N2D), (NT(I),I=1,N2D)	10
For each process, I=1,N2D, the following record type is given NL+1 times corresponding to the P_0, P_1,...P_{NL} matrices.	
Scattering Matrix (X(I),I=1,L)	12

AMPX Working Library Format

The overall structure of a working library is given below:

	Record Type
Header Record	1
Neutron Energy Boundaries (one record per nuclide)	2
Gamma Energy Boundaries	2
Nuclide Directory (one record per nuclide	3
Records for Nuclide 1	
Records for Nuclide 2	
Etc.	

The structure of the records for a nuclide is:

	Record Type
Nuclide Directory Record	3
Group-Averaged Neutron Cross Sections	9
Group-Averaged Gamma Cross Sections	9
The P_0, P_1 ..., P_{NL} total scattering matrices are presented in self-defining records.	
L,(X(I),I=1,L)	12

APPENDIX B
DEFINITIONS OF REACTION TYPES USED IN THE ANSL-V LIBRARIES

APPENDIX B

Reaction types in ENDF data are identified by integers — MT numbers. Within the AMPX system, the ENDF MT numbers are used, where possible, to identify the appropriate cross-section data. Where AMPX processed data are available that are not defined by an ENDF MT number, the data are identified by integers outside the range of the ENDF MT numbers. The following definition of reaction types is taken from Appendix B of the ENDF manual — being augmented where necessary to describe identifiers assigned to AMPX processed data, i.e., the ANSL-V master library data.

The MT number generally refers to a specific neutron-nucleus intreaction mechanism, but occasionally it indicates that a particular type of information is given. The general rules for assignment of MT numbers are as follows:

MT (range)	Description of Class of Reactions
1-100	Reaction types in which secondary particles of the same type as the incident particles are emitted
101-150	Reaction types in which no secondary particles of the same type as the incident particles are emitted
151-200	Resonance region information
201-450	Quantities derived from the basic data
451-699	Miscellaneous quantities
700-799	Excitation cross sections for reactions that emit charged particles
800-999	(not assigned)
1000-4000	AMPX special identifiers

Within AMPX, different modules can assign the same MT number to a more or less inclusive set of data pertaining to the implied reaction. Specific MT assignments are given in the following table. Situations where different AMPX modules use the same MT number to identify the more or less inclusive data are identified below.

MT	Description
1	Total cross section (redundant, equal to the sum of all partial cross sections)
2	Elastic scattering cross section
3	Nonelastic cross section (redundant, equal to the sum of all partial cross sections except elastic scattering)
4	Total inelastic cross section (redundant, equal to the sum of MT = 51, 52, 53,..., 90, 91)
5	(to be assigned)
6	(n, 2n) cross section for first excited state (describes first neutron)
7	(n, 2n) cross section for second excited state (describes first neutron)
8	(n, 2n) cross section for third excited state (describes first neutron)
9	(n, 2n) cross section for fourth excited state (describes first neutron)
10-15	(to be assigned)
16	direct (n, 2n) cross section [total (n, 2n) cross section is sum of MT = 6, 7, 8, 9, and 16]
17	(n, 3n) cross section
18	Total fission cross section (sum of MT = 19, 20, 21, 38).
19	(n,f) cross section (first chance fission)
20	(n,n'f) cross section (second chance fission)
21	(n,n'f) cross sections (third chance fission)
22	(n, n'α) cross section
23	(n, n'3α) cross section
24	(n, 2nα) cross section
25	(n, 3nα) cross section
26	(n, 2n) isomeric state cross section
27	Absorption cross section (sum of MT = 18 and 101) includes particle reactions
28	(n, n'p) cross section
29	(n, n'2α) cross section
30	(n, 2n2α) cross section
31	Used only in ENDF/B data as an LR flag to indiate that γ-emission is the mode of decay of the residual nucleus formed in the primary reaction[a]
32	(n,n'd) cross section
33	(n, n't) cross section
34	(n, n'^{3}He)
35	(n, n'd2α) cross section
36	(n, n't2α) cross section
37	(n, 4n) cross section
38	(n, 3nf) cross section (fourth chance fission)

[a]The "primary" reaction could be, for example, an (n,n'), (n,p), (n,α), (n,n'p), etc , reaction.

MT	Description
39	Used only in ENDF/B data to indicate that internal conversion is the mode of decay of the residual nucleus formed in the primary reaction[a]
40	Used only in ENDF/B to indicate that electron-position pair formation is the mode of decay of the residual nucleus formed in the primary reaction[a]
41-45	(to be assigned)
46	cross section for describing the second neutron from (n, 2n) reaction for first excited state
47	cross section for describing the second neutron from (n, 2n) reaction for second excited state
48	cross section for describing the second neutron from (n, 2n) reaction for third excited state
49	cross section for describing the second neutron from (n, 2n) reaction for fourth excited state (Note: MT = 46, 47, 48, and 49 should not be included in the sum for the total (n, 2n) cross section.)
50	(to be assigned)
51	(n,n′) to the first excited state
52	(n,n′) to the second excited state
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90	(n,n′) to the 40th excited state
91	(n,n′) to the continuum
92-100	(to be assigned)
101	neutron disappearance (sum of all cross sections in which a neutron is not in the exit channel), i.e., $$MT = 101 \text{ is } \sum_{i=2}^{14} (MT = 100 + i)$$
102	(n,γ) radiative capture cross section
103	(n,p) cross section
104	(n,d) cross section
105	(n,t) cross section
106	(n, ^{3}He) cross section
107	(n,α) cross section
108	(n,2α) cross section
109	(n,3α) cross section
110	(to be assigned)

[a]The "primary" reaction could be, for example, an (n,n'), (n,p), (n,α), (n,n'p), etc., reaction.

MT	Description
111	(n,2p) cross section
112	(n,pα) cross section
113	(n,t2α) cross section
114	(n,d2α) cross section
115-119	(to be assigned)
120	Target destruction = nonelastic less total (n, n'γ)
121-150	(to be assigned)
151	General designation for resonance information
152-200	(to be assigned for specific resonance information)
201-202	(to be assigned)
203	Total hydrogen production
204	Total deuterium production
205	Total tritium production
206	Total ^{3}He production
207	Total ^{4}He production
208-250	(to be assigned)
251	$\bar{\mu}$L; the average cosine of the scattering angle (laboratory system) for elastic scattering
252	ξ, the average logarithmic energy decrement for elastic scattering
253	γ, the average of the square of the logarithmic energy decrement for elastic scattering, divided by twice the average logarithmic decrement for elastic scattering
254-300	(to be assigned)
301-450	Energy release rate parameters, $\overline{E * \sigma}$ for total and partial cross section. Subtract 300 from this number to obtain the specific reaction type identification. For example, MT = 302 = (300 + 2) denotes elastic scattering
451	Heading or title information (given only in File 1)
452	$\bar{\upsilon}$, average total (prompt plus delayed) number of neutrons released per fission event)
453	Radioactive nuclide production
454	Fission product yield data
455	Delayed neutrons from fission
456	Prompt neutrons from fission
457	Radioactive decay data
458	Energy release in fission
459-500	(to be assigned)
501	Total photon interaction cross section
502	Photon incoherent scattering
503	(to be assigned)
504	Photon incoherent scattering
505-514	(to be assigned)
515	Pair production, electron field
516	Pair production, nuclear and electron field (i.e., pair plus triplet production)
517	Pair production, nuclear field

MT	Description
518	Photofission (γ, f)
519-526	(to be assigned)
527	Sum of all gamma-ray absorption processes
528-531	(to be assigned)
532	Photoneutron (γ, n)
533	Total photonuclear
534-601	(to be assigned)
602	Photoelectric
603-699	(to be assigned)
700	(n, p_0) cross section (cross section for leaving the residual nucleus in the ground state)
701	(n, p_1) cross section for 1st excited state
702	(n, p_2) cross section for 2nd excited state
703	(n, p_3) cross section for 3rd excited state
704	(n, p_4) cross section for 4th excited state
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718	(n, p_c) cross section for continuum excited state
719	(n, pc′) cross section for continuum specifically not included in total (redundant, used for describing outgoing proton)
720	(n, d_0) cross section for ground state
721	(n, d_1) cross section for 1st excited state
722	(n, d_2) cross section for 2nd excited state
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738	(n, d_c) cross section for continuum excited state
739	(n, dc′) cross section for continuum specificaly not included in σ total (redundant, used for describing outgoing deuteron)
740	(n, t_0) cross section for ground state
741	(n, t_1) cross section for 1st excited state
742	(n, t_2) cross section for 2nd excited state
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750	(n, t_c) cross section for continuum excited state
759	(n, t_c) cross section for continuum specifically not included in σ total (redundant, used for describing outgoing triton)
760	(n, $^3He_0$) cross section for ground state
761	(n, $^3He_1$) cross section for 1st excited state
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778	(n, ^3He_c) cross section for continuum

MT	Description
779	(n, $^{3}He_c$) cross section for continuum specifically not included in σ total (redundant, used for describing outgoing ^{3}He)
780	(n, α_0) cross section for ground state
781	(n, α_1) cross section for 1st excited state
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798	(n, α_c) cross section for continuum
799	(n, α_c) cross section for continuum specifically not included in σ_T (redundant, used to describe outgoing α)
800-999	(to be assigned)
1000[a]	XSDRNPM-produced transport cross section based on the outscatter approximation, i.e., $\sigma_{tr} = \sigma_a + (1 - \bar{\mu})\ \sigma_s$ (Note: σ_a is MT 27; σ_s is NT 2.)
1001[a]	XSDRNPM-produced transport cross section based on the inscatter approximation, i.e., $$\sigma_{tr} = \sigma_t - \frac{\int_0^\infty \sigma_1(E' \rightarrow E)J(E')dE'}{3J(E)}$$
1007	Thermal scattering matrix where upscatter is present - can be coherent and incoherent data
1018	Chi (χ) distribution of neutrons from the fission reaction. Chi is represented in ENDF File 5 data as a function of neutron energy. XLACS-2 calculates energy dependent χ's by group and weights the χ's over the fission rate ($\nu\Sigma_f\phi$) to produce a single vector.
1099	Group integral of the weight function used to generate this set of cross sections.
1452	Product of nu times the fission cross section.
1500-1501	Same as 1000-1001 except for gamma-ray cross sections.
1527	Gamma-ray energy absorption coefficients. These data are also called kerma factors.
3000+MT	These arrays contain the averaged cross section for the process MT weighted over a "smooth" weighting function taken to be 1/E above 5kT and Maxwellian below that; e.g., 3002 is the average elastic scattering value. Values for "3000" arrays are generally infinite dilution values. For MT > 1000, ids are 3000 + (MT-1000)

[a]Transport cross sections are current weighted in XSDRNPM. All other cross sections are flux weighted.

APPENDIX C
EVALUATORS OF ENDF/B-V EVALUATIONS USED IN ANSL-V

APPENDIX C

Evaluators of ENDF/B-V General Purpose Evaluations

	MAT No	Laboratory	Author
1-H-1	1301	LANL	L. Stewart, R. J. LaBauve, P. G. Young
1-H-2	1302	LANL	L. Stewart (LANL), A. Horsely (AWRE)
1-H-3	1169	LANL	Leona Stewart
2-He-3	1146	LANL	Leona Stewart (LANL)
2-He-4	1270	LANL	Nisley, Hale, Young (LANL)
3-Li-6	1303	LANL	G. Hale, L. Stewart, P. G. Young
3-Li-7	1272	LANL	R. J. LaBauve, L. Stewart, M. Battat
4-Be-9	1304	LLL	Howerton, Perkins
5-B-10	1305	LANL	G. Hale, L. Stewart, P. Young
5-B-11	1160	GE-BNL	C. Cowan
6-C	1306	ORNL	C. Y. Fu and F. G. Percy
7-N-14	1275	LANL	P. Young, D. Foster, Jr., G. Hale
7-N-15	1307	LANL	E. Authur, P. Young, G. Hale
8-O-16	1276	LANL	P. Young, D. Foster, Jr., G. Hale
8-O-17	1317	BNL	B. A. Magurno
9-F-19	1309	ORNL	C. Y. Fu, D. C. Larson, F. G. Percy
11-Na-23	1311	ORNL	D. C. Larson
12-Mg	1312	ORNL	P. G. Young, D. G. Foster, Jr.
13-Al-27	1313	LANL	P. G. Young, D. G. Foster, Jr.
14-Si	1314	ORNL	Larson, Percy, Drake, Young
15-P 31	1315	LLL	Howerton
16-S	1347	BNL	Divadeenam
16-S-32	1316	LLL	Howerton
17-Cl	1149	GGA	M. S. Allen and M. K. Drake
19-K	1150	GGA	M. K. Drake
20-Ca	1320	ORNL	C. Y. Fu and F. G. Percy
22-Ti	1322	BUR, ANL, LLL	C. Philis, A. Smith, R. Howerton
23-V	1323	ANL, LLL, HEDL	S. Smith, H. Howerton, F. Mann
24-Cr	1324	BNL	A. Prince and T. W. Burrows
25-Mn-55	1325	BNL	S. F. Mughabghab
26-Fe	1326	ORNL	C. Y. Fu and F. G. Percy
27-Co-59	1327	BNL	S. Mughabghab
28-Ni	1328	BNL	M. Divadeenam
29-Cu	1329	ORNL, SAI	Fu, Drake, Fricke
36-Kr-78	1330	BNL	A. Prince
36-Kr-80	1331	BNL	A. Prince
36-Kr-82	1332	BNL	A. Prince
36-Kr-83	1333	BNL	A. Prince

Evaluators of ENDF/B-V General Purpose Evaluations

	MAT No.	Laboratory	Author
36-Kr-84	1334	BNL	A. Prince
36-Kr-86	1336	BNL	A. Prince
40-Zr	1340	SAI	M. Drake, D. Sargis, T. Maung
40-Zr-90	1385	SAI	M. Drake, D. Sargis, T. Maung
40-Zr-91	1386	SAI	M. Drake, D. Sargis, T. Maung
40-Zr-92	1387	SAI	M. Drake, D. Sargis, T. Maung
40-Zr-94	1388	SAI	M. Drake, D. Sargis, T. Maung
40-Zr-96	1389	SAI	M. Drake, D. Sargis, T. Maung
41-Nb-93	1189	ANL, LLL	R. Howerton (LLL) and A. Smith
42-Mo	1321	LLL, HEDL	Howerton, Schmittroth, Schenter
43-Tc-99	1308	HEDL, BAW	Schenter, Livolsi, Schmittroth, et al.
45-Rh-103	1310	HEDL, BAW	Schenter, Livolsi, Schmittroth, et al.
47-Ag-107	1371	HEDL, BNL	Schenter, Bhat, Prince, Johnson, et al.
47-Ag-109	1373	HEDL, BNL	Schenter, Bhat, Prince, Johnson, et al.
48-Cd	1281	BNL	S. Pearlstein (Translated for U. K.)
48-Cd-113	1318	BNL, HEDL	Pearlstein, Mann, Schenter
54-Xe-124	1335	BNL	M. R. Bhat and S. F. Mughabghab
54-Xe-126	1339	BNL	M. R. Bhat and S. F. Mughabghab
54-Xe-128	1348	BNL	M. R. Bhat and S. F. Mughabghab
54-Xe-129	1349	BNL	M. R. Bhat and S. F. Mughabghab
54-Xe-130	1350	BNL	M. R. Bhat and S. F. Mughabghab
54-Xe-131	1351	BNL	M. R. Bhat and S. F. Mughabghab
54-Xe-132	1352	BNL	M. R. Bhat and S. F. Mughabghab
54-Xe-134	1354	BNL	M. R. Bhat and S. F. Mughabghab
54-Xe-135	1294	BNW	B. R. Leonard, Jr. and K. B. Stewart
54-Xe-136	1356	BNL	M. R. Bhat and S. F. Mughabghab
55-Cs-133	1355	HEDL, BNL	Schenter, Bhat, Prince, Johnson, et al.
56-Ba-138	1353	LLL	Howerton
62-Sm-149	1319	HEDL, BNW	Schenter, Leonard, Stewart, et al.
63-Eu-151	1357	BNL	S. F. Mughabghab
63-Eu-152	1292	BNL	H. Takahashi
63-Eu-153	1359	BNL	S. Mughabghab
63-Eu-154	1293	BNL	H. Takahashi
64-Gd-152	1362	BNL	B. A. Magurno
64-Gd-154	1364	BNL	B. A. Magurno
64-Gd-155	1365	BNL	B. A. Magurno
64-Gd-156	1366	BNL	B. A. Magurno
64-Gd-157	1367	BNL	B. A. Magurno
64-Gd-158	1368	BNL	B. A. Magurno
64-Gd-160	1370	BNL	B. A. Magurno

Evaluators of ENDF/B-V General Purpose Evaluations

	MAT No.	Laboratory	Author
66-Dy-164	1031	BNW	B. R. Leonard, Jr. and K. B. Stewart
71-Lu-175	1032	BNW	B. R. Leonard, Jr. and K. B. Stewart
71-Lu-176	1033	BNW	B. R. Leonard, Jr. and K. B. Stewart
72-Hf	1372	SAI	M. Drake, D. Sargis, T. Maung
72-Hf-174	1374	SAI	M. Drake, D. Sargis, T. Maung
72-Hf-176	1376	SAI	M. Drake, D. Sargis, T. Maung
72-Hf-177	1377	SAI	M. Drake, D. Sargis, T. Maung
72-Hf-178	1378	SAI	M. Drake, D. Sargis, T. Maung
72-Hf-179	1383	SAI	M. Drake, D. Sargis, T. Maung
72-Hf-180	1384	SAI	M. Drake, D. Sargis, T. Maung
73-Ta-181	1285	LLL	Howerton, Perkins, MacGregor
73-Ta-182	1127	AI	J. Otter, C. Dunford, and E. Ottewiwtte
74-W-182	1128	AI, LANL	Otter, Ottewitte, Rose, Young
74-W-183	1129	AI, LANL	Otter, Ottewitte, Rose, Young
74-W-184	1130	AI, LANL	Otter, Ottewitte, Rose, Young
74-W-186	1131	AI, LANL	Otter, Ottewitte, Rose, Young
75-Re-185	1083	GE(NMPO)	W. B. Henderson and J. W. Zwick
75-Re-187	1084	GE(NMPO)	W. B. Henderson and J. W. Zwick
79-Au-197	1379	BNL	S. F. Mughabghab
82-Pb	1382	ORNL	C. Y. Fu and F. G. Percy
90-Th-232	1390	BNL	Bhat, Smith, Leonard, DeSaussure, et al.
91-Pa-233	1391	HEDL, INEL	Mann, Schenter, Reich
92-U-233	1393	LANL, ORNL	Stewart et al., Weston, Mann
92-U-234	1394	BNL, HEDL	Divadeenam, Mann, Drake, Reich, et al.
92-U-235	1395	BNL,	M. R. Bhat
92-U-236	1396	BNL, HEDL	Divadeenam, Mann, McCrosson, Reich
92-U-238	1398	ANL	E. Pennington, A. Smith, W. Poenitz
93-Np-237	1337	HEDL, SRL	Mann, Benjamin, Smith, Stein, Reich
94-Pu-238	1338	HEDL, AI	Mann, Schenter, Alter, Dunford
94-Pu-239	1399	GE-FBRD	E. Kujawski, L. Stewart (LANL)
94-Pu-240	1380	ORNL	L. W. Weston
94-Pu-241	1381	ORNL	L. W. Weston, R. Q. Wright, Howerton
94-Pu-242	1342	HEDL, SRL	Mann, Benjamin, Madland, Howerton
95-Am-241	1361	HEDL, ORNL	Mann, Schenter, and Weston
95-Am-242m	1369	HEDL, SRL, LLL	Mann, Benjamin, Howerton, et al.
95-Am-243	1363	HEDL, SRL, LLL	Mann, Benjamin, Howerton, et al.
96-Cm-243	1343	HEDL, SRL, LLL	Mann, Benjamin, Howerton, et al.
95-Cm-244	1344	HEDL, SRL, LLL	Mann, Benjamin, Howerton, et al.
96-Cm-245	1345	SRL	Benjamin and McCrosson
96-Cm-246	1346	BNL, SRL, LLL	Kinsey, Benjamin, Howerton

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