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VITAMIN-E

A Coupled 174-Neutron, 38-Gamma-Ray Multigroup Cross-Section Library for Deriving Application-Dependent Working Libraries for Radiation Transport Calculations

Contributed by

Oak Ridge National Laboratory
Oak Ridge, Tennessee

RADIATION SHIELDING INFORMATION CENTER

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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," Draft ORNL/RSIC report (August 1984) 1

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RSIC DATA LIBRARY DLC-113

1. **NAME AND TITLE OF DATA LIBRARY**
   VITAMIN-E: 174n, 38g Cross-Section Library in AMPX Format.

2. **NAME AND TITLE OF DATA RETRIEVAL PROGRAM**
   No retrieval program is provided. The PSR-117/MARS or PSR-63/AMPX code packages are suggested for full implementation of the capabilities of VITAMIN-E. JCL and sample imput are provided to execute sample problems using MARS.

3. **CONTRIBUTOR**
   Oak Ridge National Laboratory, Oak Ridge, Tennessee.

4. **HISTORICAL BACKGROUND AND INFORMATION**
   The successful application of DLC-41/VITAMIN-C, based on ENDF/B-IV, to a variety of radiation transport problems led to the development of specifications for an ENDF/B-V based cross-section library, denoted VITAMIN-E.

5. **APPLICATION OF THE DATA**
   The earkly phases of this new cross-section library were focused on materials for fast reactor applications and were applied to benchmark testing of ENDF/B-V. More recently, requests have been made for additional materials to be added to the basic library for fusion applications. The library is expected to perform well for radiation transport problems where thermal upscatter is not important.

6. **SOURCE AND SCOPE OF DATA**
   The energy structure of VITAMIN-E contains 174 neutron and 38 gamma-ray groups and includes the 171 neutron and 36 photon groups of VITAMIN-C as a subset. The group structure has fine detail in the energy region where cross section minima occur for important shielding materials. The ENDF/B-V special purpose dosimetry, activation, and gas production files have also been processed into the VITAMIN-E group structure. The 174 neutron group data were processed with MINXI5; the 174 neutron, 38 photon group data were processed with LAPHNGAS (AMPX III); and the 38 gamma-ray group data with SMUG (AMPX III) from DLC-99/HUGO.

7. **DISCUSSION OF THE DATA RETRIEVAL PROGRAM**
   PSR-117/MARS can be used to translate the data into binary form and to perform a variety of useful manipulations including checking, editing, self shielding, and group collapsing.

8. **DATA FORMAT AND COMPUTER**
   EBCDIC card images; IBM 3033.

9. TYPICAL RUNNING TIME

The first part of the ZPR 6/7 infinite medium problem required 55,097 I/O's and 8.3 cpu minutes on the IBM 3033. This includes the 21,000 I/O's and 6.6 cpu minutes required to convert the formatted library to binary. The XSDRNPM run required only 37 cpu seconds and 6065 I/O's, and the GIP/ANISN run took 10,060 I/O's and 38 cpu seconds. The first part of the CTR standard blanket run required 58,066 I/O's and 10.15 cpu minutes. The XSDRNPM problem ran for 1.3 cpu minutes using 5147 I/O's while the GIP/ANISN run required 1.16 cpu minutes and 9160 I/O's.

10. REFERENCE

R. W. Roussin et al., "Vitamin-E: A Coupled 174 Neutron, 38 Gamma-Ray Multigroup Cross-Section Library for Deriving Application Dependent Working Libraries for Radiation Transport Calculations," ORNL-RSIC-xxxx (August 1984).

11. CONTENTS OF LIBRARY

The 174n group transport cross section data are separated into four files; the 174n, 38g group photon production data and the 38g group photon transport data are each included in separate files. Special purpose neutron dosimetry, activation, and gas production data files are also included. JCL is included to execute sample problems using MARS; total records, 1,248,816.

12. DATE OF ABSTRACT

July 1984; December 1984.

KEYWORDS: AMPX INTERFACE FORMAT; COUPLED NEUTRON-GAMMA-RAY CROSS SECTIONS; CTR NEUTRONICS CROSS SECTIONS; CTR PROCESSED CROSS-SECTION LIBRARY; GAMMA-RAY CROSS SECTIONS; GAMMA-RAY PRODUCTION DATA; MULTIGROUP CROSS SECTIONS; MULTIGROUP CROSS SECTIONS BASED ON ENDF/B; NEUTRON CROSS SECTIONS; REACTION CROSS SECTIONS

TAPE#		RSIC#	DLC113
DENSITY		CODE PKG NAME:	VITAMIN-E
		COMPUTER:	IBM 3033
		PACKAGED:	7/31/84

DESCRIPTION	LOGICAL RECORDS	DCB PARAMETERS
1. 174 group neutron cross sections (16 materials) These materials are used in the sample problems.	359,395	FB 12800 80

ID	Material	ID	Material	ID	Material
130301	Li-6	139701	Li-7	130601	C-12
127601	O-16	131101	Na-23	132401	Cr
132502	Mn-55	132603	Fe	132802	Ni
118901	Nb	132101	Mo	139501	U-235
139801	U-238	139901	Pu-239	138001	Pu-240
138101	Pu-241				

2. 174 group neutron cross sections (20 materials)	358,050	FB 12800 80

ID	Material	ID	Material	ID	Material
9301	H-1	130401	Be-9	104	Be-9
130501	B-10	8811	B-11	127501	N-14
130902	F-19	131201	Mg	131301	Al-27
131401	Si	131501	P-31	134701	S
114901	Cl	8824	Ar	115001	K
132003	Ca	132201	Ti	132301	V
132703	Co-59	132901	Cu		

3. 174 group neutron cross sections (20 materials)	250,076	FB 12800 80

ID	Material	ID	Material	ID	Material
135801	Ga	8841	Zr	8847	Cd
8850	Sn	8853	Gd	128502	Ta-181
182	W-182	183	W-183	184	W-184
186	W-186	8861	Au-197	138202	Pb
139001	Th-232	134201	Pu-242	136101	Am-241
136901	Am-242m	136301	Am-243	864201	Cm-242
134301	Cm-243	134401	Cm-244		

4. 174 group neutron cross sections (21 materials)	124,216	FB 12800 80

ID	Material	ID	Material	ID	Material
116901	H-3	127000	He-4	116000	B-11
920201	Y-89	134002	Zr	966901	Cs-137
137401	Hf-174	137601	Hf-176	137701	Hf-177
137801	Hf-178	138301	Hf-179	138401	Hf-180
108301	Re-185	108401	Re-187	139101	Pa-233
139301	U-233	9394	U-234	139601	U-236
133701	Np-237	133801	Pu-238	854201	Am-242

5. 174 neutron, 38 photon group cross sections (56 materials) 49,145 FB 12800 80

ID	Material	ID	Material	ID	Material
130101	H-1	130301	Li-6	139701	Li-7
130401	Be-9	104	Be-9	130501	B-10
8811	B-11	130601	C-12	127501	N-14
127601	O-16	130902	F-19	131103	Na-23
131201	Mg	131301	Al-27	131403	Si
131501	P	134701	S	114901	Cl
8824	Ar	115001	K	132003	Ca
132201	Ti	132301	V	132402	Cr
132502	Mn-55	132603	Fe	132703	Co-59
132801	Ni	132901	Cu	135801	Ga
8841	Zr	118901	Nb-93	132101	Mo
8847	Cd	8850	Sn	8853	Gd
128501	Ta-181	182	W-182	183	W-183
184	W-184	186	W-186	8861	Au-197
138202	Pb	139001	Th-232	139501	U-235
139801	U-238	139901	Pu-239	138001	Pu-240
138101	Pu-241	134201	Pu-242	136101	Am-241
136901	Am-242m	136301	Am-243	864201	Cm-242
134301	Cm-243	134401	Cm-244		

6. 38 group photon interaction library 76,284 FB 12800 80

ID	Material	ID	Material	ID	Material	ID	Material
1	H	2	He	3	Li	4	Be
5	B	6	C	7	N	8	O
9	F	10	Ne	11	Na	12	Mg
13	Al	14	Si	15	P	16	S
17	Cl	18	Ar	19	K	20	Ca
21	Sc	22	Ti	23	V	24	Cr
25	Mn	26	Fe	27	Co	28	Ni
29	Cu	30	Zn	31	Ga	32	Ge
33	As	34	Se	35	Br	36	Kr
37	Rb	38	Sr	39	Y	40	Zr
41	Nb	42	Mo	43	Tc	44	Ru
45	Rh	46	Pd	47	Ag	48	Cd
49	In	50	Sn	51	Sb	52	Te
53	I	54	Xe	55	Cs	56	Ba
57	La	58	Ce	59	Pr	60	Nd
61	Pm	62	Sm	63	Eu	64	Gd
65	Tb	66	Dy	67	Ho	68	Er
69	Tm	70	Yb	71	Lu	72	Hf
73	Ta	74	W	75	Re	76	Os
77	Ir	78	Pt	79	Au	80	Hg
81	Tl	82	Pb	83	Bi	84	Po
85	At	86	Rn	87	Fr	88	Ra
89	Ac	90	Th	91	Pa	92	U
93	Np	94	Pu	95	Am	96	Cm
97	Bk	98	Cf	99	Es	100	Fm

7. 174 group neutron special purpose dosimetry 3,757 FB 12800 80
(26 materials) *

ID	Material	ID	Material	ID	Material
6311	Na-23	6313	Al-27	6325	Mn-55
6327	Co-59	6337	Np-237	6379	Au-197
6390	Th-232	6395	U-234	6398	U-238
6399	Pu-239	6424	Li-64	6425	B-10
6426	Sc-45	6427	Ti-46	6428	Ti-47
6429	Ti-48	6430	Fe-54	6431	Fe-56
6432	Fe-58	6433	Ni-58	6434	Ni-60
6435	Cu-63	6436	Cu-65	6437	In-115
6438	I-127	6439	S-32		

8. 174 group neutron special purpose activation 7,501 FB 12800 80
(53 materials) *

ID	Material	ID	Material	ID	Material
7036	Li-6	7037	Li-7	7050	B-10
7074	N-14	7086	O-16	7099	F-19
7113	Na-23	7124	Mg-24	7137	Al-27
7151	P-31	7162	S-32	7180	Ar-40
7191	K-41	7215	Sc-45	7220	Ti-50
7226	Ti-46	7227	Ti-47	7228	Ti-48
7240	Cr-50	7242	Cr-52	7255	Mn-55
7264	Fe-54	7266	Fe-56	7268	Fe-58
7279	Co-59	7280	Ni-60	7282	Ni-62
7288	Ni-58	7293	Cu-63	7295	Cu-65
7400	Zr-90	7402	Zr-92	7404	Zr-94
7413	Nb-93	7420	Mo-100	7422	Mo-92
7428	Mo-98	7495	In-115	7500	Sn-120
7502	Sn-122	7504	Sn-124	7537	I-127
7579	La-139	7664	Dy-164	7731	Ta-181
7797	Au-197	7902	Th-232	7911	Pa-231
7913	Pa-233	7923	U-233	7928	U-238
7951	Am-241	9479	Ag-109		

9. 174 group neutron special purpose gas production 1,312 FB 12800 80
(18 materials) *

ID	Material	ID	Material	ID	Material
5160	B-11	5272	Li-7	5275	N-14
5303	Li-6	5304	Be-9	5305	B-10
5306	C-0	5309	F-19	5313	Al-27
5314	Si-28	5322	Ti-0	5323	V-0
5324	Cr-0	5325	Mn-55	5326	Fe-0
5327	Co-59	5328	Ni-0	5329	Cu-0

10.	JCL to create subroutine library	99	FB	12800	80
11.	ALIAS - PL1 source used to create subroutine library	63	¦	¦	¦
12.	Step 1 of ZPR-6/7 infinite medium problem JCL and sample problem input to compile, link and execute AIM, AJAX, BONAMI, RADE, and NITAWL. NITAWL creates an AMPX working library for XSDRNPM and an ANISN binary tape for GIP.	217	¦	¦	¦
13.	Step 2 of ZPR-6/7 infinite medium problem via XSDRNPM	71	¦	¦	¦
14.	Step 2 of ZPR-6/7 infinite medium problem via ANISN	107	¦	¦	¦
15.	Step 1 of CTR standard blanket problem JCL and sample problem input to compile, link and execute AIM, CHOX, RADE, DIAL, MALOCS and NITAWL. NITAWL creates an AMPX working library for XSDRNPM and an ANISN binary tape for GIP.	269	¦	¦	¦
16.	Step 2 of CTR standard blanket problem via XSDRNPM	71	¦	¦	¦
17.	Step 2 of CTR standard blanket problem via ANISN	154	¦	¦	¦
	Total logical records (LRECL=80)	1,230,343			
	Total bytes (LRECL=80)	98,427,440			
18.	Output from step 1 of ZPR sample problem	3865	FB	3990	133
19.	Output from step 2 of ZPR sample problem (XSDRNPM)	911	¦	¦	¦
20.	Output from step 2 of ZPR sample problem (ANISN)	1362	¦	¦	¦
21.	Output from step 1 of CTR sample problem	4979	¦	¦	¦
22.	Output from step 2 of CTR sample problem (XSDRNPM)	4894	¦	¦	¦
23.	Output from step 2 of CTR sample problem (ANISN)	2462	¦	¦	¦
	Total logical records (LRECL=133)	18,473			
	Total bytes (LRECL=133)	2,456,909			
	Total logical records in package	1,248,816			
	Total bytes in package	100,884,349			

Requires approximately 50% of 2400 foot 9-track 6250 BPI tape blocked 12800

NOTES:
Data in files 1-9 are in AMPX master library format.

Data on files 7, 8, and 9 are reaction cross sections based on the particular ENDF/B-V special purpose file.

The JCL and input data are structured as though PSR117/MARS is being used to execute the sample problems. The MARS package is recommended for use with this data package.

This tape is written in EBCDIC mode.

ORNL/RSIC-XX

VITAMIN-E: A Coupled 174-Neutron, 38 Gamma-ray Multigroup Cross-Section Library for Deriving Application-Dependent Working Libraries for Radiation Transport Calculations.

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J. Barhen, and D. G. Cacuci.

Work supported by DOE-RRT(Base Technology), DOE-OFE, and DNA>

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VITAMIN-E

INTRODUCTION

Background Information About the VITAMIN Concept

The "VITAMIN" concept is a different approach to obtaining appropriate cross sections for radiation transport analysis. With it, users have access to a fine-group data base from which application-dependent libraries can be derived. Temperature and resonance self-shielding correction can be made through the Bondarenko self-shielding factor method [1]. The VITAMIN concept extends this "table-look up" approach to a wider range of radiation transport problems by utilizing many energy groups and by accommodating photon production and transport. The concept bridges the gap between the difficult and expensive task of generating a new data library from ENDF/B for each application and using an inflexible cross section set (perhaps designed for an entirely different application).

Success of the VITAMIN Concept

The effort to produce a general-purpose fine-group cross-section library based on ENDF/B-V was undertaken because of the success of VITAMIN-C [2]. Based on ENDF/B-IV, the VITAMIN-C library has been applied to a wide variety of neutron and photon transport problems. These included fusion neutronics [3-6], fast reactor shielding [7] and physics [8-10] calculations, light water reactor shielding[11] and pressure vessel [12] dosimetry applications, shipping cask design[13], shielding benchmarks [14] and air transport analyses [15].

Development of a VITAMIN Library Based on ENDF/B-V

In anticipation of the release of a new major version of ENDF/B, comments on desired characteristics of a successor to VITAMIN-C were solicited. The results are summarized in Ref. 16, the new version being designated VITAMIN-E. Since the preparation of these specifications, work on the successor has continued at a modest pace. A few minor adjustments to the stated specifications were necessary as the production and testing of the data library proceeded.

The final characteristics of VITAMIN-E are given in the next section. In following sections the availability and suggestions for obtaining the library are discussed, some applications of VITAMIN-E are enumerated, and future plans are proposed.

SPECIFICATIONS OF THE INITIAL RELEASE OF VITAMIN-E

The acronym VITAMIN is derived from the phrase Versatile Integrated Techniques using AMPX and MINX for Investigating Neutronics. The designation "E" (the fifth letter of the alphabet) represents the alphabetic

equivalent of the fifth version of ENDF/B, the primary source of evaluated data used to produce VITAMIN-E. The package is part of the Radiation Shielding Information Center Data Library Collection and is designated DLC-113/VITAMIN-E.

Sources of Evaluated Data

As indicated above, the primary source of evaluated neutron and photon production data for VITAMIN-E is ENDF/B-V [17]. Additional materials were obtained by processing the Lawrence Livermore National Laboratory (LLNL) evaluated data library LENDL [18]. The LENDL data were used if a desired evaluation was not available from ENDF/B or, in some cases, if the ENDF/B evaluation was incomplete (e.g., no photon production data). Data sets for tungsten [19] were processed from evaluations prepared by Los Alamos National Laboratory (LANL) which have subsequently been released as part of Revision 2 of ENDF/B-V. A LANL evaluation for beryllium [20], which provides a better representation of the physics of the (n,2n) reaction than ENDF/B-V formats allow was processed and included in VITAMIN-E. The ENDF/B-V beryllium is also included for comparison purposes.

The photon interaction data used for VITAMIN-E is a recent evaluation by Hubbell [21] that has been packaged in RSIC as DLC-99/HUGO [22] and will be released as the ENDF/B-V photon interaction evaluated data.

Contents of VITAMIN-E

Two types of neutron multigroup data are provided; (1) data for transport calculations and, (2) neutron reaction cross sections. Data sets for transport calculations, 77 in number, are derived primarily from the ENDF/B-V General Purpose File. Of these, 56 data sets have both neutron and secondary photon production. The materials, Legendre expansion order for scattering matrices, number of temperatures, and background cross section values are listed in Table 1. The identification of the evaluated data sets is summarized in Table 2.

The neutron reaction cross section data are derived from the ENDF/B-V Special Purpose Files. The isotopes and particular reactions that are included are listed in Tables 3, 4, and 5, respectively.

Photon interaction transport cross sections with P_8 scattering matrices are provided for all elements in the Z-range of 1 to 100.

Energy Group Structure

The energy group structure is a slight refinement of that used for VITAMIN-C. The neutron group energy and lethargy boundaries are shown in Table 6. The photon group energy boundaries are shown in Table 7. The (174n,38g) (VITAMIN-C has 171n,36g) group structures allowed for the formal inclusion of the group boundaries of the Defense Nuclear Agency few-group library DLC-31/FEWG1 [23]. Both the neutron and photon energy ranges now extend to (essentially) 20 MeV.

~~extend to (essentially) 20 MeV~~.

Weighting Functions

Several considerations enter into the selection of a smooth weighting function, W(E). Experience in cooperative processing methods testing has shown that the use of a more realistic smooth weighting function would help in assuring computation of proper group constants. In particular, below approximately 25 keV the weighting function used previously was 1/E while it is known that typical LMFBR spectra in this energy range are roughly constant and do not increase with decreasing energy. Similarly, the point at which the 1/E spectrum was joined to the fission spectrum was 821 keV in the VITAMIN-C library, while testing has shown that a higher value might be more reasonable. One must be quite careful with these types of arguments however, since for typical shielding situations the flux spectrum is often 1/E and sometimes even $1/E^n$, where $n > 1$. Also, it is perceived that the use of any particular reactor-dependent weighting function may detract from the generality of the resulting shielding factor library. Finally, built-in spectral corrections (e.g. elastic removal) are intended to account for some of these effects and the associated theories are currently based on a 1/E model. It is also recognized that in special circumstances other sets of weighting functions may be required.

Thus, the weighting function specified obviously represents a set of compromises which hopefully can provide adequate accuracies over the range of intended applications. Basically, the energy at which the fission spectrum can be smoothly joined to the 1/E spectrum (i.e. where flux/lethargy is constant) is found to be equal to 3/2 of the fission spectrum temperature (taken here as ~ 1.4 MeV). To provide consistency with the selected energy group boundaries, the (1/E)/fission spectrum cutpoint was raised to 2.1224 MeV, a boundary equal to $3/2\,\theta$ (θ = 1.415 MeV). Below this energy and above 0.414 eV, the W(E) spectrum is assumed to be 1/E with the knowledge that elastic removal corrections may be used to improve estimates of flux shapes. Shielding and thermal reactor problems also favor this shape. It should also be noted that the Maxwellian spectrum in the thermal range was fixed at 300 K for the VITAMIN-C library; it is temperature dependent in the VITAMIN-E library. For the infinitely dilute data, the smooth flux spectrum temperature in the thermal range is still set at 300 K. However, for the 300-, 900-, and 2100-K cases the thermal spectrum temperature is consistent with the temperature of the broadened ENDF data. The cutpoint for the Maxwellian and 1/E shapes is fixed at 0.414 eV independent of temperature (the original 5 kT guideline was only nominal).

The weighting function for the photon interaction cross sections is constant in energy.

The functional form of the neutron and photon production weighting function is summarized below.

Functional form	Energy limits	(see Table 6)
1) Maxwellian Thermal Spectrum $W(E) = C_1 E \exp[-E/kT]$	10^{-5}eV to 0.414 eV	173-174
2) "1/E" Slowing-Down Spectrum $W_2(E) = C_2/E$	0.414 eV to 2.12 MeV	48-172
3) Fission Spectrum (= 1.415 MeV) $W_3(E) = C_3 E^{1/2} \exp[-E/\ ]$	2.12 MeV to 10.0 MeV	16-47
4) "1/E" Spectrum $W_4(E) = C_4 E$	10.0 MeV to 12.52 MeV	11-15
5) Velocity Exponential Fusion Peak (E_p = 14.07 MeV) (kT = 0.025 MeV) $W_5(E) = C_5 \exp\{-\frac{5}{kT}(E^{1/2}-E_p^{1/2})^2\}$	12.52 MeV to 15.68 MeV	5-10
6) "1/E" Spectrum $W_6(E) = C_6/E$	15.68 MeV to 19.64 MeV	1-4

The values of the group integrals of the weighting functions for neutron and photon groups are listed in Tables 6 and 7, respectively.

Processing Codes

The neutron transport cross section and Bondarenko factor processing was done using the MINX 1.5 [24,25] system. Photon production and transport cross section processing was done using the LAPHNGAS and SMUG modules of AMPX [26]. The processing codes were upgraded to handle Version V formats [27] and tested as part of the CSEWG Data Testing Process. This combination was chosen again because of its successful performance in the generation of VITAMIN-C, the experience obtained by ORNL and General Electric as part of the preliminary ENDF/B-V Data Testing [28], and the improvements to the modules as described below:

(1) Inclusion of competitive widths, interference scattering, and resonance overlap in the unresolved energy region derived from modified versions [29] of the MC^2-2 algorithm [30]. This was done for the total, elastic, fission, and capture cross sections.
(2) A current weighting of the total cross section including competitive widths, interference scattering, within sequence and sequence-sequence overlap corrections [29].
(3) Improvement of the numerical procedure [31] employed in MINX to obtain the infinitely-dilute group averaged cross section in the unresolved energy region.
(4) Upgrade to handle Version V formats
    (a) Energy dependent Watt fission spectra are read. As in VITAMIN-C, only the 1 MeV spectrum is used in the VITAMIN-E

library (MINX does not currently process fission matrices).
(b) MOD number insert
(c) Isotropic flag in file 4
(d) Addition of competitive width processing.

(5) New and upgraded modules for Doppler broadening, linearization, and thinning which:
(a) thin the 0 K data before broadening. This process saves enormous amounts of computing time.
(b) Guarantee of a sufficient number of points in the Doppler broadened tape such that linear interpolation is accurate to within a fixed input criteria.

(6) Added capability to process elastic removal f-factors.

(7) Improvements to Photon Production Processing.
(a) Experience with ENDF/B-IV photon production processing suggested linearization of the weight function to help improve the precision of the LAPHNGAS calculation.
(b) A coding change was made to the LAPHNGAS code to allow better precision when an input weighting spectrum is used (the case for VITAMIN-C). The change improves the integration scheme by incorporating the weighting function energy grid into the energy grid used by LAPHNGAS to calculate the group averaged multiplicities or photon production cross sections.
(c) A feature was also added to LAPHNGAS to allow calculation of multi-group multiplicities for materials without resonance parameters. This permits self shielding of photon production cross sections, e.g., from capture, using the Bondarenko scheme.

(8) A MINX change was made to permit the smooth energy weighting function to be temperature dependent in the thermal range.

Convergence Parameters Used in Processing

The following criteria were used in the various stages of processing the neutron transport cross sections.

(1)	reconstruction	0.5%
(2)	linearization	0.2%
(3)	thinning	0.2%
(4)	integration	0.1%

Special Purpose File Processing

The special purpose reaction cross sections were processed with the XLACS3 module of AMPX-III. The XLACS3 module was used to reduce data manipulation problems that were encountered because MINX I.5 expects a total cross section in its general processing and one is not provided for the special purpose files. The approach was used only after verifying that MINX 1.5 and XLACS3 yield the same group averages for simple reaction cross section processing.

Output Formats

The initial release of VITAMIN-E is in AMPX format (as was DLC-41/VITAMIN-C). It is expected that later releases will be provided in CCCC and MATXS formats.

The reaction identifiers used in VITAMIN-E are the same as for ENDF/B-V (see Appendix B of Ref. 27) except as indicated in Table 8.

AVAILABILITY OF VITAMIN-E

RSIC Data Library Package DLC-113/VITAMIN-E

The initial release of the library in AMPX format is DLC-113/VITAMIN-E which is available from

Radiation Shielding Information Center
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37831

Because of the large amount of data involved, requesters should provide four, 2400-ft reels of magnetic tape to be written at 1600 bpi (1 tape written at 6250 bpi will suffice).

Computer Codes for Retrieving and Manipulating VITAMIN-E

As with VITAMIN-C, VITAMIN-E is designed to be used in conjunction with the PSR-63/AMPX-II or PSR-117/MARS package. The sample problems in DLC-113 can be executed using the AMPX modules from PSR-63 or PSR-117.

VITAMIN-E on the NMFECC

A selected portion of the VITAMIN-E materials have been placed on the National Magnetic Fusion Energy Computing Center (NMFECC) at Lawrence Livermore National Laboratory in formats directly useable in discrete ordinates codes. In the future, versions of VITAMIN-E in AMPX and MATXS formats will be placed on NMFECC.

APPLICATIONS OF VITAMIN-E

During the development stages and prior to the general release of VITAMIN-E from RSIC, the data have been used in a wide range of applications. The experience gained through these exercises has provided confidence in the performance of the library.

CSEWG Fast Reactor Data Testing

Early versions of VITAMIN-E were used in the Cross Section Evaluation Working group fast reactor and shielding data testing [32] of ENDF/B-V. This exercise provided a test of both the basic ENDF/B data the the processing methods used for VITAMIN-E. Satisfactory results were obtained and agreement was found with other laboratories that participated in the data testing effort.

DNA Few-Group Cross-Section Library

The ENDF/B-V based versions of the Defense Nuclear Agency (DNA) Few Group Cross-Section Library [23] will be derived by collapsing VITAMIN-E. The library will be available in the same 37-neutron, 21-photon group structure used in the current version (based on ENDF/B-IV). In addition, a new, somewhat expanded group structure will be developed to resolve some current questions [33] associated with radiation transport in the atmosphere. A new library will be prepared by collapsing VITAMIN-E into the new, recommended group structure.

NEACRP Fast Reactor Benchmark Exercise

The Nuclear Energy Agency Committee on Reactor Physics (NEACRP) fast reactor shielding benchmark exercise had participants from several countries. Results [14] based on both VITAMIN-C and VITAMIN-E were provided by ORNL and served as a useful basis for comparison among the participants.

Fusion Integral Experiment Analysis

The fusion integral experiment program at ORNL is a systematic program to study shielding and duct streaming configurations important for fusion reactor designs. Analysis of these experiments is performed both with continuous energy Monte Carlo and multigroup discrete ordinates methods [34]. The latter employs multigroup cross section data based on VITAMIN-E.

Analysis of Fusion/Fission Hybrid Design

The inclusion of processed multigroup data for fissile materials allows the applications of VITAMIN-E to fusion-fission hybrid systems. An example is a recent ORNL study described in Ref. 35.

Analysis of Accelerator Breeder Concepts

Another application for VITAMIN-E is the study of the accelerator breeder concept. Recent work at ORNL evaluated the ternary metal accelerator breeder blanket concept [36].

Studies in Support of Fusion Neutronics Data Needs Assessment

As was indicated earlier, a selected set of materials from VITAMIN-E was placed on NMFECC. These were used in studies by GA Technologies in support of fusion neutronics data needs assessment [37].

Fusion Blanket Comparison Selection Study

The most recent evaluated data for Li-7, MAT 1397, from Los Alamos National Laboratory has been included among the selected VITAMIN-E data sets on NMFECC. This, along with necessary structural material was an important addition to the resources available for the Argonne National Laboratory blanket comparison selection study [38].

Light Water Reactor Dosimetry

Both VITAMIN-C and VITAMIN-E have been used in the development and demonstration of advanced methodology for LWR dosimetry applications [12].

FUTURE PLANS FOR ADDITIONS TO VITAMIN-E

Additional Materials for Radiation Transport

The VITAMIN-E data library is maintained and updated at RSIC. Feedback on its use is very important and users are urged to contact RSIC about problems and suggestions.

New evaluations or revisions to ENDF/B-V which are of interest in the application areas of VITAMIN-E will be processed as needs arise.

Additional Special Purpose Data Libraries

It is expected that additional special purpose libraries in the group structure of VITAMIN-E will be produced at ORNL and elsewhere. For example, an activation library based on LLNL's ACTL82 has already been produced. A Kerma factor data library is likely to be developed within a year. Producers of such data libraries are urged to make them available via the Radiation Shielding Information Center.

Expected Release of Updated Retrieval and Manipulation Programs

A new release of AMPX, AMPX-III, is expected in the next fiscal year. This will allow the improved treatment of the Bondarenko factor data in VITAMIN-E.

REFERENCES

[1] I. I. Bondarenko, ed., "Group Constants for Nuclear Reactor Calcula-

tions," Consultants Bureau, New York (1964).

[2] R. W. Roussin, et al, VITAMIN-C: The CTR Processed Multigroup Cross-Section Library for Neutronics Studies, ORNL/RSIC-37 (ENDF-296), 1978.

[3] R. T. Santoro, R. A. Lillie, R. G. Alsmiller, Jr., and J. M. Barnes, "Two- and Three-Dimensional Neutronics Calculations for the TFTR Neutral Beam Injectors," Nucl. Sci. Eng. 70: 225 (1979).

[4] R. T. Santoro, R. A. Lillie, R. G. Alsmiller, Jr., and J. M. Barnes, "Shielding Calculations for the TFTR Neutral Beam Injectors," Nucl. Technol. 47: 412 (1980).

[5] R. G. Alsmiller, Jr., et al, "Dose Rates from Induced Activity in the TFTR Test Cell," Nucl. Technol. 48: 187 (1980).

[6] R. A. Lillie, R. T. Santoro, R. G. Alsmiller, Jr., and J. M. Barnes, Neutron and Gamma Ray Streaming Calculations for the ETF Neutral Beam Injectors, submitted to Nucl. Technol., preprinted as ORNL/TM-7705 (1980).

[7] R. E. Maerker, F. J. Muckenthaler, and C. E. Clifford, SB4. Measurements and Calculations of the ORNL CRBR Upper Axial Shield Experiment, ORNL-5259(ENDF-258) (1977).

[8] D. L. Selby and G. F. Flanagan, Analysis of ZPPR-5 Source Level Flux Monitor Experiments, ORNL-5314 (1973).

[9] E. M. Bohn et al., eds., Benchmark Testing of ENDF/B-IV, Brookhaven National Laboratory, Upton, New York, BNL-NCS-21118 (Vol. I and II) (1976).

[10] T. J. Burns and J. R. White, Preliminary Evaluation of Alternative LMFBR Fuel Cycle Options, ORNL-5389.

[11] American National Standard, Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations for Nuclear Power Plants, ANSI/ANS-6.1.2-1983, American Nuclear Society, LaGrange Park, Ill. (1983).

[12] R. E. Maerker, J. J. Wagschal, and B. L. Broadhead, Development and Demonstration of an Advanced Methodology for LWR Dosimetry Applications, Electric Power Research Institute report EPRI-NP-2188 (1981).

[13] W. E. Ford, III, et al., FCXSEC: Multigroup Cross-Section Libraries for Nuclear Fuel Cycle Shielding Benchmark Calculations, ORNL/TM-7038 (ENDF-287) (1980).

[14] G. Palmiotti and M. Salvatores, Proceedings of the NEACRP Specialists' Meeting on Shielding Benchmark Calculations, Part 1, LMFBR Shielding Benchmark, NEACRP-L-263 (1983).

[15] J. V. Pace, III, J. R. Knight, and D. E. Bartine, "Transport in an

Air-Over-Ground Environment of Prompt Neutrons and Gammas from the Hiroshima and Nagasaki Weapons," Reevaluations of Dosimetric Factors for Hiroshima and Nagasaki, Dept. of Energy Symposium Series 55, CONF-810928 (1981).

[16] C. R. Weisbin, et al., VITAMIN-E: An ENDF/B-V Multigroup Cross-Section Library for LMFBR Core and Shield, LWR Shield, Dosimetry and Fusion Blanket Technology, ORNL-5505(ENDF-274) (1979).

[17] R. Kinsey, comp., ENDF/B Summary Documentation, Brookhaven National Laboratory, Upton, New York, BNL-NCS-17541(ENDF-201), 3d Ed. (ENDF/B-V) (1979).

[18] R. J. Howerton and M. H. MacGregor, The LLL Evaluated Nuclear Data Library (ENDL): Description of Individual Evaluations for Z=0-98, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-50400 Vol. 15 (May 1978).

[19] E. D. Arthur, P. G. Young, A. B. Smith, and C. A. Philis, "New Tungsten Isotope Evaluations for Neutron Energies Between 0.1 and 20 MeV," Trans. Am. Nucl. Soc., 39: 793 (1981).

[20] P. G. Young and L. Stewart, Evaluated Data for n + Be-9 Reactions, Los Alamos National Laboratory, Los Alamos, New Mexico, LA-7932-MS (1979).

[21] J. H. Hubbell, H. A. Gimm, and I. Overbo, "Pair, Triplet, and Total Atomic Cross Sections for 1 MeV - 100 GeV Photons in Elements Z = 1 to 100," J. Phys. Chem. Ref. Data, 9:1023 (1980).

[22] R. W. Roussin, J. R. Knight, J. H. Hubbell, and R. J. Howerton, Description of the DLC-99/HUGO Package of Photon Interaction Data in ENDF/B-V Format, ORNL/RSIC-46(ENDF-335).

[23] D. E. Bartine, J. R. Knight, J. V. Pace, III, and R. W. Roussin, Production and Testing of the DNA Few-Group Coupled Neutron-Gamma Cross-Section Library, ORNL/TM-4840 (1977).

[24] J. E. White, et al, "MINX I.5: User's Manual for the ENDF/B-V IBM Version of the MINX Cross Section Processing Program," to be published.

[25] C. R. Weisbin, et al., MINX, A Multigroup Interpretation of Nuclear Cross Sections, Los Alamos National Laboratory, Los Alamos, New Mexico, LA-6486-MS (1976).

[26] N. M. Greene, et al., AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B, ORNL/TM-3706 (1976).

[27] R. Kinsey and B. A. Magurno, eds., Data Formats and Procedures for the Evaluated Nuclear Data File, ENDF/B-V, Brookhaven National Laboratory, Upton, New York, BNL-NCS-50496(ENDF-102) 2d Ed., (revised 1983).

[28] R. Q. Wright, J. E. White, D. G. Cacuci, and C. R. Weisbin, "ORNL Fast Reactor Data Testing Results Based on Pre-Preliminary ENDF/B-V Data," Sept. 11, 1978, see "Report of the CSEWG Data Testing - Evaluator Meeting, Sept. 11-13, 1978, Brookhaven National Laboratory, Upton, New York, CSEWG Newsletter #57; see also E. Kujawski and S. L. Beamon, "Phase II Testing of Preliminary ENDF/B-V at GE-ARSK." Letter to the Cross Section Evaluation Working Group, Sept. 5, 1978).

[29] D. G. Cacuci, "Cross Section Treatment in the Unresolved Energy Region at ORNL for ENDF/B-V Using MINX," Sept. 7, 1978, ORNL Internal Memo. See also J. Barhen and D. G. Cacuci, "Cross Section Processing for the Unresolved Energy Region," ORNL Internal Memo, Dec. 4, 1978.

[30] H. Henryson, II, B. J. Toppel, and C. G. Stenberg, MC2-2: A Code to Calculate Fast Neutron Spectra and Multigroup Nuclear Cross Section Processing, Argonne National Laboratory, Argonne, Ill., ANL-8144 (ENDF-239) (June 1976).

[31] J. Barhen, et al., VITAMIN-E: A Multipurpose ENDF/B-V Coupled Neutron-Gamma Cross-Section Library, Proc. Inter. Conf. Nucl. Cross Sections for Technology, Knoxville, Tenn., NBS Spec. Publ. 594, 1980.

[32] C. R. Weisbin, et al, eds., Benchmark Data Testing of ENDF/B-V, BNL-NCS-31531 (ENDF-311) (1982).

[33] G. D. Kerr, J. V. Pace, III, and W. H. Scott, Tissue Kerman vs. Distance Relationship for Initial Nuclear Radiation from the Atomic Devices Detonated Over Hiroshima and Nagasaki, ORNL/TM-8727 (1983).

[34] R. T. Santoro, J. M. Barnes, R. G. Alsmiller, Jr., and J. E. Drischler, "Monte carlo and Discrete Ordinates Calculations of 14-MeV Neutrons Streaming Through a Cylindrical SS 304 Duct: Comparison with Experiment I," to be published.

[35] J. O. Johnson and T. J. Burns, Neutronic Evaluation of Fissile Fuel breeding Blankets for the Fission-Suppressed Tandem-Mirror Hybrid Reactor, ORNL/TM-8976 (1984).

[36] J. O. Johnson, D. E. Bartine, and T. A. Gabriel, Accelerator Breeder Fuel Reproduction: A Concept Evaluation of a Ternary Metal Accelerator Breeder Blanket, ORNL/TM-8999 (1984).

[37] E. T. Cheng and J. H. Huang, "A Nuclear Data Library Comparison Study for Tritium Breeding in Helium-Cooled Fusion Blankets," Trans. Am. Nucl. Soc. 46: 279 (1984).

[38] D. Smith, et al, Blanket Comparison Selection Study---Final Report, ANL/FPP-84-1 (1984).

[39] R. T. Santoro, J. E. White, and J. D. Drischler, ORACT: A 174-Neutron-Group Activation Cross-Section Library for Fusion and Fission Reactor Design Studies, ORNL/TM-9203 (1984).

Table 1. Background cross section values at which Bondarenko factors are tabulated in VITAMIN-E.

	Background cross sections (barns)													
Material	10^{10}	10^{5}	10^{4}	10^{3}	300	10^{2}	30	10^{1}	3	10^{0}	10^{-1}	10^{-6}	Temps*	Leg. order
H-1	X			X		X		X		X		X	4	8
H-3	X			X		X		X		X		X	4	8
He-4	X		X	X		X		X		X		X	4	8
Li-6	X			X	X	X	X	X	X	X	X	X	4	8
Li-7	X			X	X	X	X	X	X	X	X	X	4	8
Be-9	X			X		X		X		X		X	4	8
B-10	X			X		X		X		X		X	4	8
B-11(1160/0)	X	X	X	X		X		X		X		X	4	8
B-11(8811)	X	X	X	X		X		X		X		X	4	3
C-12	X			X	X	X	X	X		X	X	X	4	8
N-14	X			X		X		X		X		X	4	8
O-16	X			X		X	X	X	X	X	X	X	4	8
F-19	X		X	X		X		X		X		X	4	3
Na-23	X			X	X	X	X	X	X	X	X	X	4	8
Mg	X		X	X		X		X		X		X	4	8
Al-27	X		X	X		X	X	X	X	X		X	4	8
Si	X			X		X	X	X	X	X		X	4	8
P	X			X		X		X		X		X	4	3
S	X			X		X		X		X		X	4	3
Cl	X			X		X		X		X		X	4	3
Ar	X		X	X		X		X		X		X	4	3
K	X			X		X		X		X		X	4	8
Ca	X			X		X		X		X		X	4	8
Ti	X		X	X		X		X		X		X	4	8
V	X		X	X		X		X		X		X	4	3
Cr	X			X	X	X	X	X	X	X	X	X	4	8
Mn-55	X		X	X		X		X		X		X	4	8
Fe	X			X	X	X	X	X	X	X	X	X	4	8
Co-59	X			X		X		X		X		X	4	8
Ni	X			X	X	X	X	X	X	X		X	4	8
Cu	X			X		X		X		X	X	X	4	8
Ge	X		X	X		X		X		X		X	4	5
Y-89	X												2	3
Zr(1340/2)	X												2	3
Zr(8841)	X			X		X		X		X		X	4	3
Nb-93	X		X	X		X		X		X		X	4	3
Mo	X		X	X		X		X		X		X	4	8
Cd	X			X		X		X		X		X	4	3
Sn	X		X	X		X		X		X		X	4	3
Cs-137	X												2	3
Gd	X		X	X		X		X		X		X	4	3
Hf-174	X												2	3
Hf-176	X												2	3

Table 1 (continued)

	Background cross sections (barns)													
Material	10^{10}	10^{5}	10^{4}	10^{3}	300	10^{2}	30	10^{1}	3	10^{0}	10^{-1}	10^{-6}	Temps*	Leg. order
Hf-177	X												2	3
Hf-178	X												2	3
Hf-179	X												2	3
Hf-180	X												2	3
Ta-181	X												2	3
W-182	X		X	X		X		X		X		X	4	8
W-183	X		X	X		X		X		X		X	4	8
W-184	X		X	X		X		X		X		X	4	8
W-186	X		X	X		X		X		X		X	4	8
Re-185	X												2	3
Re-187	X												2	3
Au-197	X		X	X		X		X		X		X	4	3
Pb	X			X		X		X		X	X	X	4	8
Th-232	X		X	X	X	X	X	X		X	X	X	5	3
Pa-233	X		X	X		X		X		X		X	4	3
U-233	X		X	X	X	X		X		X		X	5	3
U-234	X	X	X	X		X		X		X		X	4	5
U-235	X		X	X	X	X		X		X		X	5	5
U-236	X	X	X	X		X		X		X		X	4	3
U-238	X		X	X	X	X	X	X		X	X	X	5	5
Np-237	X	X	X	X		X		X		X		X	4	3
Pu-238	X	X	X	X		X		X		X		X	4	3
Pu-239	X		X	X	X	X		X		X		X	5	5
Pu-240	X	X	X	X		X		X		X		X	5	5
Pu-241	X	X	X	X		X		X		X		X	5	5
Pu-242	X	X	X	X		X		X		X		X	5	3
Am-241	X	X	X	X		X		X		X		X	4	3
Am-242	X	X	X	X		X		X		X		X	5	3
Am-242m	X	X	X	X									5	3
Am-243	X	X	X	X		X		X		X		X	5	3
Cm-242	X	X	X	X		X		X		X		X	5	3
Cm-243	X	X	X	X		X							5	3
Cm-244	X	X	X	X		X		X		X		X	5	3

*2 temperatures implies Bondarenko factors at 0 and 300K; 4 implies 0,300,900, and 2100K; and 5 implies 0,300,900, 2100, and 3500K.

Table 2. Identification of VITAMIN-E cross sections for transport calculations

Material	Neutron MAT/MOD	Photon prod. MAT/MOD	Laboratory (see Key)	Evaluators
H-1	9301	9301	ORNL	L. Stewart, R. J. LaBauve, P. G. Young
H-3	1169/0	N.A.*	LASL	L. Stewart
He-4	1270/0	N.A.	LASL	G. Hale, R. Nisley, P. Young
Li-6	1303/1	1303/1	LASL	G. Hale, L. Stewart, P. Young
Li-7	1397/1	1397/1	LASL	L. Stewart, D. Foster, M. Battat, R. LaBauve
Be-9	1304/1	1304/1	LLL	R. Howerton, T. Perkins
Be-9	104**	104	LASL	P. Young, L. Stewart
B-10	1305/1	1305/1	LASL	G. Hale, L. Stewart, P. Young
B-11	1160/0	N.A.	GE, BNL	C. Cowan
B-11	8811***	8811	LLL	R. Howerton, T. Perkins
C-12	1306/1	1306/1	ORNL	C. Y. Fu, F. G. Perey
N-14	1275/1	1275/1	LASL	P. Young, D. Foster, Jr., G. Hale
O-16	1276/1	1276/1	LASL	P. Young, D. Foster, Jr., G. Hale
F-19	1309/2	1309/2	ORNL	C. Y. Fu, D. C. Larson, F. G. Perey
Na-23	1311/1	1311/1	ORNL	D. C. Larson
Mg	1312/1	1312/1	ORNL	D. C. Larson
Al-27	1313/1	1313/1	LASL	P. Young, D. Foster, Jr.
Si	1314/1	1314/1	ORNL	D. C. Larson, F. G. Perey, M. K. Drake, P. Young
P	1315/1	1315/1	LLL	R. Howerton
S	1347/1	1347/1	BNL	M. Divadeenam
Cl	1149/1	1149/1	GGA	M. S. Allen, M. K. Drake
Ar	8824***	8824	LLL	R. Howerton
K	1150/1	1150/1	GGA	M. K. Drake
Ca	1320/3	1320/3	ORNL	C. Y. Fu, F. G. Perey
Ti	1322/1	1322/1	BUR, ANL, LLL	C. Philis, A. Smith, R. Howerton
V	1323/1	1323/1	ANL, LLL, HEDL	A. Smith, R. Howerton, F. Mann
Cr	1324/1	1324/1	BNL	Prince, T. W. Burrows
Mn-55	1325/2	1325/2	BNL	S. F. Mughabghab
Fe	1326/3	1326/3	ORNL	C. Y. Fu, F. G. Perey
Co-59	1327/3	1327/3	BNL	S. F. Mughabghab
Ni	1328/2	1328/2	BNL	M. Divadeenam
Cu	1329/1	1329/1	ORNL, SAI	C. Y. Fu, M. K.Drake, Fricke
Ga	1358/1	1358/1	LLL, LASL	R. Howerton, P. Young
Y-89	9202/1	N.A.	HEDL	R. Schenter, F. Schmittroth
Zr	1340/2	N.A.	SAI	M. K. Drake, D. Sargis, T. Maung
Zr	8841***	8841	LLL	R. Howerton
Nb-93	1189/1	1189/1	ANL, LLL	R. Howerton, A. Smith
Mo	1321/1	1321/1	LLL, HEDL	R. Howerton, F. Schmittroth, R. Schenter
Cd	8847***	8847	LLL	R. Howerton
Sn	8850***	8850	LLL	R. Howerton
Cs-137	9669/1	N.A.	HEDL, INEL	R. Schenter, F. Schmittroth, Reich
Gd	8853***	8853	LLL	R. Howerton
Hf-174	1374/1	N.A.	SAI	M. K. Drake, D. Sargis, T. Maung
Hf-176	1376/1	N.A.	SAI	M. K. Drake, D. Sargis, T. Maung
Hf-177	1377/1	N.A.	SAI	M. K. Drake, D. Sargis, T. Maung
Hf-178	1378/1	N.A.	SAI	M. K. Drake, D. Sargis, T. Maung

Table 2. (continued)

Material	Neutron MAT/MOD	Photon prod. MAT/MOD	Laboratory (see Key)	Evaluators
Hf-179	1383/1	N.A.	SAI	M. K. Drake, D. Sargis, T. Maung
Hf-180	1384/1	N.A.	SAI	M. K. Drake, D. Sargis, T. Maung
Ta-181	1285/2	1285/2	LLL	R. Howerton, T. Perkins, MacGregor
W-182	182**	182	AI, LASL	Otter, Ottewitte, Rose, P. Young
W-183	183**	183	AI, LASL	Otter, Ottewitte, Rose, P. Young
W-184	184**	184	AI, LASL	Otter, Ottewitte, Rose, P. Young
W-186	186**	186	AI, LASL	Otter, Ottewitte, Rose, P. Young
Re-185	1083/1	N.A.	GE-NMPO	W. Henderson, J. Zwick
Re-187	1084/1	N.A.	GE-NMPO	W. Henderson, J. Zwick
Au-197	8861***	8861	LLL	R. Howerton
Pb	1382/2	1382/2	ORNL	C. Y. Fu, F. G. Perey
Th-232	1390/1	1390/1	BNL	Bhat, Smith, Leonard, DeSaussure, et al
Pa-233	1391/1	N.A.	HEDL, INEL	F. Mann, R. Schenter, Reich
U-233	1393/1	N.A.	LASL, ORNL, HEDL	Stewart et al, L. weston, F. Mann
U-234	9394	N.A.	BNL, HEDL, INEL	M. Divadeenam, F. Mann, M. K. Drake, Reich et al
U-235	1395/1	1395/1	BNL	M. Bhat
U-236	1396/1	N.A.	BNL, HEDL, INEL	M. Divadeenam, F. Mann, McCrosson, Reich
U-238	1398/1	1398/1	ANL	Pennington, A. Smith, Poenitz
Np-237	1337/1	N.A.	HEDL, SRL, INEL	F. Mann, Benjamin, Smith, Stein, Reich
Pu-238	1338/1	N.A.	HEDL, AI, BNL	F. Mann, R. Schenter, Alter, Dunford
Pu-239	1399/1	1399/1	GE-FBRD, LASL	E. Kujawski, L. Stewart
Pu-240	1380/1	1380/1	ORNL	L. Weston
Pu-241	1381/1	1381/1	ORNL, LLL	L. Weston, R. Wright, R. Howerton
Pu-242	1342/1	1342/1	HEDL, SRL, LLL	F. Mann, Benjamin, Madland, R. Howerton
Am-241	1361/1	1361/1	HEDL, ORNL	F. Mann, R. Schenter, L. weston
Am-242	8542/1	N.A.	SRL	Benjamin, McCrosson
Am-242m	1369/1	1369/1	HEDL, SRL, LLL	F. Mann, Benjamin, R. Howerton
Am-243	1363/1	1363/1	HEDL, SRL, LLL	F. Mann, Benjamin, R. Howerton
Cm-242	8642/1	8642/1	HEDL, SRL, LLL	F. Mann, Benjamin, R. Howerton
Cm-243	1343/1	1343/1	HEDL, SRL, LLL	F. Mann, Benjamin, R. Howerton
Cm-244	1344/1	1344/1	HEDL, SRL, LLL	F. Mann, Benjamin, R. Howerton

* Gamma-ray production not available on ENDF/B-V.
** New evaluations from Los Alamos National Lab.
*** Evaluated data from Livermore Library(LENDL).

Laboratory Key

AI	=	Atomics International	HEDL	=	Hanford Engineering Development Laboratory
ANL	=	Argonne National Laboratory	INEL	=	Idaho National Engineering Laboratory
BNL	=	Brookhaven National Laboratory	LASL	=	Los Alamos National Laboratory
BUR	=	Bruyere la Chatel	LLL	=	Lawrence Livermore Laboratory
GE	=	General Electric	ORNL	=	Oak Ridge National Laboratory
GE-FBRD	=	Fast Breeder Reactor Division	SAI	=	Science Applications Incorportated
GE-NMPO	=	Nuclear Materials and Propulsion	SRL	=	Savannah River Laboratory
GGA	=	General Atomic			

Table 3. Dosimetry File Data in VITAMIN-E.

Z	Isotope	ENDF/B mat no.	Reaction Types
3	Li6	6424	He-4 Production
5	B10	6425	He-4 Production
11	Na23	6311	(N,G)
13	Al27	6313	(N,P), (N,A)
16	S32	6439	(N,P)
21	Sc45	6426	(N,G)
22	Ti46	6427	(N,P)
	Ti47	6428	(N,P), [(N,D)+(N,NP)]
	Ti48	6429	(N,P), [(N,D)+(N,NP)]
25	Mn55	6325	(N,2N)
26	Fe54	6430	(N,P)
	Fe56	6431	(N,P)
	Fe58	6432	(N,G)
27	Co59	6327	(N,2N), (N,G), (N,A)
28	Ni58	6433	(N,2N), (N,P)
	Ni60	6434	(N,P)
29	Cu63	6435	(N,G), (N,A)
	Cu65	6436	(N,2N)
49	In115	6437	(N,N)M, (N,G)
53	I127	6438	(N,2N)
79	Au197	6379	(N,G)
90	Th232	6390	(N,F), (N,G)
92	U235	6395	(N,F)
	U238	6398	(N,F), (N,G)
93	Np237	6337	(N,F)
94	Pu239	6399	(N,F)

Key: A = Alpha
D = Deuteron
F = Fission
G = Gamma Ray
M = Metastable state
N = Neutron
P = Proton

Table 4. Activation File Data in VITAMIN-E.
(Table continued on next page)

Z	Isotope	ENDF/B mat no.	Reaction Types
3	Li6	7036	(N,T)
	Li7	7037	(N,N)T, (N,G)
5	B10	7050	(N,T)2A
7	N14	7074	(N,P)
8	O16	7086	(N,P)
9	F19	7099	(N,2N)
11	Na23	7113	(N,2N), (N,G), (N,P)
12	Mg24	7124	(N,P)
13	Al27	7137	(N,G), (N,P), (N,A)
15	P31	7151	(N,P)
16	S32	7162	(N,P)
18	Ar40	7180	(N,G)
19	K41	7191	(N,P)
21	Sc45	7215	(N,2N), (N,G)
23	Ti46	7226	(N,P)
	Ti47	7227	(N,P), [(N,D)+(N,NP)]
	Ti48	7228	(N,P), [(N,D)+(N,NP)]
	Ti50	7220	(N,A)
24	Cr50	7240	(N,2N), (N,G)
	Cr52	7242	(N,2N)
25	Mn55	7255	(N,2N), (N,G)
26	Fe54	7264	(N,G), (N,P), (N,A)
	Fe56	7266	(N,2N), (N,P)
	Fe58	7268	(N,G)
27	Co59	7279	(N,2N), (N,G), (N,P)
28	Ni58	7288	(N,2N), (N,G), (N,P),
		.	(N,D), (N,A)
	Ni60	7280	(N,2N), (N,P)
	Ni62	7282	(N,G), (N,A)
29	Cu63	7293	(N,G), (N,P), (N,A)
	Cu65	7295	(N,2N)
40	Zr90	7400	(N,2N)
	Zr92	7402	(N,G)
	Zr94	7404	(N,G)
41	Nb93	7413	(N,N), (N,2N), (N,G)
42	Mo92	7422	(N,G)
	Mo98	7428	(N,G)
	Mo100	7420	(N,G)
47	Ag109	9479	(N,G)
49	In115	7495	(N,N)M, (N,G)
50	Sn120	7500	(N,G)
	Sn122	7502	(N,G)
	Sn124	7504	(N,G)
53	I127	7537	(N,2N), (N,G)
57	La139	7579	(N,G)
66	Dy164	7664	(N,G)

73	Ta181	7731	(N,G)
79	Au197	7797	(N,2N), (N,3N), (N,G)
90	Th232	7902	(N,2N), (N,G)
91	Pa231	7911	(N,G)
91	Pa233	7913	(N,2N)
92	U233	7923	(N,2N)
92	U238	7928	(N,G)
95	Am241	7951	(N,G)

Key: A = Alpha
D = Deuteron
F = Fission
G = Gamma Ray
M = Metastable state
N = Neutron
P = Proton
T = Triton

Table 5. Gas Production File Data in VITAMIN-E.

Z	Material	ENDF/B mat no.	Gas production reaction types
3	Li6	5303	H-1, H-2, H-3, He-4
	Li7	5272	H-2, H-3, He-4
4	Be9	5304	H-1, H-2, H-3, He-4
5	B10	5305	H-1, H-2, He-4
	B11	5160	H-1, H-3, He-4
6	C	5306	H-1, He-4
7	N14	5275	H-1, He-4
9	F	5309	H-1, He-4
13	Al27	5313	H-1, He-4
14	Si28	5314	H-1, He-4
22	Ti	5322	H-1, He-4
23	V	5323	H-1, He-4
24	Cr	5324	H-1, H-2, H-3, He-3, He-4
25	Mn55	5325	H-1, He-4
26	Fe	5326	H-1, He-4
27	Co59	5327	H-1, He-4
28	Ni	5328	H-1, H-2, He-4
29	Cu	5329	H-1, He-4

Table 6. Neutron group energy boundaries and weight function integrals for VITAMIN-E.

Fine group	Neutron Upper energy	Neutron Upper lethargy	Weight function Group integral
1	1.964029E+07	-6.749979E-01	7.966001E-03
2	1.733250E+07	-5.499981E-01	1.593000E-03
3	1.690459E+07	-5.250002E-01	1.593000E-03
4	1.648720E+07	-4.999993E-01	3.187000E-03
5	1.568310E+07	-4.499986E-01	3.124000E-01
6	1.491830E+07	-4.000036E-01	2.135000E+00
7	1.454990E+07	-3.749990E-01	6.420000E+00
8	1.419070E+07	-3.500018E-01	8.325000E+00
9	1.384030E+07	-3.249996E-01	4.787000E+00
10	1.349860E+07	-3.000009E-01	1.409000E+00
11	1.252320E+07	-2.249978E-01	9.204999E-04
12	1.221400E+07	-1.999978E-01	1.841000E-03
13	1.161830E+07	-1.499963E-01	1.841000E-03
14	1.105170E+07	-9.999919E-02	1.841000E-03
15	1.051270E+07	-4.999897E-02	1.841000E-03
16	1.000000E+07	0.0	2.120000E-03
17	9.512290E+06	5.000032E-02	2.762000E-03
18	9.048370E+06	1.000003E-01	3.539000E-03
19	8.607080E+06	1.499996E-01	4.462998E-03
20	8.187310E+06	1.999995E-01	5.546000E-03
21	7.788010E+06	2.499991E-01	6.793998E-03
22	7.408180E+06	2.999996E-01	8.211002E-03
23	7.046880E+06	3.499999E-01	9.796999E-03
24	6.703200E+06	3.999996E-01	3.647000E-03
25	6.592380E+06	4.166700E-01	7.899001E-03
26	6.376280E+06	4.499997E-01	1.345000E-02
27	6.065310E+06	4.999991E-01	1.549000E-02
28	5.769500E+06	5.499992E-01	1.766000E-02
29	5.488120E+06	5.999990E-01	1.993000E-02
30	5.220460E+06	6.499994E-01	2.228000E-02
31	4.965850E+06	7.000006E-01	2.468000E-02
32	4.723670E+06	7.499989E-01	2.711000E-02
33	4.493290E+06	7.999997E-01	6.143000E-02
34	4.065700E+06	8.999990E-01	7.068998E-02
35	3.678790E+06	1.000001E+00	7.911998E-02
36	3.328710E+06	1.099999E+00	4.237000E-02
37	3.166370E+06	1.149999E+00	4.401000E-02
38	3.011940E+06	1.200000E+00	4.547000E-02
39	2.865050E+06	1.249999E+00	4.673000E-02
40	2.725320E+06	1.299998E+00	4.779000E-02
41	2.592400E+06	1.350000E+00	4.864000E-02
42	2.465970E+06	1.400000E+00	3.276000E-02
43	2.385210E+06	1.433297E+00	8.301999E-03
44	2.365250E+06	1.441701E+00	8.214999E-03
45	2.345700E+06	1.450001E+00	1.657000E-02
46	2.306860E+06	1.466698E+00	3.315000E-02
47	2.231300E+06	1.500000E+00	4.994000E-02

Table 6. (Continued)

Fine group	Neutron Upper energy	Neutron Upper lethargy	Weight function Group integral
48	2.122480E+06	1.549999E+00	5.000000E-02
49	2.018970E+06	1.599998E+00	5.000000E-02
50	1.920500E+06	1.649999E+00	5.000000E-02
51	1.826840E+06	1.699997E+00	5.000000E-02
52	1.737740E+06	1.749999E+00	5.000000E-02
53	1.652990E+06	1.799999E+00	5.000000E-02
54	1.572370E+06	1.850000E+00	5.000000E-02
55	1.495690E+06	1.899997E+00	5.000000E-02
56	1.422740E+06	1.950000E+00	5.000000E-02
57	1.353350E+06	2.000002E+00	5.000000E-02
58	1.287350E+06	2.049999E+00	5.000000E-02
59	1.224560E+06	2.100003E+00	5.000000E-02
60	1.164840E+06	2.150001E+00	5.000000E-02
61	1.108030E+06	2.200001E+00	1.000000E-01
62	1.002590E+06	2.299998E+00	4.170000E-02
63	9.616400E+05	2.341700E+00	5.830000E-02
64	9.071800E+05	2.400000E+00	5.000000E-02
65	8.629360E+05	2.450000E+00	5.000000E-02
66	8.208500E+05	2.500000E+00	5.000000E-02
67	7.808170E+05	2.549999E+00	5.000000E-02
68	7.427360E+05	2.599999E+00	5.000000E-02
69	7.065120E+05	2.650000E+00	5.000000E-02
70	6.720550E+05	2.700000E+00	5.000000E-02
71	6.392790E+05	2.749999E+00	5.000000E-02
72	6.081010E+05	2.799998E+00	5.000000E-02
73	5.784440E+05	2.849998E+00	5.000000E-02
74	5.502320E+05	2.900000E+00	5.000000E-02
75	5.233970E+05	2.949999E+00	5.000000E-02
76	4.978710E+05	2.999998E+00	1.000000E-01
77	4.504920E+05	3.099999E+00	1.000000E-01
78	4.076220E+05	3.200000E+00	5.000000E-02
79	3.877420E+05	3.250000E+00	5.000000E-02
80	3.688320E+05	3.299998E+00	1.000000E-01
81	3.337330E+05	3.399999E+00	1.000000E-01
82	3.019740E+05	3.499999E+00	1.160000E-02
83	2.984910E+05	3.511600E+00	4.299000E-03
84	2.972110E+05	3.515898E+00	9.098999E-03
85	2.945180E+05	3.525000E+00	2.500000E-02
86	2.872470E+05	3.549997E+00	5.000000E-02
87	2.732370E+05	3.600000E+00	1.000000E-01
88	2.472350E+05	3.700001E+00	5.000000E-02
89	2.351770E+05	3.750002E+00	5.000000E-02
90	2.237080E+05	3.799998E+00	5.000000E-02
91	2.127970E+05	3.850001E+00	5.000000E-02
92	2.024190E+05	3.900001E+00	5.000000E-02
93	1.925470E+05	3.950000E+00	5.000000E-02
94	1.831560E+05	4.000002E+00	5.000000E-02
95	1.742240E+05	4.049998E+00	5.000000E-02

Table 6. (Continued)

Fine group	Neutron Upper energy	Neutron Upper lethargy	Weight function Group integral
96	1.657270E+05	4.099998E+00	5.000000E-02
97	1.576440E+05	4.150001E+00	5.000000E-02
98	1.499560E+05	4.199998E+00	5.000000E-02
99	1.426420E+05	4.250002E+00	5.000000E-02
100	1.356860E+05	4.299997E+00	5.000000E-02
101	1.290680E+05	4.350000E+00	5.000000E-02
102	1.227730E+05	4.400003E+00	5.000000E-02
103	1.167860E+05	4.449997E+00	5.000000E-02
104	1.110900E+05	4.499999E+00	1.250000E-01
105	9.803656E+04	4.625000E+00	1.250000E-01
106	8.651687E+04	4.750001E+00	4.750000E-02
107	8.250337E+04	4.797501E+00	3.710000E-02
108	7.949869E+04	4.834599E+00	9.909999E-02
109	7.199806E+04	4.933702E+00	6.629997E-02
110	6.737937E+04	5.000001E+00	1.750000E-01
111	5.656220E+04	5.174999E+00	7.499999E-02
112	5.247520E+04	5.250000E+00	1.250000E-01
113	4.630920E+04	5.375000E+00	1.250000E-01
114	4.086770E+04	5.500000E+00	1.750000E-01
115	3.430670E+04	5.674999E+00	7.499999E-02
116	3.182780E+04	5.750000E+00	1.104000E-01
117	2.850110E+04	5.860397E+00	5.410000E-02
118	2.700010E+04	5.914499E+00	3.550000E-02
119	2.605840E+04	5.950000E+00	5.000000E-02
120	2.478750E+04	6.000000E+00	2.500000E-02
121	2.417550E+04	6.025001E+00	2.500000E-02
122	2.357860E+04	6.050001E+00	7.499999E-02
123	2.187490E+04	6.125000E+00	1.250000E-01
124	1.930450E+04	6.250002E+00	2.500000E-01
125	1.503440E+04	6.499999E+00	2.500000E-01
126	1.170880E+04	6.750000E+00	1.000000E-01
127	1.059460E+04	6.849997E+00	1.500000E-01
128	9.118816E+03	7.000001E+00	2.500000E-01
129	7.101738E+03	7.250001E+00	2.500000E-01
130	5.530840E+03	7.500001E+00	2.500000E-01
131	4.307418E+03	7.750002E+00	1.500000E-01
132	3.707440E+03	7.899999E+00	1.000000E-01
133	3.354630E+03	7.999999E+00	1.000000E-01
134	3.035390E+03	8.100000E+00	1.000000E-01
135	2.746540E+03	8.199999E+00	5.000000E-02
136	2.612590E+03	8.249999E+00	5.000000E-02
137	2.485170E+03	8.299999E+00	1.000000E-01
138	2.248670E+03	8.400002E+00	1.000000E-01
139	2.034680E+03	8.500002E+00	2.500000E-01
140	1.584610E+03	8.750002E+00	2.500000E-01
141	1.234100E+03	8.999999E+00	2.500000E-01
142	9.611160E+02	9.250001E+00	2.500000E-01
143	7.485178E+02	9.500001E+00	2.500000E-01

Table 6. (Continued)

Fine group	Neutron Upper energy	Neutron Upper lethargy	Weight function Group integral
144	5.829468E+02	9.750000E+00	2.500000E-01
145	4.539988E+02	1.000000E+01	2.500000E-01
146	3.535750E+02	1.025000E+01	2.500000E-01
147	2.753638E+02	1.050000E+01	2.500000E-01
148	2.144540E+02	1.075000E+01	2.500000E-01
149	1.670170E+02	1.100000E+01	2.500000E-01
150	1.300730E+02	1.125000E+01	2.500000E-01
151	1.013010E+02	1.150000E+01	2.500000E-01
152	7.889319E+01	1.175000E+01	2.500000E-01
153	6.144209E+01	1.200000E+01	2.500000E-01
154	4.785120E+01	1.225000E+01	2.500000E-01
155	3.726649E+01	1.250000E+01	2.500000E-01
156	2.902319E+01	1.275000E+01	2.500000E-01
157	2.260329E+01	1.300000E+01	2.500000E-01
158	1.760349E+01	1.325000E+01	2.500000E-01
159	1.370960E+01	1.350000E+01	2.500000E-01
160	1.067700E+01	1.375000E+01	2.500000E-01
161	8.315289E+00	1.400000E+01	2.500000E-01
162	6.475949E+00	1.425000E+01	2.500000E-01
163	5.043480E+00	1.450000E+01	2.500000E-01
164	3.927859E+00	1.475000E+01	2.500000E-01
165	3.059019E+00	1.500000E+01	2.500000E-01
166	2.382370E+00	1.525000E+01	2.500000E-01
167	1.855390E+00	1.550000E+01	2.500000E-01
168	1.444980E+00	1.575000E+01	2.500000E-01
169	1.125350E+00	1.600000E+01	2.500000E-01
170	8.764250E-01	1.625000E+01	2.500000E-01
171	6.825600E-01	1.650000E+01	2.500000E-01
172	5.315779E-01	1.675000E+01	2.500000E-01
173	4.139940E-01	1.700000E+01	3.578000E+03
174	1.000010E-01	1.842067E+01	3.161000E+04
Lower bound	1.000010E-05	2.763101E+01	

Table 7. Photon group definitions and weighting fluxes

Fine group	Photon group Upper energy	Weighting function Group integral
1	2.000000E+07	6.000000E+00
2	1.400000E+07	2.000000E+00
3	1.200000E+07	2.000000E+00
4	1.000000E+07	2.000000E+00
5	8.000000E+06	5.000000E-01
6	7.500000E+06	5.000000E-01
7	7.000000E+06	5.000000E-01
8	6.500000E+06	5.000000E-01
9	6.000000E+06	5.000000E-01
10	5.500000E+06	5.000000E-01
11	5.000000E+06	5.000000E-01
12	4.500000E+06	5.000000E-01
13	4.000000E+06	5.000000E-01
14	3.500000E+06	5.000000E-01
15	3.000000E+06	5.000000E-01
16	2.500000E+06	5.000000E-01
17	2.000000E+06	3.400000E-01
18	1.660000E+06	1.600000E-01
19	1.500000E+06	1.700000E-01
20	1.330000E+06	3.300000E-01
21	1.000000E+06	2.000000E-01
22	8.000000E+05	9.999996E-02
23	7.000000E+05	9.999996E-02
24	6.000000E+05	8.800000E-02
25	5.120000E+05	2.000000E-03
26	5.100000E+05	6.000000E-02
27	4.500000E+05	5.000000E-02
28	4.000000E+05	9.999996E-02
29	3.000000E+05	9.999996E-02
30	2.000000E+05	5.000000E-02
31	1.500000E+05	5.000000E-02
32	1.000000E+05	2.500000E-02
33	7.500000E+04	4.999999E-03
34	7.000000E+04	9.999998E-03
35	6.000000E+04	1.500000E-02
36	4.500000E+04	1.500000E-02
37	3.000000E+04	9.999998E-03
38	2.000000E+04	9.999998E-03
Lower bound	1.000000E+04	

Table 8. Special reaction identifiers used for VITAMIN-E.

Identification number (MT)	Description
254	Transport cross section
255	Elastic removal cross section
285	Potential scattering cross section
1001	Current-weighted total cross section
1018	Fission spectrum for 1 MeV
1033	Deviation of MT 1001 from the total cross section
1099	VITAMIN-E weighting function