CONF-871101--33

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
L. M. Petrie
R. T. Primm, III**
M. W. Waddell
C. C. Webster
R. M. Westfall
R. Q. Wright



CONF-871101--33

DE87 011675



COMPUTING AND TELECOMMUNICATIONS DIVISION
at
Oak Ridge National Laboratory***
Post Office Box X
Oak Ridge, Tennessee 37831



Proposed paper for presentation at the
Annual Meeting of the
American Nuclear Society
Los Angeles, California
November 15-19, 1987





*Work funded by U. S. Department of Energy Office of Basic Energy Sciences, Washington, DC 20545.



**ORNL Engineering Physics and Mathematics Division

***Operated by Martin Marietta Energy Systems, Inc., under contract DE-AC05-84OR21400 with the U.S. Department of Energy.

# 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
L. M. Petrie
R. T. Primm, III
M. W. Waddell
C. C. Webster
R. M. Westfall
R. Q. Wright

Research and development for an Advanced Neutron Source (ANS) Reactor - formerly called the Center for Neutron Research - is being funded by the U. S. Department of Energy. The reactor is to provide the world's most intense steady-state source of low-energy neutrons for a national experimental user facility.[1]

In this work, pseudo-problem-independent, multigroup, $P_3$ neutron, $P_0$-$P_3$ secondary gamma-ray production (SGRP), and $P_6$ gamma-ray interaction (GRI) cross-section libraries were generated to support design work on the ANS Reactor. The libraries, designated ANSL-V,[a] are data bases in AMPX[2] master format for the subsequent generation of problem-dependent cross sections for use with codes such as KENO,[3] ANISN,[4] XSDRNPM,[5] VENTURE,[6] DOT,[7] and MORSE.[8] Materials included in ANSL-V neutron and SGRP libraries are listed in Table 1.

ANSL-V was generated with AMPX[2] and NJOY[9] modules and with evaluated data from various sources. Hierarchy of such sources for processing neutron and SGRP data was: (1) ENDF/B-V General Purpose Library,[10] (2) LENDL Library,[11,12] (3) ENDF/B-V Actinide (special purpose) and Fission Product Files, (4) Japanese Evaluated Nuclear Data File.[13,14]

Experiences gained with CSRL,[15-17] CSRL-V,[18,19] and Hansen-Roach[20] libraries influenced selection of group structures and point-to-multigroup weighting functions used to generate ANSL-V. As was done in the CSRL and CSRL-V libraries, consideration was given in the 3 eV to 20 MeV range to resonance structures of prominent nuclei, thresholds of important reactions, and various fission spectra. Groups in the 0.104 to 0.765 eV range are for ANS hot source design needs; groups in the 4.45 x $10^{-3}$ to $10^{-5}$ eV range are for cold source design needs.[21] The ANSL-V neutron fine-group structure (99 groups) includes 29 thermal groups in the range 1.00-5 to 3.00 eV; the broad-group structure (39 groups) includes 25 thermal groups in the same range.

Thermal bound data for H-1, H-2, Be-9, and graphite were processed from ENDF/B thermal scattering law data and included with appropriate epithermal data. Special data were included to represent kinematics of very low temperature scattering for ortho and para forms of hydrogen and deuterium. Work of Cohen and Feynman[22] regarding liquid He-4 scattering kernels was modelled to provide data to compensate for lack of scattering data in the ENDF/B-V He-4 evaluation. Delayed fission gamma-ray multiplicities were included in $U^{235}$ and $U^{238}$ SGRP data sets.

Cross sections for the ANSL-V GRI Library (44 groups) were processed from the Radiation Shielding Information Center (RSIC) DLC-99/HUGO photon interaction data collection.[23] Gamma-ray energy absorption coefficients (kerma factors) were included in each GRI data set.

ANSL-V data were checked for first-order inconsistencies with the RADE module of AMPX. Selected ANSL-V data were plotted and compared against appropriate ENDF/B-V File 3 data. Validity of selected data sets from the fine- and broad-group neutron libraries was satisfactorily tested in performance parameter calculations for the TRX-1, BAPL-1, and ZEEP-1 thermal reactor benchmarks. These results are presented in a companion paper.[24]

The ANSL-V libraries are available on magnetic tape from RSIC.[b]

---

a. ANSL-V is an acronym for Advanced Neutron Source Cross-Section Libraries based on ENDF/B-V.

b. RSIC's address is as follows: Radiation Shielding Information Center, Oak Ridge National Laboratory, Building 6025, Room 15W, MS 362, P. O. Box X, Oak Ridge, Tennessee 37831-6362.

Table 1. ANSL-V Neutron and Secondary Gamma-Ray Production Materials

Neutron Libraries

| Nuclide | ENDF/B-V MAT/MOD[a] | Re. Res. Data[b] | Unr. Res. Data[c] | Weight Fctn.[d] | SGRP Lib. MAT/ MOD No.[e] | Remarks |
|---|---|---|---|---|---|---|
| H-1 | 1301/1 | N | N | A | 1301/1 | Free gas |
| H-1 | 1301/1 | N | N | A | 1301/1 | MAT 1002 $H_2O$ bound thermal @ 296,350, 400,450,500,600, 800, and 1000 K. |
| H-1 | 1301/1 | N | N | A | 1301/1 | Ortho |
| H-1 | 1301/1 | N | N | A | 1301/1 | Para |
| H-2 | 1302/2 | N | N | A | 1302/2 | MAT 1004 $D_2O$ bound thermal @ 296,350, 400,450,500,600, 800, and 1000 K. |
| H-2 | 1302/2 | N | N | A | 1302/2 | Ortho |
| H-2 | 1302/2 | N | N | A | 1302/2 | Para |
| H-3 | 1169/2 | N | N | A | | |
| He-3 | 1146/1 | N | N | A | | |
| He-4 | 1270/0 | N | N | A | | w/special scatt. data. |
| Be-9 | 1304/2 | N | N | A | 1304/2 | MAT 1064 bound thermal @ 296,400,500,600,700, 800,1000, and 1200 K. |
| B-10 | 1305/1 | N | N | A | 1305/1 | |
| B-11 | 1160/1 | N | N | A | 7811[f] | |
| C-12 | 1306/2 | N | N | A | 1306/2 | |
| Graphite | 1306/2 | N | N | A | 1306/2 | MAT 1065 graphite thermal scatt. kernel @ 296,500, 800,1200, and 2000 K. |
| N-14 | 1275/2 | N | N | A | 1275/2 | |
| N-15 | 1307/1 | N | N | A | 1307/1 | |
| O-16 | 1276/2 | N | N | A | 1276/2 | |
| Na-23 | 1311/3 | Y | N | B | 1311/3 | |
| Mg | 1312/1 | N | N | B | 1312/1 | |
| Al-27 | 1313/1 | N | N | B | 1313/1 | |
| Al-27 | 1313/1 | N | N | C | 1313/1 | |
| Al-27 | 4313[g] | Y | N | B | 1313/1 | |
| Si | 1314/3 | N | N | B | 1314/3 | |
| Si | 1314/3 | N | N | C | 1314/3 | |
| K | 1150/1 | N | N | B | 1150/1 | |
| Ti | 1322/1 | N | N | B | 1322/1 | |
| V | 1323/1 | N | N | B | 1323/1 | |
| Cr | 1324/2 | Y | N | B | 1324/2 | |
| Mn-55 | 1325/2 | Y | N | B | 1325/2 | |
| Fe | 1326/3 | Y | N | B | 1326/3 | |
| Co-59 | 1327/3 | Y | N | B | 1327/3 | |

Table 1. (Continued)

Neutron Libraries

| Nuclide | ENDF/B-V MAT/MOD[a] | Re. Res. Data[b] | Unr. Res. Data[c] | Weight Fctn.[d] | SGRP Lib. MAT/ MOD No.[e] | Remarks |
|---|---|---|---|---|---|---|
| Ni | 1328/2 | Y | N | B | 1328/2 | |
| Cu | 1329/1 | Y | N | B | 1329/1 | |
| Kr-82 | 1332/1 | Y | N | C | | Nucl. in fis. prod. chain |
| Kr-83 | 1333/1 | Y | N | C | | Nucl. in fis. prod. chain |
| Zr | 1340/2 | Y | N | B | 7841[f] | |
| Zr-90 | 1385/2 | Y | N | C | | |
| Zr-91 | 1386/2 | Y | N | C | | |
| Zr-92 | 1387/2 | Y | N | C | | |
| Zr-93 | 9232/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Zr-94 | 1388/2 | Y | N | C | | |
| Zr-96 | 1389/2 | Y | N | C | | |
| Mo | 1321/1 | Y | Y | B | 1321/1 | |
| Mo-97 | 9284/1[i] | Y | N | C | | Nucl. in fis. prod. chain |
| Tc-99 | 1308/2 | Y | Y | C | | Nucl. in fis. prod. chain |
| Ru-101 | 9330/1[i] | Y | N | C | | Nucl. in fis. prod. chain |
| Ru-103 | 9332/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Rh-103 | 1310/1 | Y | N | C | | Nucl. in fis. prod. chain |
| Rh-105 | 9355/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Cd | 1281/1 | N | N | B | 7847[f] | |
| Cd-113 | 1318/1 | Y | N | C | | |
| Ag-107 | 1407/2 | Y | N | B | 7845[f] | |
| Ag-107 | 1407/2 | Y | N | C | 7845[f] | |
| Ag-109 | 1409/2 | Y | N | B | 7846[f] | |
| Ag-109 | 1409/2 | Y | N | C | 7846[f] | Nucl. in fis. prod. chain |
| I-135 | 9618/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Xe-131 | 1351/1 | Y | N | C | | Nucl. in fis. prod. chain |
| Xe-133 | 9643/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Xe-135 | 1294/1 | N | N | C | | Nucl. in fis. prod. chain |
| Cs-133 | 1355/1 | Y | N | C | | Nucl. in fis. prod. chain |
| Cs-134 | 4663[g] | N | N | C | | Nucl. in fis. prod. chain |
| Cs-135 | 9665/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Ce-141 | 9725/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Pr-143 | 9745/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Nd-143 | 9764/1[i] | Y | N | C | | Nucl. in fis. prod. chain |
| Nd-145 | 9766/1[i] | Y | N | C | | Nucl. in fis. prod. chain |
| Nd-147 | 4768/1[g] | N | N | C | | Nucl. in fis. prod. chain |
| Pm-147 | 9783/1[i] | Y | N | C | | Nucl. in fis. prod. chain |
| Pm-148 | 9784/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Pm-148m | 9785/1[i] | Y | N | C | | Nucl. in fis. prod. chain |
| Pm-149 | 9786/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Sm-149 | 1319/1 | Y | N | C | | Nucl. in fis. prod. chain |
| Sm-150 | 9809/1[i] | Y | N | C | | Nucl. in fis. prod. chain |
| Sm-151 | 9810/1[i] | Y | N | C | | Nucl. in fis. prod. chain |
| Sm-152 | 9811/1[i] | Y | N | C | | Nucl. in fis. prod. chain |

Table 1. (Continued)

Neutron Libraries

| Nuclide | ENDF/B-V MAT/MOD[a] | Re. Res. Data[b] | Unr. Res. Data[c] | Weight Fctn.[d] | SGRP Lib. MAT/ MOD No.[e] | Remarks |
|---|---|---|---|---|---|---|
| Sm-153 | 9812/1[i] | N | N | C | | Nucl. in fis. prod. chain |
| Eu | 9463/1[g] | Y | Y | B | 7852[f] | |
| Eu | 9463/1[g] | Y | Y | C | 7852[f] | |
| Eu-151 | 1357/1 | Y | Y | C | 1357/1 | |
| Eu-152 | 4292/2[g] | Y | Y | C | | |
| Eu-153 | 1359/1 | Y | Y | C | 1359/1 | Nucl. in fis. prod. chain |
| Eu-154 | 4293/1[g] | Y | Y | C | | Nucl. in fis. prod. chain |
| Eu-155 | 4832/1[g] | N | N | C | | Nucl. in fis. prod. chain |
| Hf | 1372/1 | Y | Y | B | 8305[f] | |
| Hf | 1372/1 | Y | Y | C | 8305[f] | |
| Ta-181 | 1285/2 | Y | Y | B | 1285/2 | |
| Ta-182 | 1127/1 | Y | Y | B | | |
| Ir-191 | 7160[k] | N | N | B | | |
| Ir-193 | 7161[k] | N | N | B | | |
| Pb | 1382/2 | N | N | B | 1382/2 | |
| Th-232 | 1390/2 | Y | Y | C | 1390/2 | |
| U-233 | 1393/2 | Y | Y | C | 7866[f] | |
| U-234 | 1394/3 | Y | Y | C | 7867[f] | |
| U-235 | 1395/3 | Y | Y | C | 1395/3 | |
| U-235 | 1395/3 | Y | Y | C | 1395/3 | Includes delayed fission gamma-ray data |
| U-236 | 1396/3 | Y | Y | C | 7869[f] | |
| U-237 | 8237/1[j] | Y | Y | C | | |
| U-238 | 1398/3 | Y | Y | C | 1398/3 | |
| U-238 | 1398/3 | Y | Y | C | 1398/3 | Includes delayed fission gamma-ray data |
| Np-237 | 1337/2 | Y | Y | C | | |
| Np-238 | 8338/1[j] | Y | N | C | | |
| Np-239 | 2932/1[h] | N | N | C | | |
| Pu-238 | 1338/3 | Y | Y | C | 7875[f] | |
| Pu-239 | 1399/2 | Y | Y | C | 1399/2 | |
| Pu-240 | 1380/3 | Y | Y | C | 1380/3 | |
| Pu-241 | 1381/2 | Y | Y | C | 1381/2 | |
| Pu-242 | 1342/2 | Y | Y | C | 1342/2 | |
| Am-241 | 1361/2 | Y | Y | C | | |
| Am-243 | 1363/2 | Y | Y | C | | |
| Cm-242 | 8642/1[j] | Y | Y | C | | |
| Cm-244 | 1344/2 | Y | Y | C | | |
| Cm-245 | 1345/2 | Y | N | C | | |
| Cm-246 | 1346/1 | Y | Y | C | | |
| Cm-247 | 8647/1[j] | Y | Y | C | | |
| Cm-248 | 8648/1[j] | Y | Y | C | | |
| Bk-249 | 8749/1[j] | Y | Y | C | | |
| Cf-249 | 8849/1[j] | Y | Y | C | | |

Table 1. (Continued)

Neutron Libraries

| Nuclide | ENDF/B-V MAT/MOD[a] | Re. Res. Data[b] | Unr. Res. Data[c] | Weight Fctn.[d] | SGRP Lib. MAT/ MOD No.[e] | Remarks |
|---|---|---|---|---|---|---|
| Cf-250 | 8850/1[j] | Y | Y | C | | |
| Cf-251 | 8851/1[j] | Y | Y | C | | |
| Cf-252 | 8852/1[j] | Y | Y | C | | |
| Cf-253 | 8853/1[j] | Y | Y | C | | |
| 1/V | N/A | N | N | D | | |

a. Evaluations from the ENDF/B-V General Purpose Library unless noted otherwise.

b. "Y" denotes resolved resonance Nordheim data are included in the ANSL/V multigroup libraries for subsequent problem-dependent processing; "N" denotes that such data are not included in the libraries.

c. "Y" denotes unresolved resonance Bondarenko factor data are included in the multigroup libraries for subsequent problem-dependent processing; "N" denotes that such data are not included in the libraries.

d. Weighting functions used in generation of ANSL-V-neutron data:

| Weight Fctn. Designation | Weight Function |
|---|---|
| A | $10^{-5}$ <= $E_n$ <= 0.1265 eV, Maxwellian spectrum with a temperature of 300 K; 0.1265 eV < $E_n$ <= 1.4 MeV, 1/(E*Sigma$_T$) spectrum; 1.4 < $E_n$ <= 20 MeV, fission spectrum with a temperature of $1.27 \times 10^6$ eV. The "Sigma$_T$" is the microscopic total cross sections for the respective materials. |
| B | $10^{-5}$ <= $E_n$ <= 0.1265 eV, Maxwellian spectrum with a temperature of 300K; 0.1265 eV < $E_n$ <= 0.75 MeV, 1/(E*Sigma$_T$) spectrum; 0.75 < $E_n$ <= 20 MeV, fission spectrum with a temperature of $1.27 \times 10^6$ eV. The "Sigma$_T$" is the microscopic total cross sections for the respective materials. |
| C | $10^{-5}$ <= $E_n$ <= 0.1265 eV, Maxwellian spectrum with a temperature of 300K; 0.1265 eV < $E_n$ <= 0.1 MeV, 1/(E*Sigma$_T$) spectrum; 0.1 < $E_n$ <= 20 MeV, fission spectrum with a temperature of $1.27 \times 10^6$ eV. The "Sigma$_T$" is the macroscopic total cross sections for a homogenized representation of the fuel region. |

D $10^{-5} <= E_n <= 0.1265$ eV, Maxwellian spectrum with a temperature of 300 K; $0.1265$ eV $< E_n <=$ 0.75 Mev, 1/E spectrum; $0.75 < E_n <= 20$ MeV, fission spectrum with a temperature of $1.27 \times 10^6$ eV.

e. Evaluations used to process ANSL-V secondary gamma-ray production data sets.
f. Evaluation from the LENDL library.[10]
g. Inhouse modification of ENDF/B-V evaluation.
h. Evaluation from the Japanese Evaluated Nuclear Data Library (JENDL-2) as subsequently modified. See Refs. 12 and 13.
i. ENDF/B-V Special Purpose Fission Product File.
j. ENDF/B-V Special Purpose Actinide File.
k. Evaluation from the LENDL library.[11]

## REFERENCES

1. F. C. Difilippo et al., "A Preliminary Reactor Design for the Center for Neutron Research," Nucl. Instr. & Meth., A249, 56-65 (1986).

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

3. L. M. Petrie and N. F. Cross, "KENO-IV An Improved Monte Carlo Criticality Program," ORNL-4938 (November 1975).

4. W. W. Engle, Jr., "A Users Manual for ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1693, Union Carbide Corporation, Nuclear Division (March 1967).

5. N. M. Greene and C. W. Craven, Jr., "XSDRN: A Discrete Ordinates Spectral Averaging Code," ORNL/TM-2500 (July 1969).

6. D. R. Vondy, T. B. Fowler, and G. W. Cunningham, "VENTURE: A Code Block for Solving Multigroup Neutronics Problems Applying the Finite-Difference Diffusion-Theory Approximation to Neutron Transport, Version II," ORNL-5062/R1 (November 1977).

7. W. A. Rhoades and F. R. Mynatt, "The DOT-III Two-Dimensional Transport Code," ORNL/TM-4280 (1973).

8. E. A. Straker et al., "The MORSE Code - A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code," ORNL/TM-4585 (1970).

9. R. E. MacFarlane, D. M. Muir, and R. M. Boicourt, "The NJOY Nuclear Data Processing System, Volume 1: User's Manual," LA-9303-M, Vol. 1 (ENDF-324) (May 1982).

10. R. Kinsey, Ed., "ENDF/B Summary Documentation," BNL-NCS-17541 (ENDF-201), 3rd ed., Brookhaven National Laboratory (1979).

11. "LENDL, Lawrence Livermore National Laboratory Evaluated Nuclear Data Library in ENDF/B-V Format," RSIC Data Library Collection DLC-120 (September 24, 1986).

12. R. J. Howerton, "The Lawrence Livermore Laboratory Evaluated Nuclear Data Library (ENDL) Translated into the ENDF/B Format," UCID-16727 (March 1975).

13. "JENDL2, Japanese Evaluated Neutron Cross Section Data in ENDF/B-IV Format," RSIC Data Library Collection documented for DLC-122 (December 31, 1986).

REFERENCES (CONTINUED)

14. T. Nakagawa, Ed., "Summary of JENDL-2 General Purpose File," Department of Physics, Tokai Research Establishment, JAERI-M-84-103 (1984).

15. W. E. Ford, III, R. M. Westfall, and C. C. Webster, "A 218-Neutron-Group Master Cross-Section Library for Criticality Safety Studies," Trans. Am. Nucl. Soc., 22, 290 (1975).

16. W. E. Ford, III, C. C. Webster, and R. M. Westfall, "A 218-Group Neutron Cross-Section Library in the AMPX Master Interface Format for Criticality Safety Studies," ORNL/CSD/TM-4 (July 1976).

17. "CSRL A 218-Group Neutron Cross-Section Reference Library in the AMPX Master Interface Format for Criticality Safety Studies," Radiation Shielding Information Center documentation for DLC-43, Oak Ridge National Laboratory (March 1978).

18. W. E. Ford, III, R. M. Westfall, B. R. Diggs, C. C. Webster, and J. R. Knight, "CSRL-V: An ENDF/B-V 227-Group Cross-Section Library for Criticality Safety Studies," Trans. Am. Nucl. Soc., 34, 331-333 (1980).

19. 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).

20. G. E. Hansen and W. H. Roach, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543 (1961).

21. C. D. West, R. T. Primm, III, et al., "Center for Neutron Research Project Status Report," ORNL/TM-10065 (September 1986).

22. M. Cohen and R. P. Feynman, Theory of Inelastic Scattering of Cold Neutrons from Liquid Helium, Phys. Rev., Vol. 107, No. 1, p. 13 (July 1, 1957).

23. 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) (December 1983).

24. R. Q. Wright, W. E. Ford, III, N. M. Greene, L. M. Petrie, R. T. Primm, III, and R. M. Westfall, "Benchmarking of the 99-Group ANSL-V Library," proposed paper for presentation at November 1987 American Nuclear Society meeting (June 3, 1987).