Application of FENDL Nuclear Data to ITER Neutronics and Shielding Analyses

M.E. Sawan

March 1996

UWFDM-1006

Presented at the 1996 Radiation Protection and Shielding Topical Meeting, North Falmouth, MA, April 21–25, 1996.

Application of FENDL Nuclear Data to ITER Neutronics and Shielding Analyses

M.E. Sawan

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

March 1996

UWFDM-1006

Presented at the 1996 Radiation Protection and Shielding Topical Meeting, North Falmouth, MA, April 21–25, 1996.

APPLICATION OF FENDL NUCLEAR DATA TO ITER NEUTRONICS AND SHIELDING ANALYSES

Mohamed E. Sawan
Fusion Technology Institute
University of Wisconsin-Madison
Madison, Wisconsin 53706 U.S.A.
(608)263-5093

ABSTRACT

Calculations have been performed for a benchmark problem representative of an ITER design to assess the impact of nuclear data evaluation, nuclear data processing, and multigroup structure on the flux and design relevant nuclear parameters. The differences in nuclear responses calculated with FENDL and ENDF/B-V using the same processing codes are smaller than the differences between results obtained using different processing codes. The results based on FENDL with the original 175n-42g group structure and a 46n-21g collapsed group structure are nearly identical.

I. INTRODUCTION

The IAEA Nuclear Data Section, in co-operation with several national nuclear data centers and research groups, has created the first version of an internationally accessible Fusion Evaluated Nuclear Data Library (FENDL-1).[1] The FENDL library has been selected to serve as a comprehensive source of processed and tested nuclear data for neutronics analyses in the Engineering Design Activities (EDA) of the International Thermonuclear Experimental Reactor (ITER) project and other fusion-related projects. Several experimental benchmarks have been developed to test and validate the FENDL library. Since the immediate need for the FENDL library is for ITER calculations, the IAEA Advisory Group Meeting on "Improved Evaluations and Integral Data Testing for FENDL" recommended that a calculational benchmark representative of the ITER design should be developed.[2] This benchmark problem can be used to assess the impact of nuclear data evaluation, nuclear data processing, and multigroup structure on the flux and design relevant nuclear parameters (heating, damage, gas production) in a fusion reactor relevant configuration. The detailed description and specifications of the neutronics and shielding calculational benchmark were documented in an IAEA Nuclear Data Section Report.[3] The radial build for the benchmark problem is given in Fig. 1. In this paper, results obtained for the calculational benchmark are presented and analyzed. Input from the fusion neutronics community to this calculational benchmark is highly encouraged.

II. CALCULATIONAL APPROACH

The discrete ordinates one-dimensional, diffusion-accelerated, neutral particle transport code ONEDANT[4] was used with the P_3S_8 approximation. The calculations used the FENDL/E-1.0 library[1] processed into the 175n-42g multigroup library FENDL/MG-1.0 by R. MacFarlane[5] using NJOY[6] and

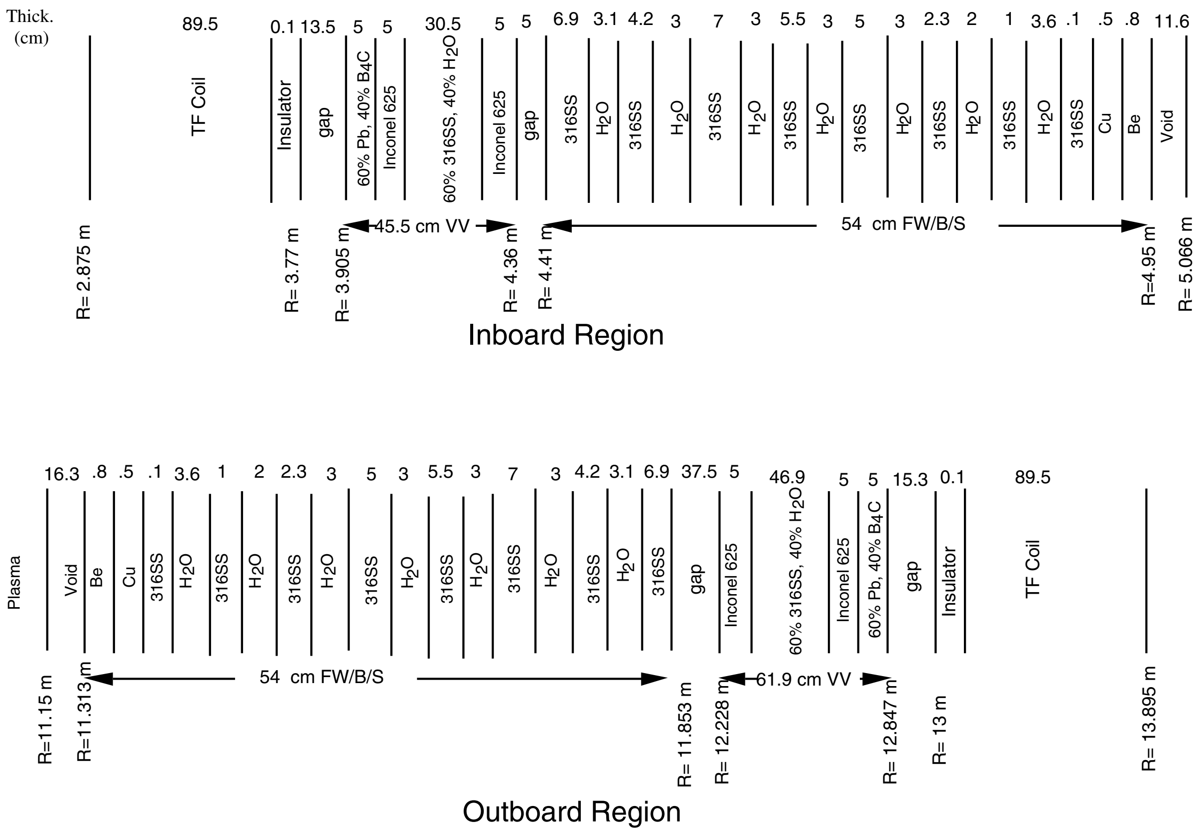

Fig.1. Radial build for the neutronics and shielding benchmark.

the VITAMIN-E weight function. We used the TRANSX[7] code to generate two working libraries from FENDL/MG-1.0 for use in the benchmark calculations. The libraries include the nuclear responses of interest such as nuclear heating, displacements per atom (dpa), tritium production, helium production, and hydrogen production. One library has the same group structure as FENDL/MG-1.0 (175n-42g) and the other is collapsed into a 46n-21g group structure using the VITAMIN-E weight function. Calculations were performed also using two other widely used libraries based on ENDF/B-V.[8] One library was generated using TRANSX from the MATXS5 library obtained by processing ENDF/B-V with NJOY. This library has a 30n-12g group structure. The other library has 46n-21g groups and is based on VITAMIN-E (processed from ENDF/B-V with the MINX[9] and AMPX[10] systems) for transport cross sections and KAOS/LIB[11] for nuclear responses.

III. NEUTRON FLUX

Table 1 gives the peak neutron flux values obtained in the different calculations. Comparing the results based on FENDL to those based on ENDF/B-V using the same processing codes (NJOY/TRANSX), the neutron fluxes differ by <4% at the front of the blanket. The difference increases as one moves away from the blanket reaching ~8% at the magnet. The difference between the results based on FENDL with 175 and 46 neutron groups is very small. The results are almost identical in the first wall (FW) and the differences are less than ~3% at the vacuum vessel (VV) and magnet. Comparing the results based on FENDL to those based on ENDF/B-V with the same group structure using different

Table 1. Peak Neutron Flux Values (n/cm^{2}s)

Evaluation	FENDL/E-1.0	FENDL/E-1.0	ENDF/B-V	ENDF/B-V
Processing Code	NJOY,TRANSX	NJOY,TRANSX	NJOY,TRANSX	MINX,AMPX, KAOS
Energy Groups	175n-42g	46n-21g	30n-12g	46n-21g
<u>INBOARD</u>				
First Wall Be	3.445×10^{14}	3.447×10^{14}	3.551×10^{14}	3.549×10^{14}
Cu	3.076×10^{14}	3.080×10^{14}	3.174×10^{14}	3.173×10^{14}
SS	2.918×10^{14}	2.922×10^{14}	3.011×10^{14}	3.009×10^{14}
Vacuum Vessel	9.775×10^{11}	9.445×10^{11}	9.276×10^{11}	1.060×10^{12}
Magnet	2.428×10^{9}	2.367×10^{9}	2.492×10^{9}	3.227×10^{9}
<u>OUTBOARD</u>				
First Wall Be	4.115×10^{14}	4.116×10^{14}	4.227×10^{14}	4.218×10^{14}
Cu	3.774×10^{14}	3.776×10^{14}	3.873×10^{14}	3.868×10^{14}
SS	3.619×10^{14}	3.620×10^{14}	3.711×10^{14}	3.705×10^{14}
Vacuum Vessel	1.352×10^{12}	1.311×10^{12}	1.293×10^{12}	1.467×10^{12}
Magnet	3.567×10^{8}	3.515×10^{8}	3.767×10^{8}	5.090×10^{8}

processing codes (MINX/AMPX), the ENDF/B-V data give higher results. The neutron fluxes differ by <3% at the front of the blanket with the difference reaching ~45% at the magnet.

IV. GAMMA FLUX

Comparing the results based on FENDL to those based on ENDF/B-V using the same processing codes (NJOY/TRANSX), the gamma fluxes differ by <6% at the front of the blanket. The difference increases as one moves away from the blanket reaching ~20% at the VV. The difference between results based on FENDL with 175n-42g and 46n-21g group structures is very small (<2%). Comparing the results based on FENDL to those based on ENDF/B-V with the same group structure using different processing codes (MINX/AMPX), the gamma fluxes differ by ~17% at the front of the blanket. The difference increases as one moves away from the blanket reaching ~60% at the magnet.

V. NUCLEAR HEATING

Table 2 gives the peak power density (neutron and gamma heating) in the FW, blanket, VV, and magnet. The FW power density results based on FENDL and ENDF/B-V using the same processing codes (NJOY/TRANSX) differ by ~5% in Be, ~6% in Cu and ~3% in SS. The differences are <3 for the VV and magnet. The results based on FENDL with 175n-42g and 46n-21g group structures are almost identical with differences <1%. The KAOS library based on ENDF/B-V gives higher peak power densities than the FENDL library processed by NJOY with the same group structure. The results are higher by ~1% in Be, ~21% in Cu and ~10% in SS of the FW. The results are higher by ~21% in the VV and ~54% in the magnet. The higher heating is partially attributed to the inclusion of the decay energy in the KAOS library and to the higher neutron and gamma fluxes calculated with the ENDF/B-V data processed by MINX/AMPX as discussed above.

The peak nuclear heating was calculated using two kerma factors provided in the KAOS library to assess the impact of neglecting the decay energy of short lived radionuclides. One of these includes the decay energy carried by gammas and betas emitted from decay of short lived (half life <1 day)

Table 2. Peak Power Density Values (W/cm^{3})

Evaluation	FENDL/E-1.0	FENDL/E-1.0	ENDF/B-V	ENDF/B-V
Processing Code	NJOY,TRANSX	NJOY,TRANSX	NJOY,TRANSX	MINX,AMPX, KAOS
Energy Groups	175n-42g	46n-21g	30n-12g	46n-21g
INBOARD				
First Wall Be	1.06×10^{1}	1.06×10^{1}	1.01×10^{1}	1.07×10^{1}
Cu	2.08×10^{1}	2.08×10^{1}	2.19×10^{1}	2.53×10^{1}
SS	1.82×10^{1}	1.82×10^{1}	1.87×10^{1}	2.01×10^{1}
Vacuum Vessel	3.32×10^{-2}	3.28×10^{-2}	3.18×10^{-2}	3.97×10^{-2}
Magnet	3.07×10^{-5}	3.05×10^{-5}	3.09×10^{-5}	4.46×10^{-5}
OUTBOARD				
First Wall Be	1.36×10^{1}	1.35×10^{1}	1.30×10^{1}	1.36×10^{1}
Cu	2.50×10^{1}	2.49×10^{1}	2.64×10^{1}	3.04×10^{1}
SS	2.21×10^{1}	2.21×10^{1}	2.28×10^{1}	2.44×10^{1}
Vacuum Vessel	4.47×10^{-2}	4.44×10^{-2}	4.32×10^{-2}	5.36×10^{-2}
Magnet	4.53×10^{-6}	4.57×10^{-6}	4.70×10^{-6}	7.04×10^{-6}

radionuclides. Since nuclear heating is dominated by gamma heating, the underestimate in total nuclear heating is small. Neglecting decay energy results in underestimating the power densities by ~2% in Be, ~10% in Cu and ~4% in SS of the FW. The underestimate is <1% in the Inconel VV and <2% in the TF coil. The differences between the peak power densities with the FENDL library processed by NJOY and the KAOS library based on ENDF/B-V are still large even with the decay energy neglected in the KAOS library. The differences are ~1% in Be, ~11% in Cu and ~6% in SS of the FW. The differences are ~20% in the VV and ~51% in the magnet. These differences are mostly attributed to large differences in the calculated neutron and gamma fluxes resulting from the MINX/AMPX processed transport cross sections.

VI. RADIATION DAMAGE

Table 3 gives the peak end-of-life atomic displacement damage (dpa) in the FW and VV. The FW dpa results based on FENDL and ENDF/B-V using the same processing codes (NJOY/TRANSX) differ by ~5% in Cu and ~2% in SS. The difference in the peak dpa in the Inconel VV is ~2%. The peak dpa results based on FENDL with 175n-42g and 46n-21g group structures are almost identical in the FW and VV (<1% difference). The KAOS library based on ENDF/B-V gives higher dpa values than the FENDL library processed by NJOY with the same group structure. The results are higher by ~6% in Cu FW, ~3% in SS FW and ~22% in the Inconel VV. These differences are partially contributed by the large differences in the calculated neutron fluxes.

VII. GAS PRODUCTION

The peak end-of-life gas production values (helium, hydrogen and tritium) in the FW and VV have been calculated. Table 4 gives the results for helium (He) production. The peak He production results based on FENDL with 175n-42g and 46n-21g group structures are almost identical in the FW and VV (<1% difference). The KAOS library based on ENDF/B-V gives higher He production values than the FENDL library processed by NJOY with the same group structure. The results for the FW are higher by ~5% in Be, ~11% in Cu, ~7% in SS. The peak Inconel VV He production is higher by ~52%.

Table 3. Peak End-of-Life dpa in FW and VV (dpa @ 3 FPY)

Evaluation	FENDL/E-1.0	FENDL/E-1.0	ENDF/B-V	ENDF/B-V
Processing Code	NJOY,TRANSX	NJOY,TRANSX	NJOY,TRANSX	MINX,AMPX, KAOS
Energy Groups	175n-42g	46n-21g	30n-12g	46n-21g
<u>INBOARD</u>				
First Wall Cu	28.2	28.1	29.2	29.7
SS	26.5	26.5	27.1	27.2
Vacuum Vessel	3.88×10^{-2}	3.83×10^{-2}	3.91×10^{-2}	4.69×10^{-2}
<u>OUTBOARD</u>				
First Wall Cu	38.5	38.4	39.8	40.4
SS	36.9	36.8	37.4	37.7
Vacuum Vessel	5.32×10^{-2}	5.27×10^{-2}	5.40×10^{-2}	6.43×10^{-2}

Table 5 gives the peak hydrogen (H) production results. The peak H production results based on FENDL with 175n-42g and 46n-21g group structures are almost identical in the FW and VV (<2% difference). The KAOS library based on ENDF/B-V gives H production values different than the FENDL library processed by NJOY with the same group structure. The results for the FW are different by ~5% in Be, ~4% in Cu, ~13% in SS. The peak Inconel VV H production is higher by ~20%.

Table 6 gives the peak tritium (T) production results. The peak T production results based on FENDL with 175n-42g and 46n-21g group structures are almost identical in the FW and VV (<2% difference). The KAOS library based on ENDF/B-V gives T production values different than the FENDL library processed by NJOY with the same group structure. The results for the the FW are different by ~5% in Be, ~3% in Cu, and ~72% in SS. The peak T production in the Inconel VV is higher by a factor of ~26.

Table 4. Peak End-of-Life Helium Production in FW and VV (appm @ 3 FPY)

Evaluation	FENDL/E-1.0	FENDL/E-1.0	ENDF/B-V
Processing Code	NJOY,TRANSX	NJOY,TRANSX	MINX,AMPX,KAOS
Energy Groups	175n-42g	46n-21g	46n-21g
<u>INBOARD</u>			
First Wall Be	1.30×10^{4}	1.30×10^{4}	1.37×10^{4}
Cu	7.02×10^{2}	7.07×10^{2}	7.80×10^{2}
SS	6.02×10^{2}	6.02×10^{2}	6.41×10^{2}
Vacuum Vessel	3.61×10^{-1}	3.57×10^{-1}	5.44×10^{-1}
<u>OUTBOARD</u>			
First Wall Be	1.73×10^{4}	1.73×10^{4}	1.81×10^{4}
Cu	9.72×10^{2}	9.70×10^{2}	1.08×10^{3}
SS	7.89×10^{2}	7.88×10^{2}	8.41×10^{2}
Vacuum Vessel	4.88×10^{-1}	4.83×10^{-1}	7.36×10^{-1}

Table 5. Peak End-of-Life Hydrogen Production in FW and VV (appm @ 3 FPY)

Evaluation ENDL/E-1.0	FENDL/E-1.0	ENDF/B-V	
Processing Code	NJOY,TRANSX	NJOY,TRANSX	MINX,AMPX,KAOS
Energy Groups	175n-42g	46n-21g	46n-21g
<u>INBOARD</u>			
First Wall Be	1.73×10^{2}	1.69×10^{2}	1.78×10^{2}
Cu	2.15×10^{3}	2.16×10^{3}	2.08×10^{3}
SS	1.99×10^{3}	1.98×10^{3}	1.75×10^{3}
Vacuum Vessel	2.33	2.32	2.78
<u>OUTBOARD</u>			
First Wall Be	2.27×10^{2}	2.23×10^{2}	2.35×10^{2}
Cu	2.99×10^{3}	3.00×10^{3}	2.88×10^{3}
SS	2.82×10^{3}	2.79×10^{3}	2.47×10^{3}
Vacuum Vessel	3.15	3.15	3.77

VIII. MAGNET RADIATION EFFECTS

Table 7 gives the peak end-of-life magnet radiation effects. This includes the fast neutron fluence (E>0.1 MeV), the absorbed dose in the organic insulator, and the displacement damage in the Cu stabilizer. The peak magnet radiation effects based on FENDL and ENDF/B-V using the same processing codes (NJOY/TRANSX) differ by less than ~8%. The peak magnet radiation effects based on FENDL with 175n-42g and 46n-21g group structures are almost identical (<1% difference). Comparing the results based on FENDL to those based on ENDF/B-V with the same group structure using different processing codes (MINX/AMPX), the peak end-of-life magnet radiation effects differ by as much as 50%. This can be attributed mostly to the large difference in the calculated neutron and gamma fluxes at the magnet.

Table 6. Peak End-of-Life Tritium Production in FW and VV (appm @ 3 FPY)

Evaluation ENDL/E-1.0	FENDL/E-1.0	ENDF/B-V	
Processing Code	NJOY,TRANSX	NJOY,TRANSX	MINX,AMPX,KAOS
Energy Groups	175n-42g	46n-21g	46n-21g
<u>INBOARD</u>			
First Wall Be	1.73×10^{2}	1.69×10^{2}	1.78×10^{2}
Cu	4.57	4.51	4.64
SS	0.92	0.92	1.57
Vacuum Vessel	1.74×10^{-5}	1.78×10^{-5}	4.69×10^{-4}
<u>OUTBOARD</u>			
First Wall Be	2.27×10^{2}	2.23×10^{2}	2.35×10^{2}
Cu	6.37	6.29	6.46
SS	1.30	1.30	2.24
Vacuum Vessel	2.36×10^{-5}	2.41×10^{-5}	6.35×10^{-4}

Table 7. Peak End-of-Life (@ 3 FPY) Magnet Radiation Effects

Evaluation	ENDL/E-1.0	FENDL/E-1.0	ENDF/B-V	ENDF/B-V
Processing Code	NJOY,TRANSX	NJOY,TRANSX	NJOY,TRANSX	MINX,AMPX, KAOS
Energy Groups	175n-42g	46n-21g	30n-12g	46n-21g
INBOARD				
Fast n fluence (E>0.1 MeV) (n/cm^2)	1.50×10^{17}	1.49×10^{17}	1.56×10^{17}	2.04×10^{17}
Insulator dose (eV/cm^3)	1.51×10^{22}	1.51×10^{22}	1.60×10^{22}	2.14×10^{22}
Cu dpa	9.65×10^{-5}	9.64×10^{-5}	1.03×10^{-4}	1.33×10^{-4}
OUTBOARD				
Fast n fluence (E>0.1 MeV) (n/cm^2)	2.21×10^{16}	2.21×10^{16}	2.36×10^{16}	3.22×10^{16}
Insulator dose (eV/cm^3)	2.23×10^{21}	2.25×10^{21}	2.43×10^{21}	3.37×10^{21}
Cu dpa	1.42×10^{-5}	1.44×10^{-5}	1.56×10^{-5}	2.09×10^{-5}

IX. SUMMARY AND CONCLUSIONS

The differences in nuclear responses calculated with FENDL and ENDF/B-V using the same processing codes (NJOY/TRANSX) are in general smaller than the differences between results obtained using different processing codes. Comparing the results based on FENDL to those based on ENDF/B-V using the same processing codes (NJOY/TRANSX), the flux and design relevant nuclear parameters differ by <8%. The only exceptions are a 20% difference in gamma flux at the VV and a 30% difference in nuclear heating in the B_4C/Pb back layer. The difference between flux and design relevant nuclear parameters based on FENDL with the original 175n-42g group structure and a 46n-21g collapsed group structure is very small (<3%).

The effect on nuclear heating of neglecting the decay energy associated with the short lived ($T_{1/2}$ < 1 day) radionuclides in the kerma factor is <10%. ENDF/B-V processed with MINX/AMPX/KAOS gives larger heating than FENDL processed by NJOY/TRANSX using the same group structure. The differences are large even when the effect of inclusion of decay energy is factored out. The differences can be as high as 25% in Cu, 40% in Inconel, and 60% in the magnet. Comparing the results based on FENDL to those based on ENDF/B-V with the same group structure using different processing codes (MINX/AMPX/KAOS), large differences are obtained in fluxes and design relevant nuclear parameters. In addition to large differences in nuclear heating, the most serious discrepancies are 45% in neutron flux at the magnet, 17% and 60% in gamma fluxes at the FW and magnet, respectively, 22% in dpa in Inconel VV, 52% in He production in Inconel VV, 72% in T production in SS, a factor of 26 in T production in Inconel VV, and up to 50% in peak magnet radiation effects.

Due to the observed large differences in nuclear heating and other nuclear responses calculated using data processed by NJOY/TRANSX and MINX/AMPX/KAOS, the processing methods used need to be investigated and evaluated. Integral experiments for nuclear responses of interest to the designers will be useful to validate the processed nuclear response data. The total gas production cross sections should be included in the FENDL library to eliminate errors that can result from leaving them to be calculated by the user by adding all possible partial cross sections provided in the library.

REFERENCES

1. A. Pashchenko, H. Wienke, S. Ganesan, and P. McLaughlin, "FENDL/E-1.0, Evaluated Nuclear Data Library of Neutron Nuclear Interaction Cross Sections and Photon Production Cross Sections and Photon-Atom Interaction Cross Sections for Fusion Applications," Report IAEA-NDS-128 Rev. 2, International Atomic Energy Agency (Nov. 1995).

2. IAEA Advisory Group Meeting on "Improved Evaluations and Integral Data Testing for FENDL," Garching, Germany, September 1994; Summary Report Prepared by S. Ganesan and Published as IAEA Nuclear Data Section Report INDC(NDS)-312 (December 1994).

3. M. Sawan, "FENDL Neutronics Benchmark: Specifications for the Calculational Neutronics and Shielding Benchmark," IAEA Nuclear Data Section Report INDC(NDS)-316 (December 1994).

4. R. D. O'Dell, F.W. Brinkley, Jr., and D. R. Marr, User's Manual for ONEDANT: A Code Package for One-Dimensional, Diffusion-Accelerated, Neutral Particle Transport, Los Alamos National Laboratory Report, LA-9184-M (1982).

5. R. MacFarlane, "FENDL/MG-1.0, Library of Multigroup Cross Sections in GENDF and MATXS Format for Neutron-Photon Transport Calculations," Summary Documentation by A. Pashchenko, H. Wienke and S. Ganesan, Report IAEA-NDS-129 Rev. 3, International Atomic Energy Agency (Nov. 1995).

6. R. MacFarlane, "The NJOY Nuclear Data Processing System, Version 91," Los Alamos National Laboratory Report, LA-12740-M (1994).

7. R. MacFarlane, "TRANSX 2: A Code for Interfacing MATX Cross Section Libraries to Nuclear Transport Codes," Los Alamos National Laboratory Report, LA-12312-MS (1992).

8. R. Kinsey, comp., "ENDF/B Summary Documentation," National Nuclear Data Center, Brookhaven National Laboratory Report BNL-NCS-17541 (ENDF-201), 3rd Ed., ENDF/B-V (1979).

9. C. Weisbin et al., "MINX, A Multigroup Interpretation of Nuclear Cross Sections, Los Alamos National Laboratory Report, LA-6486-MS (1976).

10. N. Greene et al., "AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," Oak Ridge National Laboratory Report, ORNL/TM-3706 (1976).

11. Y. Farawila, Y. Gohar and C. Maynard, "KAOS/LIB-V: A Library of Nuclear Response Functions Generated by KAOS-V Code from ENDF/B-V," Argonne National Laboratory Report ANL/FPP/TM-241 (April 1989).