Application of the Processed FENDL Libraries to the Calculational Benchmark and ITER Design

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UWFDM-1041

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Introduction

During the IAEA Advisory Group Meeting on "Improved Evaluations and Integral Data Testing for FENDL" held in Garching, Germany on 12-16 September 1994, the "Working Group II on Experimental and Calculational Benchmarks on Fusion Neutronics for FENDL Validation" recommended that a calculational benchmark representative of the ITER design should be developed [1]. This benchmark problem can be used to assess the impact of transport codes, 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 provided to the IAEA Nuclear Data Section and documented in the IAEA Nuclear Data Section Report INDC(NDS)-316 [2]. The radial build for the benchmark problem is given in Fig. 1. Results for the calculational benchmark were presented during the IAEA Advisory Group Meeting on "Completion of FENDL-1 and Start of FENDL-2" held in Del Mar, USA on 5-9 December 1995 [3].

It was recommended that the total gas production should be added in the processed multigroup 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. It was recommended also that the decay heat for radionuclides with half-lives less than one day be added to the processed kerma factors. R.E. MacFarlane of Los Alamos National Laboratory provided a new processed MATXS library in 175 neutron - 42 gamma energy groups that includes gas production cross sections and radionuclide production cross sections [4]. An algorithm for TRANSX [5] was provided to combine the radionuclide production cross sections with decay information to determine the decay heat contribution. The beta decay energy is assumed to be deposited locally and is added to the neutron kerma while the gamma production cross sections are modified to include the produced decay gamma thus allowing them to be transported before depositing their energy. We used the TRANSX code to generate a new multigroup working library using the new processed MATXS multigroup library and the algorithm for inclusion of decay heat. The new library was used to perform calculations for the calculational benchmark and the results were compared to those from the previous libraries.

In addition to the processed multigroup library, the FENDL/MC-1.0 sublibrary that contains pointwise cross sections in ACE format was processed from the FENDL/E-1.0 [6] library using NJOY [7] for use with the Monte Carlo code MCNP-4A [8]. A recent version of the ACE files was provided by R.E. MacFarlane of Los Alamos National Laboratory. This version includes gas production cross sections and damage energy cross sections which are important design relevant parameters. Reaction number 444 is used in tally multipliers to calculate atomic displacements. We used this recent version of the library to perform detailed three-dimensional neutronics and shielding calculations for the ITER reactor [9]. In this report, observations and comments based on our experience with the processed library for MCNP-4A are given.

Benchmark Calculational Approach

The discrete ordinates one-dimensional, diffusion-accelerated, neutral particle transport code ONEDANT was used with the P_3S_8 approximation. The calculations used the FENDL/E-1.0 [6] library processed into the 175n-42g multigroup library FENDL/MG by R. MacFarlane using NJOY [7] and the VITAMIN-E weight function. We used the TRANSX [5] 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, 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 [10]. 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 [11] and AMPX [12] systems) for transport cross sections and KAOS/LIB [13] for nuclear responses.

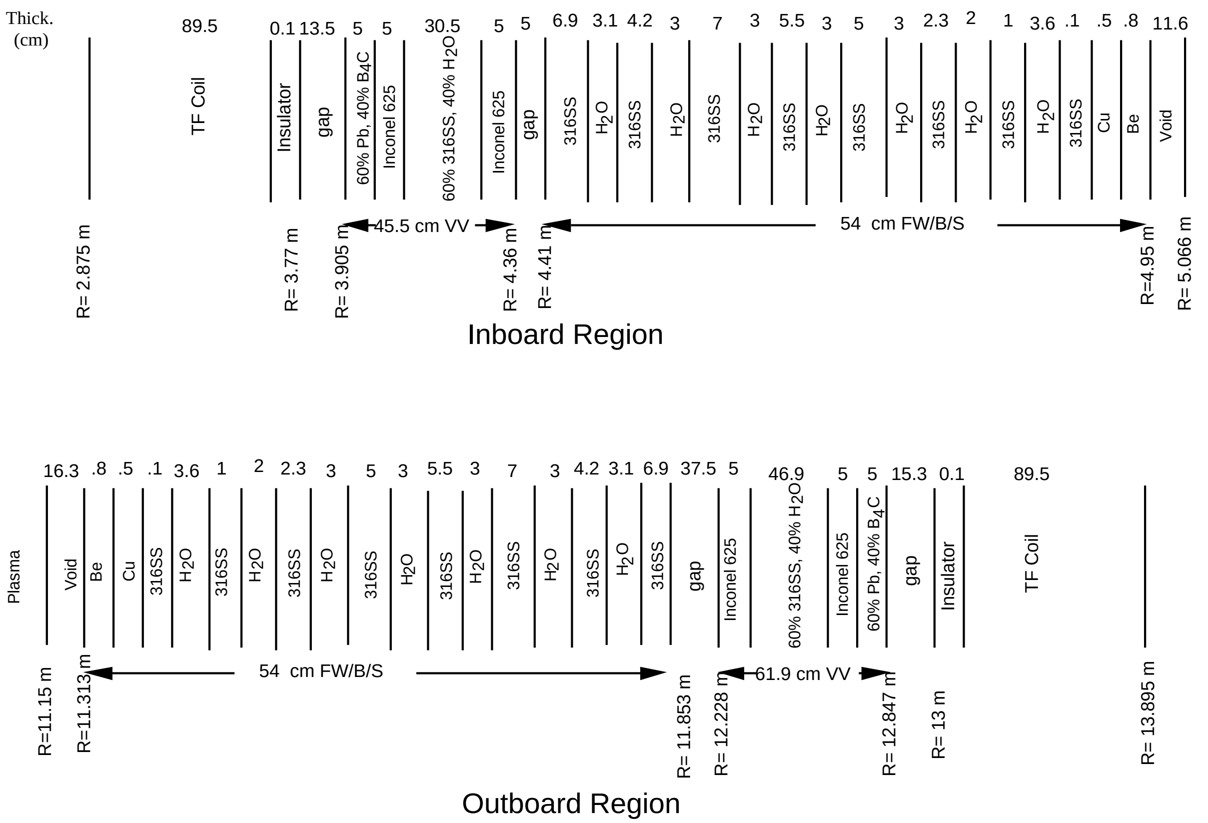

Fig. 1. Radial build for the neutronics and shielding benchmark.

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 processing codes (MINX/AMPX), the neutron fluxes differ by <3% at the front of the blanket. The difference increases as one moves away from the blanket reaching ~45% at the magnet. Table 2 shows the neutron flux results obtained using the new and old processed FENDL libraries. The neutron flux results obtained

Table 1. Peak Neutron Flux Values ($n/cm^2 \cdot s$)

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	3.438×10^{14}	3.447×10^{14}	3.551×10^{14}	3.549×10^{14}
Cu	3.075×10^{14}	3.080×10^{14}	3.174×10^{14}	3.173×10^{14}
SS	2.920×10^{14}	2.923×10^{14}	3.011×10^{14}	3.009×10^{14}
Vacuum Vessel	9.716×10^{11}	9.448×10^{11}	9.276×10^{11}	1.060×10^{12}
Magnet	2.426×10^{9}	2.368×10^{9}	2.492×10^{9}	3.227×10^{9}
<u>OUTBOARD</u>				
First Wall				
Be	4.109×10^{14}	4.117×10^{14}	4.227×10^{14}	4.218×10^{14}
Cu	3.772×10^{14}	3.776×10^{14}	3.873×10^{14}	3.868×10^{14}
SS	3.618×10^{14}	3.621×10^{14}	3.711×10^{14}	3.705×10^{14}
Vacuum Vessel	1.350×10^{12}	1.311×10^{12}	1.293×10^{12}	1.467×10^{12}
Magnet	3.584×10^{8}	3.516×10^{8}	3.767×10^{8}	5.090×10^{8}

Table 2. Peak Neutron Flux Values from New and Old Processed Libraries ($n/cm^2 \cdot s$)

Evaluation	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0
Processing Code	NJOY, TRANSX	NJOY, TRANSX	NJOY, TRANSX	NJOY, TRANSX
Energy Groups	175n-42g	46n-21g	175n-42g	46n-21g
New/Old	New	New	Old	Old
<u>INBOARD</u>				
First Wall				
Be	3.438×10^{14}	3.447×10^{14}	3.445×10^{14}	3.447×10^{14}
Cu	3.075×10^{14}	3.080×10^{14}	3.076×10^{14}	3.080×10^{14}
SS	2.920×10^{14}	2.923×10^{14}	2.918×10^{14}	2.922×10^{14}
Vacuum Vessel	9.746×10^{11}	9.448×10^{11}	9.775×10^{11}	9.445×10^{11}
Magnet	2.426×10^{9}	2.368×10^{9}	2.428×10^{9}	2.367×10^{9}
<u>OUTBOARD</u>				
First Wall				
Be	4.109×10^{14}	4.117×10^{14}	4.115×10^{14}	4.116×10^{14}
Cu	3.772×10^{14}	3.776×10^{14}	3.774×10^{14}	3.776×10^{14}
SS	3.618×10^{14}	3.621×10^{14}	3.619×10^{14}	3.620×10^{14}
Vacuum Vessel	1.350×10^{12}	1.311×10^{12}	1.352×10^{12}	1.311×10^{12}
Magnet	3.584×10^{8}	3.516×10^{8}	3.567×10^{8}	3.515×10^{8}

with the new processed library are nearly identical (<0.5% difference) to those obtained using the previous processed library.

Gamma Flux

Table 3 gives the peak gamma 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 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. Table 4 shows the gamma flux results obtained using the new and old processed FENDL libraries. The gamma flux values obtained using the new processed library are higher than those obtained using the previously processed library by up to ~4%. The higher gamma flux is related to the added decay gamma. The gamma production cross sections in the library were modified to include the produced decay gamma thus allowing them to be transported before depositing their energy.

Nuclear Heating

Table 5 gives the peak power density 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 <7%. The differences are <4 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 nearly identical power densities in Be compared to the FENDL library processed by NJOY with the same group structure. The results are higher by ~14% in Cu and ~9% in the SS of the FW and ~20% in the VV and ~50% 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.

Table 3. Peak Gamma Flux Values (g/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
INBOARD				
First Wall				
Be	3.03×10^{14}	3.00×10^{14}	2.80×10^{14}	3.37×10^{14}
Cu	2.93×10^{14}	2.86×10^{14}	2.67×10^{14}	3.23×10^{14}
SS	2.90×10^{14}	2.85×10^{14}	2.61×10^{14}	3.15×10^{14}
Vacuum Vessel	5.68×10^{11}	5.60×10^{11}	4.56×10^{11}	6.91×10^{11}
Magnet	6.45×10^{8}	6.39×10^{8}	5.80×10^{8}	9.52×10^{8}
OUTBOARD				
First Wall				
Be	3.43×10^{14}	3.40×10^{14}	3.22×10^{14}	3.84×10^{14}
Cu	3.38×10^{14}	3.33×10^{14}	3.12×10^{14}	3.74×10^{14}
SS	3.40×10^{14}	3.35×10^{14}	3.11×10^{14}	3.71×10^{14}
Vacuum Vessel	7.73×10^{11}	7.61×10^{11}	6.22×10^{11}	9.39×10^{11}
Magnet	9.70×10^{7}	9.65×10^{7}	8.90×10^{7}	1.52×10^{8}

Table 4. Peak Gamma Flux Values from New and Old Processed Libraries (g/cm^2·s)

Evaluation	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0
Processing Code	NJOY TRANSX	NJOY TRANSX	NJOY TRANSX	NJOY TRANSX
Energy Groups	175n-42g	46n-21g	175n-42g	46n-21g
New/Old	New	New	Old	Old
INBOARD				
First Wall				
Be	3.03×10^{14}	3.00×10^{14}	2.91×10^{14}	2.88×10^{14}
Cu	2.93×10^{14}	2.86×10^{14}	2.82×10^{14}	2.77×10^{14}
SS	2.90×10^{14}	2.85×10^{14}	2.79×10^{14}	2.74×10^{14}
Vacuum Vessel	5.68×10^{11}	5.60×10^{11}	5.48×10^{11}	5.39×10^{11}
Magnet	6.45×10^{8}	6.39×10^{8}	6.29×10^{8}	6.24×10^{8}
OUTBOARD				
First Wall				
Be	3.43×10^{14}	3.40×10^{14}	3.30×10^{14}	3.27×10^{14}
Cu	3.38×10^{14}	3.33×10^{14}	3.25×10^{14}	3.20×10^{14}
SS	3.40×10^{14}	3.35×10^{14}	3.27×10^{14}	3.22×10^{14}
Vacuum Vessel	7.73×10^{11}	7.61×10^{11}	7.43×10^{11}	7.33×10^{11}
Magnet	9.70×10^{7}	9.65×10^{7}	9.42×10^{7}	9.43×10^{7}

Table 5. 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
<u>INBOARD</u>				
First Wall				
Be	1.07×10^{1}	1.08×10^{1}	1.01×10^{1}	1.07×10^{1}
Cu	2.23×10^{1}	2.23×10^{1}	2.19×10^{1}	2.53×10^{1}
SS	1.85×10^{1}	1.85×10^{1}	1.87×10^{1}	2.01×10^{1}
Vacuum Vessel	3.29×10^{-2}	3.27×10^{-2}	3.18×10^{-2}	3.97×10^{-2}
Magnet	3.15×10^{-5}	3.14×10^{-5}	3.09×10^{-5}	4.46×10^{-5}
<u>OUTBOARD</u>				
First Wall				
Be	1.38×10^{1}	1.38×10^{1}	1.30×10^{1}	1.36×10^{1}
Cu	2.67×10^{1}	2.66×10^{1}	2.64×10^{1}	3.04×10^{1}
SS	2.24×10^{1}	2.24×10^{1}	2.28×10^{1}	2.44×10^{1}
Vacuum Vessel	4.45×10^{-2}	4.45×10^{-2}	4.32×10^{-2}	5.36×10^{-2}
Magnet	4.68×10^{-6}	4.69×10^{-6}	4.70×10^{-6}	7.04×10^{-6}

The impact of neglecting the decay energy of short lived radionuclides was assessed by comparing the results obtained from the new and old processed FENDL libraries. The results are given in Table 6. Neglecting decay energy results in underestimating the power densities by ~2% in Be, ~7% in Cu and ~2% in SS of the FW. The underestimate is <3% in the TF coil. The results are almost identical in the Inconel VV. This is consistent with the results calculated using the two kerma factors provided in the KAOS library with and without decay energy which indicated that 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 new processed FENDL library includes decay energy for elements up to Cu only. Decay heat is not included yet for important elements such as Mo, Ni, Zr, Nb, Sn, W, and Pb.

Table 6. Impact of Neglecting Decay Heat on Peak Power Density Values (W/cm^3)

Evaluation	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0
Processing Code	NJOY, TRANSX	NJOY, TRANSX	NJOY, TRANSX	NJOY, TRANSX
Energy Groups	175n-42g	46n-21g	175n-42g	46n-21g
New/Old	New	New	Old	Old
Decay Energy	Yes	Yes	No	No
<u>INBOARD</u>				
First Wall				
Be	1.07×10^1	1.08×10^1	1.06×10^1	1.06×10^1
Cu	2.23×10^1	2.23×10^1	2.08×10^1	2.08×10^1
SS	1.85×10^1	1.85×10^1	1.82×10^1	1.82×10^1
Vacuum Vessel	3.31×10^{-2}	3.28×10^{-2}	3.32×10^{-2}	3.28×10^{-2}
Magnet	3.15×10^{-5}	3.14×10^{-5}	3.07×10^{-5}	3.05×10^{-5}
<u>OUTBOARD</u>				
First Wall				
Be	1.38×10^1	1.38×10^1	1.36×10^1	1.35×10^1
Cu	2.67×10^1	2.66×10^1	2.50×10^1	2.49×10^1
SS	2.24×10^1	2.24×10^1	2.21×10^1	2.21×10^1
Vacuum Vessel	4.47×10^{-2}	4.45×10^{-2}	4.47×10^{-2}	4.44×10^{-2}
Magnet	4.68×10^{-6}	4.69×10^{-6}	4.53×10^{-6}	4.57×10^{-6}

Radiation Damage

Table 7 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 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 FW and VV (<1% difference). The peak dpa results from the new and old processed FENDL libraries are identical. 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 the Cu FW, ~3% in the SS FW and ~22% in the Inconel VV. These differences are partially contributed by the large differences in the calculated neutron fluxes.

Table 7. 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	26.6	27.1	27.2
Vacuum Vessel	3.87×10^{-2}	3.85×10^{-2}	3.91×10^{-2}	4.69×10^{-2}
<u>OUTBOARD</u>				
First Wall				
Cu	38.5	38.4	39.8	40.4
SS	36.8	36.7	37.4	37.7
Vacuum Vessel	5.33×10^{-2}	5.29×10^{-2}	5.40×10^{-2}	6.43×10^{-2}

Gas Production

The peak end-of-life gas production values (helium, hydrogen and tritium) in the FW and VV have been calculated. Table 8 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 previously processed FENDL with 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%. These differences are much larger than predicted by differences in neutron flux values. Comparing the He production results from the recently processed FENDL multigroup library in which total gas production is included and from the previously processed library where we calculated the total gas production by adding the partial cross sections, good agreement exists for all elements except for Be (notice that Cu alloy has Be) where the difference is a factor of 37. Further examination of partial reactions revealed that the contribution from (n,2n) reaction in Be that produces 2 alphas was not included in the total He production cross section in the new library.

Table 8. Peak End-of-life Helium Production in FW and VV (appm @ 3 FPY)

Evaluation	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0	ENDF/B-V
Processing Code	NJOY TRANSX	NJOY TRANSX	NJOY TRANSX	NJOY TRANSX	MINX AMPX KAOS
Energy Groups	175n-42g	46n-21g	175n-42g	46n-21g	46n-21g
New/Old	New	New	Old	Old	
INBOARD					
First Wall					
Be	3.61×10^{2}	3.62×10^{2}	1.30×10^{4}	1.30×10^{4}	1.37×10^{4}
Cu	3.48×10^{2}	3.46×10^{2}	7.02×10^{2}	7.07×10^{2}	7.80×10^{2}
SS	6.04×10^{2}	6.02×10^{2}	6.02×10^{2}	6.02×10^{2}	6.41×10^{2}
Vacuum Vessel	3.58×10^{-1}	3.57×10^{-1}	3.61×10^{-1}	3.57×10^{-1}	5.44×10^{-1}
OUTBOARD					
First Wall					
Be	4.69×10^{2}	4.71×10^{2}	1.73×10^{4}	1.73×10^{4}	1.81×10^{4}
Cu	4.82×10^{2}	4.80×10^{2}	9.72×10^{2}	9.70×10^{2}	1.08×10^{3}
SS	7.90×10^{2}	7.88×10^{2}	7.89×10^{2}	7.88×10^{2}	8.41×10^{2}
Vacuum Vessel	4.85×10^{-1}	4.83×10^{-1}	4.88×10^{-1}	4.83×10^{-1}	7.36×10^{-1}

Table 9 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%. These are partially contributed by differences in neutron flux. The hydrogen production results are identical from the new processed FENDL library and the previously processed library where we calculated the total gas production by adding the partial cross sections.

Table 10 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

Table 9. Peak End-of-life Hydrogen Production in FW and VV (appm @ 3 FPY)

Evaluation	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0	ENDF/B-V
Processing Code	NJOY TRANSX	NJOY TRANSX	NJOY TRANSX	NJOY TRANSX	MINX AMPX KAOS
Energy Groups	175n-42g	46n-21g	175n-42g	46n-21g	46n-21g
New/Old	New	New	Old	Old	
INBOARD					
First Wall					
Be	1.72×10^2	1.69×10^2	1.73×10^2	1.69×10^2	1.78×10^2
Cu	2.15×10^3	2.16×10^3	2.15×10^3	2.16×10^3	2.08×10^3
SS	2.00×10^3	1.98×10^3	1.99×10^3	1.98×10^3	1.75×10^3
Vacuum Vessel	2.31	2.32	2.33	2.32	2.78
OUTBOARD					
First Wall					
Be	2.27×10^2	2.23×10^2	2.27×10^2	2.23×10^2	2.35×10^2
Cu	2.98×10^3	3.00×10^3	2.99×10^3	3.00×10^3	2.88×10^3
SS	2.82×10^3	2.79×10^3	2.82×10^3	2.79×10^3	2.47×10^3
Vacuum Vessel	3.14	3.15	3.15	3.15	3.77

Table 10. Peak End-of-life Tritium Production in FW and VV (appm @ 3 FPY)

Evaluation	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0	FENDL/E-1.0	ENDF/B-V
Processing Code	NJOY TRANSX	NJOY TRANSX	NJOY TRANSX	NJOY TRANSX	MINX AMPX KAOS
Energy Groups	175n-42g	46n-21g	175n-42g	46n-21g	46n-21g
New/Old	New	New	Old	Old	
INBOARD					
First Wall					
Be	1.72×10^2	1.69×10^2	1.73×10^2	1.69×10^2	1.78×10^2
Cu	4.57	4.52	4.57	4.51	4.64
SS	0.93	0.92	0.92	0.92	1.57
Vacuum Vessel	1.75×10^{-5}	1.78×10^{-5}	1.74×10^{-5}	1.78×10^{-5}	4.69×10^{-4}
OUTBOARD					
First Wall					
Be	2.27×10^2	2.27×10^2	2.27×10^2	2.23×10^2	2.35×10^2
Cu	6.37	6.29	6.37	6.29	6.46
SS	1.30	1.30	1.30	1.30	2.24
Vacuum Vessel	2.37×10^{-5}	2.42×10^{-5}	2.36×10^{-5}	2.41×10^{-5}	6.35×10^{-4}

than the FENDL library processed by NJOY with the same group structure. The results for the FW are different by ~5% in Be, ~3% in Cu, and ~72% in SS. The peak Inconel VV T production is higher by a factor of ~26. These differences are very large and need further investigation. The tritium production results are identical from the new processed FENDL library and the previously processed library where we calculated the total gas production by adding the partial cross sections.

Magnet Radiation Effects

Table 11 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 ~7%. 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 same group structure using different processing codes (MINX/AMPX), the peak end-of-life magnet radiation effects differ by as much as 45%. This can be attributed mostly to the large difference in the calculated neutron and gamma fluxes at the magnet. The new processed library with decay energy gives identical results for fast neutron fluence and Cu dpa but gives ~7% higher insulator dose compared to the previously processed library without decay energy. This difference is due to the contribution of decay energy.

Observations and Comments on the New Processed Multigroup Library

Total gas production cross sections (.h1, .h2, .h3, .he3, .he4) have been added in the MATXS processed library. Gas production data are missing for Sc45, Y89, and Au197. Helium production for Be includes only the (n,α) reaction. The contribution from the (n,2n) reaction in Be that produces 2 alphas should also be included.

An algorithm for TRANSX was provided to combine the radionuclide production cross sections with decay information to determine the decay heat contribution. The beta decay energy

Table 11. Peak End-of-Life (@ 3 FPY) Magnet Radiation Effects

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>				
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)	1.62×10^{22}	1.62×10^{22}	1.60×10^{22}	2.14×10^{22}
Cu dpa	9.66×10^{-5}	9.65×10^{-5}	1.03×10^{-4}	1.33×10^{-4}
<u>OUTBOARD</u>				
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)	2.40×10^{21}	2.41×10^{21}	2.43×10^{21}	3.37×10^{21}
Cu dpa	1.43×10^{-5}	1.44×10^{-5}	1.56×10^{-5}	2.09×10^{-5}

is assumed to be deposited locally and is added to the neutron kerma while the gamma production cross sections are modified to include the produced decay gamma thus allowing them to be transported before depositing their energy. This is a better treatment for decay gamma than assuming that all energy is deposited locally. The new processed FENDL library includes decay energy for elements up to Cu only. Decay heat is not included yet for important elements such as Mo, Ni, Zr, Nb, Sn, W, and Pb.

Observations and Comments on the FENDL Continuous Energy Library

Adding gas production and damage data to the processed library is very helpful and convenient for calculating design relevant parameters. Gas production and damage data are not included for carbon. The file for Na-23 from JENDL 3.1 has not been updated and presumably does not include gas production and damage data. ENDF/B-VI files are provided for He-3, He-4, Sc-45, and Au-197 which are not in the FENDL/E-1.0 material list. Data for helium have been used in the ITER calculation where liquid He is used in the magnets. The BROND-2 files for

H-2 and N-14 did not work with MCNP. The ENDF/B-VI files were substituted for these isotopes.

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 differences between results 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 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%).

Comparing 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. The KAOS library based on ENDF/B-V gives higher power densities compared to the FENDL library processed by NJOY with the same group structure. The differences can be as high as ~14% in the Cu FW, ~20% in the Inconel VV and ~50% 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. Large differences in damage and gas production results of the KAOS library based on ENDF/B-V and the FENDL library processed by NJOY are contributed mostly by large differences in calculated neutron and gamma fluxes except for a 52% difference in He production in Inconel VV, 72% in T production in SS and a factor of 26 in T production in Inconel VV. 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 new processed multigroup library that includes total gas production cross sections and decay energy has been used. Gas production results from the new library are identical to those from the previously processed library where we calculated the total gas production by adding the partial cross sections. The only exception is helium production for Be where the contribution from the (n,2n) reaction in Be that produces 2 alphas was not included in the total He production cross section in the new library.

The effect on nuclear heating of the decay energy associated with short lived ($T_{1/2}$ < 1 day) radionuclides in the kerma factor is <7%. The new processed FENDL library includes decay energy for elements up to Cu only. Decay heat is not included yet for important elements such as Mo, Ni, Zr, Nb, Sn, W, and Pb.

3-D neutronics and shielding analyses have been performed for the divertor region of ITER using the recently processed ACE formatted FENDL/MC-1.0 sublibrary that includes gas production cross sections and damage energy cross sections. The inclusion of gas production and damage data facilitates the calculation of the design relevant parameters. Gas production and damage data need to be included for carbon. The files for H-2 and N-14 based on BROND-2 which is the designated evaluation for them in FENDL-1 do not work with MCNP. The ENDF/B-VI files were substituted for these isotopes.

References

[1] 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).

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

[3] IAEA Advisory Group Meeting on "Completion of FENDL-1 and Start of FENDL-2", Del Mar, California, December 1995; Summary Report Prepared by A.B. Pashchenko and Published as IAEA Nuclear Data Section Report INDC(NDS)-352 (March 1996).

[4] R.E. MacFarlane, Los Alamos National Laboratory, Private communications (December 1996).

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

[6] IAEA Specialists Meeting on the Fusion Evaluated Nuclear Data Library (FENDL), Vienna, Austria, May 1989; Summary Report Edited by V. Goulo and Published as IAEA Nuclear Data Section Report INDC(NDS)-223/GF (1989).

[7] R. MacFarlane et al., "NJOY 91.38, A Code System for Producing Pointwise and Multi-group Neutron and Photon Cross Sections from ENDF/B Evaluated Nuclear Data," PSR-171, Radiation Shielding Information Center, Oak Ridge National Laboratory (July 1992).

[8] J. Briesmeister , Ed., "MCNP, A General Monte Carlo N-Particle Transport Code, Version 4A," LA-12625-M, (1993).

[9] M. E. Sawan, L. Petrizzi, R. Santoro, and D. Valenza, " Three-Dimensional Neutronics and Shielding Analyses for the ITER Divertor," Fusion Technology, Vol. 30, pp. 601 (December 1996).

[10] 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).

[11] C. Weisbin et al., "MINX, A Multi-group Interpretation of Nuclear Cross Sections, Los Alamos National Laboratory Report, LA-6486-MS (1976).

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

[13] 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).