Review of experimental validation of FENDL-2.1 nuclear data library, needs for improvements and recommendations for FENDL-3

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1. Introduction and methodology

FENDL-2.1 has been adopted and is being used as reference nuclear data library for ITER design. To this purpose, extensive work of validation and verification of FENDL-1/2.0 libraries for neutronics calculations was performed during the R&D activities for ITER design and Test Blanket Module (TBM) project through *ad hoc* benchmark experiments performed on mock-ups of components at 14 MeV neutron generators.

This report presents the validation of the latest FENDL version, FENDL-2.1, using the existing benchmark experiments carried out at the FNS (JAEA), FNG (ENEA) and TUD (Technical University of Dresden) 14 MeV neutron generators. We investigate a wide spectrum of nuclear relevant issues, including those related to shielding blankets (with and without streaming paths), breeding blanket, vessel, magnets, and divertor.

All experiments have been analyzed with FENDL-2.1 and using the MCNP-4C/5 code. In all cases, very detailed geometrical models of the experimental set ups have been used, including the detectors employed in the measurements, the neutron generator and the bunker hall. Ratios of the Calculated (C) over the measured (E) quantities were obtained.

The unsatisfactory results obtained with FENDL-2.1 are pointed out as well data to be improved in FENDL-3.0.

In table 1, the original sources of data in the three FENDL releases are given for the main materials present in the experiments.

Table 1. Origin of data in FENDL-1.0/2.0/2.1 for main materials present in experiments carried out at FNG, FNS and TUD, and proposed sources for FENDL-3 data.

Material	FENDL-1.0	FENDL-2.0	FENDL-2.1	Proposed FENDL-3	Comments
H-1		ENDF/B-VI mod 1	JENDL-3.3	ENDF/B-VII	Changed
Li-6 Li-7	ENDF/B-VI mod 1 ENDF/B-VI mod 0	ENDF/B-VI mod 1 ENDF/B-VI mod 0	ENDF/B-VI mod 1 ENDF/B-VI mod 0	ENDF/B-VII ENDF/B-VII	Unchanged
Be-9	ENDF/B-VI mod 1	JENDL-FF	ENDF/B-VI mod 2	ENDF/B-VII	Unchanged
C-12	ENDF/B-VI mod 2 (C-nat)	JENDL-FF	JENDL-FF	JENDL-HE	Changed
N-14	Brond-2	JENDL-FF	JENDL-FF	JENDL-HE	Changed
O-16	ENDF/B-VI mod 1	JENDL-FF	ENDF/B-VI mod 3	ENDF/B-VII	Changed
Si-28, 30 Si-29	Brond-2 (Si-nat) Brond-2 (Si-nat)	ENDF/B-VI mod 1 ENDF/B-VI mod 1	ENDF/B-VI mod 2 ENDF/B-VI mod 3	ENDF/B-VII ENDF/B-VII	Si-28 changed
V-51	ENDF/B-VI mod 1 (V-nat)	JENDL-FF	JENDL-3.3 (V-nat)	JENDL-HE	Changed
Cr-50, 54 Cr-52, 53	ENDF/B-VI mod 2 ENDF/B-VI mod 2	ENDF/B-VI mod 2 ENDF/B-VI mod 2	ENDF/B-VI mod 5 ENDF/B-VI mod 4	ENDF/B-VII	Unchanged
Fe-54 Fe-56 Fe-57, 58	ENDF/B-VI mod 2 ENDF/B-VI mod 2 ENDF/B-VI mod 2	ENDF/B-VI mod 2 EFF-3.0 ENDF/B-VI mod 2	ENDF/B-VI mod 5 EFF-3.1 ENDF/B-VI mod 4	ENDF/B-VII EFF-3.1 ENDF/B-VII	Unchanged
Ni-58, 60 Ni-61, 62 Ni-64	ENDF/B-VI mod 2 ENDF/B-VI mod 2 ENDF/B-VI mod 2	ENDF/B-VI mod 2 ENDF/B-VI mod 2 ENDF/B-VI mod 2	EFF-3.0 ENDF/B-VI mod 5 ENDF/B-VI mod 4	EFF-3.1 ENDF/B-VII ENDF/B-VII	Unchanged
Cu-63, 65	ENDF/B-VI mod 3	ENDF/B-VI mod 3	ENDF/B-VI mod 5	ENDF/B-VII	Unchanged
Mo-92, 94, 95, 96, 97, 98, 100	JENDL-3.1 (Mo-nat)	JENDL-FF (Mo-nat)	JENDL-3.3	JENDL-HE	Changed above 20 MeV
W-182, 183, 184, 186	ENDF/B-VI mod 0	JENDL-FF (W-nat)	ENDF/B-VI mod 2	ENDF/B-VII	Unchanged
Pb-206, 208 Pb-207	ENDF/B-VI mod 1 ENDF/B-VI mod 2	ENDF/B-VI mod 1 ENDF/B-VI mod 2	ENDF/B-VI mod 2 ENDF/B-VI mod 3	ENDF/B-VII ENDF/B-VII	Changed Changed

2. Benchmark experiments at FNG

Several integral benchmark experiments have been carried out at the 14-MeV Frascati Neutron Generator (FNG) of ENEA since 1992, for fusion nuclear data validation and for verification of fusion reactor nuclear design. Some data were measured at the 14 MeV neutron generator of TUD. The experiments are presented in the following. Analyses were carried out with MCNP4/5C and using FENDL-1, FENDL-2.0, FENDL-2.1 and JEFF-3.1 nuclear data libraries. Typical results are plotted in Appendix A.

1) Bulk Shield Experiment [1]

The first integral experiment considered here is the *Bulk Shield Experiment*, consisting in a mockup of the ITER shielding blanket/vacuum vessel, simulated by alternating layers of stainless steel, AISI-316-type, (SS316) and Perspex material to simulate water, and followed by the superconducting magnet region simulated by alternating layers of SS316 and copper as shown in Fig. A1. Several activation reactions, ^{197}Au(n,γ), ^{55}Mn(n,γ), ^{115}In(n,n'), ^{58}Ni(n,p), ^{56}Fe(n,p), ^{27}Al(n,α), ^{58}Ni(n,2n), and ^{93}Nb(n,2n), were used to derive the neutron flux. Gamma heating was measured all along the mock-up depth and in the magnet region using TLD-300 dosimeters (CaF_2:Tm).
Neutron and photon flux spectra were measured at positions A (41.4 cm) and B (87.6 cm). A set of gas-filled proportional counters and a stilbene scintillation spectrometer were used in the neutron energy range up to 3 MeV. An NE-213 scintillation spectrometer was used for neutron flux spectra from 1 to 15 MeV. Photon flux spectra were measured with the NE-213 spectrometer above 0.2 MeV. Results of this experiment are shown in Figs. A2-A14 and are discussed with the following one.

2) The Streaming Experiment[2]

This experiment consisted in a variation of the *Bulk Shield Experiment*, where a circular central channel and a rectangular cavity were introduced in the shielding blanket/vessel block to simulate streaming paths between ITER blanket modules. The arrangement of this experiment is shown in Figs. A15 and A16. Neutron flux was measured by activation technique, using several reactions along the central channel and inside the cavity at locations out-of-site of the point source. Neutron and photon flux spectra were measured at positions A (41.4 cm) and B (87.6 cm) with source on axis (A0, B0). Additional measurements were carried out with detectors shifted off the axis by 7.5 cm, 15.0 and 9.0 cm (A1, A2 and B1). Neutron spectra were measured in the energy range between about 20 keV and 15 MeV. A set of gas-filled proportional counters and a stilbene scintillation spectrometer were used in the energy range up to 3 MeV. NE-213 scintillation spectrometer was used for the energy range 1 to 15 MeV. Photon flux spectra were measured with the NE-213 spectrometer above 0.2 MeV. Results of this experiment are shown in Figs. A17-A29.

In both experiments the following results were obtained:

- the fast neutron fluxes are underestimated by up to about 10% at 40 cm depth, and by 25-30% at about 1 m depth (outside uncertainty).
- the gamma ray flux is also underestimated by ~20% at 1 m depth.
- a better agreement is found for the thermal neutron flux (within ±10% uncertainty).
- Nuclear heating is underestimated by 10% (within ±10% uncertainty up to 70 cm, within ±30% at > 70 cm).
- no significant differences are found between FENDL-1.0 (main materials in stainless steel from ENDF/B-VI), FENDL-2.0 and FENDL-2.1.

From these results it is concluded that the neutron and gamma ray fluxes are predicted in stainless steel/water shield assemblies by FENDL-2.1 within +/- 30% uncertainty at 1 m depth.

3) Tungsten Experiment[3]

The experimental set-up consisted of a block of DENSIMET-176 (W = 92.3%w, Fe = 2.6%w, Ni = 4.2%w) and DENSIMET-180 (W = 95%w, Fe = 1.5%w, Ni = 3.4%w) of size of about 42-47 cm x 46.85 cm and 49 cm in thickness as shown in Fig. A30. Eight different reactions, ^{197}Au(n,γ), ^{55}Mn(n,γ), ^{115}In(n,n'), ^{58}Ni(n,p), ^{56}Fe(n,p), ^{27}Al(n,α), ^{58}Ni(n,2n), ^{90}Zr(n,2n) and ^{93}Nb(n,2n), were used to derive the neutron flux. Gamma heating was measured using TLD-300 dosimeters (CaF_2:Tm). Neutron and gamma ray spectra were measured in four positions in W assembly, at 5 cm (P1), 15 cm (P2), 25 cm (P3) and 35 cm penetration depth. Results of this experiment are shown in Figs. A31-A47.

In the case of tungsten, FENDL-2.1 performs better than FENDL-2.0. In fact:

- an improvement is observed in the prediction of the fast neutron flux that is well predicted by FENDL-2.1.
- the thermal neutron flux is well predicted by all libraries.
- the photon flux is overestimated at all depths.
- the nuclear heating, which is dominated by the neutron contribution, is underestimated by all libraries, especially in the front position.

4) Silicon Carbide (SiC) Experiment[4]

The SiC composite is a low activation structural material potentially important for the fusion reactor. The experimental setup consisted of a block of sintered SiC composite with the dimensions of 45.7 x 45.7 x 71.1 cm^3 as shown in Fig. A48. Nb-93(n,2n), Al-27(n,α), Ni-58(n,p), and Au-197(n,γ) reaction rates, neutron spectra and nuclear heating rates were measured at various positions in the mock-up. Neutron and gamma ray spectra were measured in four positions in the SiC assembly, at 12.70 cm (P1), 27.94 cm (P2), 43.18 cm (P3) and 58.42 cm penetration depth. Results of this experiment are shown in Figs. A49-A56.

- FENDL-2.1 strongly underestimates the fast neutron flux in SiC (30% at about 55 cm in depth).
- All libraries underestimate by 10 – 20 % the low energy neutron flux which, however, depends on the concentration of boron impurity in the SiC material that is not accurately known.
- The results for nuclear heating, also influenced by the used value of the boron concentration, are consistent with those obtained in the analysis of the ^{197}Au(n,γ) measurements.

5) Test Blanket Module – Helium Cooled Pebble Bed (HCPB) Experiment[5]

The experiment investigated a mock up of the Test Blanket Module – Helium Cooled Pebble Bed concept, to be tested in ITER. In this experiment, the neutron flux was measured as a function of depth in the beryllium block in the module using activation foil technique. The neutron and gamma ray flux spectra were measured behind the module at the neutron generator of the Technical University of Dresden (TUD), using an NE-213 liquid scintillator (fast neutrons and gammas), ^{3}He proportional counter (TOA (Time-Of-Arrival) of slow neutrons)(Positions P1 and P2). The tritium production rate was measured in the Li composite in the breeder cassettes (using Li_2CO_3pellets). Results of this experiment are shown in Figs. A57-A64. The following conclusions were derived :

- The neutron flux in the Be layer is well predicted by FENDL-2.0/2.1 and by JEFF-3.1 within the total combined uncertainties (~ ±5%) up to about 24 cm depth. No significant differences between FENDL-2.0/2.1 and JEFF-3 are observed.
- The fast neutron flux (E > 1 MeV) was found to be slightly overestimated by about 10% behind the mock-up (P2 in Fig.XX). This indicates that shielding calculations for the HCPB blanket are conservative.
- The γ-ray flux is underestimated by all libraries by about 10% at the back of the mock-up
- The slow neutron flux investigated by time-of-arrival spectroscopy is underestimated in the mock-up by about 20%.
- Consistently, a slight underestimation is found in the calculation of the tritium production in the breeder material in the cassettes, the underestimation being smaller at deeper positions. C/E average values for the tritium production rate range from 0.86 (front position) to 0.92 at larger depths. Also in this case, no significant differences between FENDL-2.0/2.1 and JEFF-3 are observed.

6) Test Blanket Module – Helium Cooled Lithium Lead (HCLL) Experiment[6]

The experiment investigated a mock up of the Test Blanket Module – Helium Cooled Lithium Lead concept, to be tested in ITER. The tritium production rate was measured in the Li composite (Li_2CO_3) as a function of depth in the Pb-Li block. The neutron flux was measured as a function of depth in the Pb-Li module using activation foil technique. The measurement of the neutron and gamma flux spectra behind the module is still in progress at the TUD neutron generator. Results of this experiment are shown in Figs. A65-A67. The following conclusions were derived :

- The neutron flux in thePb-Li is well predicted by FENDL-2.1 and by JEFF-3.1 within the total combined uncertainties (~ ±5%) up to about 30 cm depth. No significant differences between FENDL-2.1 and JEFF-3.1 are observed.

- The tritium production in the breeder material is very well predicted by FENDL-2.1 and by JEFF-3.1 within the total combined uncertainties (~ ±7.4%). Also in this case, no significant differences between FENDL-2.0/2.1 and JEFF-3.1 are observed.

3. *Benchmark experiments at FNS*

Many integral benchmark experiments with DT neutrons have been carried out for nuclear data verification for fusion nuclear design at the Fusion Neutronics Source (FNS) facility in Japan Atomic Energy Agency (JAEA) since 1981. Three types of integral benchmark experiments for nuclear data verification with DT neutrons have been performed for long time at JAEA/FNS. The first is an *in situ* benchmark experiment, the second is a Time-Of-Flight (TOF) experiment, and the third is a breeding blanket experiment. The first and second experiments are carried out for benchmarking nuclear data, so analysis results for these experiments will be described in this report.

(1) *In situ* benchmark experiment[7]

Figure 1 shows a typical experimental configuration. Neutron spectra of almost the whole neutron energy, reaction rates for various reactions, gamma heating rates and so on were measured inside the experimental assembly of simple geometry. Size of experimental assemblies is different for each experiment depending on material amounts which we have. We have experimental data of lithium oxide, beryllium, graphite, silicon carbide, vanadium, iron, type 316 stainless steel (SS316), copper, tungsten, etc.

(2) Time-Of-Flight (TOF) experiment[8]

Figure 2 shows a typical experimental configuration. Angular neutron leakage spectra above 100 keV from the simple geometry slab were measured for lithium oxide, beryllium, graphite, nitrogen, oxygen, iron, copper, lead, etc. by using a collimator system. Size of experimental assemblies is different for each experiment depending on material amounts which we have.

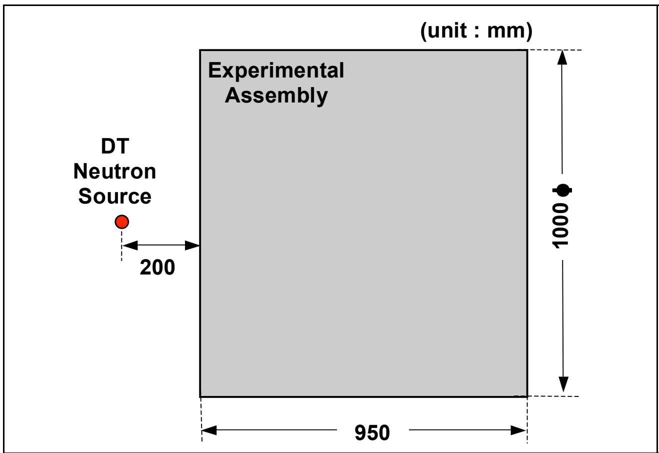

Fig. 1. Experimental setup for *in situ* experiments.

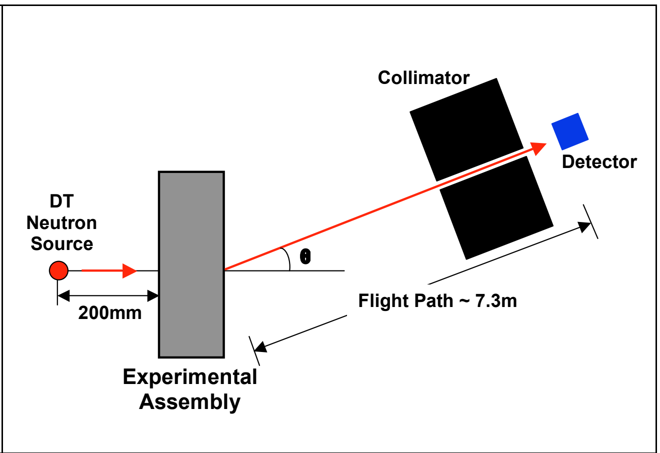

Fig.2. Experimental setup for TOF experiments.

The experimental configurations are summarized in Table 2. Analyses of these experiments were carried out with MCNP4C and the nuclear data libraries; FENDL-1, FENDL-2.0, FENDL-2.1, JENDL-3.3[9], JEFF-3.1 and ENDF/B-VII.0[10]. Typical results are plotted in Appendix B. Discussion to the results is also described in Table 2.

Table 2. Experimental configuration and discussion to results.

Experiment		Assembly		Discussion
		Shape	Size	
Li_2O	*in situ*	Quasi cylinder as shown in Fig. 3	630 mm in effective diameter 610 mm in thickness	All the calculation results are almost the same and agree with the measured data well within the experimental error except for the reaction rate of the $^{27}Al(n,\alpha)^{24}Na$ reaction and the fission rate of ^{235}U, C/Es of which are slightly out of the experimental errors but are considered to be good.
	TOF	Quasi cylinder as shown in	630 mm in effective diameter 48, 200, 400 mm in	All the calculation results represent the measured leakage neutron spectra from the lithium oxide slabs very well.

		Fig. 3	thickness	
Be	*in situ*	Quasi cylinder as shown in Fig. 3	630 mm in effective diameter 455 mm in thickness	The calculation results with FENDLs are almost the same except for the reaction rate of the ^{115}In(n,n')^{115m}In reaction, where FENDL-1.0 and -2.1 are better than FENDL-2.0. These calculation results are by 1.1 to 1.4 larger than the measured reaction rates of the ^{6}Li(n,α)^{3}T, ^{197}Au(n,γ)^{198}Au and ^{235}U(n,fission) reactions, which are sensitive to low energy neutrons. The calculation result with JENDL-3.3 shows the same tendency as that with FENDL-2.0, while that with ENDF/B-VII.0 is almost the same as that with FENDL-2.1. That with JEFF-3.1 agrees with the measured neutron flux of 3 to 10 MeV and reaction rate of the ^{115}In(n,n')^{115m}In reaction better. It is concluded that JEFF-3.1 is slightly better than others, while all the libraries are not good for low energy neutrons.
	TOF	Quasi cylinder as shown in Fig. 3	630 mm in effective diameter 51, 152 mm in thickness	The calculation results with FENDL are almost the same and represent the measured neutron flux well except for neutron flux of 3 to 10 MeV from 152 mm thick beryllium slab. The calculation with JEFF-3.1 agrees with the measured neutron flux of 3 to 10 MeV from 152 mm thick beryllium slab as well. JEFF-3.1 is slightly better than others
C (Graphite)	*in situ*	Quasi cylinder as shown in Fig. 3	630 mm in effective diameter 610 mm in thickness	The calculation result with FENDL-1.0 is by 10% larger than measured reaction rates of the ^{93}Nb(n,2n)^{92m}Nb and ^{115}In(n,n')^{115m}In reactions, while those with FENDL-2.0 and -2.1 agree with them within 10%. The C/E for the reaction rate of the ^{197}Au(n,γ)^{198}Au reaction is over 2.0, which is not consistent to the fission rate of ^{235}U. The measured reaction rate of the ^{197}Au(n,γ)^{198}Au reaction might have some problems. The calculation results with JEFF-3.1 and ENDF/B-VII.0 are almost the same as that with FENDL-1.0, while that with JENDL-3.3 is almost the same as those with FENDL-2.0 and -2.1.
	TOF	Quasi cylinder as shown in Fig. 3	630 mm in effective diameter 51, 202, 405 mm in thickness	The calculation result with FENDL-1.0 underestimates the measured neutron flux below 2 MeV from 202 mm thick slab, while those with FENDL-2.0 and -2.1 agree with it better. The difference among the calculation results with FENDLs is small for the leakage neutron flux from 405 mm thick slab. The calculation results with ENDF/B-VII.0 and JEFF-3.1 are almost the same as that with FENDL-1.0.
Liq. N_2	TOF	Cylinder tank	600 mm in diameter 200 mm in thickness	The calculation result with FENDL-1.0 underestimates the measured neutron flux more, while those with FENDL-2.0 and -2.1 agree it well except for at 66.6 deg. The calculation results with ENDF/B-VII.0, JEFF-3.1 and JENDL-3.3 are almost the same as that with FENDL-2.1.
Liq. O_2	TOF	Cylinder tank	600 mm in diameter 200 mm in thickness	The calculation results with FENDLs are almost the same and underestimate the neutron flux more in the larger angles. The calculation results with ENDF/B-VII.0, JEFF-3.1 and JENDL-3.3 show the similar tendency with those with FENDLs.
SiC	*in situ*	Rectangular as shown in Fig. 4	457 mm x 457 mm x 711 mm in thickness	The calculation result with FENDL-1.0 underestimates the measured reaction rates of the ^{93}Nb(n,2n)^{92m}Nb, ^{27}Al(n,α)^{24}Na and ^{115}In(n,n')^{115m}In reactions with the depth, while those with FENDL-2.0 and -2.1 agree them well. The agreement between the measurement and calculation results with FENDLs for the reaction rate of the ^{197}Au(n,γ)^{198}Au reaction is not so bad, if the calculation error is considered. On the contrary, the measured gamma ray heating rate is not represented in the calculations with FENDLs. The tendency for the calculation results with ENDF/B-VII.0, JEFF-3.1 and JENDL-3.3 is similar with that with FENDL-2.0 and -2.1, but there are small differences. The calculation results with ENDF/B-VII.0, JEFF-3.1 and JENDL-3.3 are slightly larger than those with FENDL-2.0 and -2.1 for the reaction rates of the ^{93}Nb(n,2n)^{92m}Nb, ^{27}Al(n,α)^{24}Na and ^{115}In(n,n')^{115m}In reactions and the calculation result with JENDL-3.3 agrees with the measurement best. This experiment was also carried out with the same experimental assembly at FNG. The C/E trend of the reaction rates of the ^{93}Nb(n,2n)^{92m}Nb is similar, while that of the reaction rates of the ^{197}Au(n,γ)^{198}Au reaction is slightly different.
V	*in situ*	Rectangular as shown in Fig. 5	254 mm x 254 mm x 254 mm in thickness covered with 50 mm thick graphite	The calculation results with FENDLs agree the measured reaction rates of the ^{93}Nb(n,2n)^{92m}Nb and ^{115}In(n,n')^{115m}In reactions well. On the contrary, the agreement between the calculation and measured results are not good for low energy neutrons, though the calculation result with FENDL-1.0 is better than those with FENDL-2.0 and -2.1. The calculation result with JENDL-3.3 is almost the same as that with FENDL-2.1. The calculation result with ENDF/B-VII.0 is better than that with JENDL-3.3, but it has a strange peak in the neutron spectra of 4 to 10 MeV. The calculation result with JEFF-3.1 agrees the measurement best, though it still underestimates

				the measured low energy neutron flux.
Fe	*in situ*	Cylinder	1000 mm in diameter 950 mm in thickness	The calculation results with FENDLs are similar and agree with the measurement within 15%. Only the exception is the calculated neutron flux above 10 MeV with FENDL-1.0, which underestimates the measurement more than 15%. It is already specified that this reason is because the forward part in the angular distribution of the elastic scattering is smaller in ^{56}Fe of FENDL-1.0[11]. The calculation result with JEFF-3.1 represents the measurement as well as that with FENDL-2.1, while that with ENDF/B-VII show the same underestimation for the neutron above 10 MeV as that with FENDL-1.0. That with JENDL-3.3 clearly overestimates the measured neutron flux below a few keV, which is due to larger cross section data of the first inelastic scattering of ^{57}Fe in JENDL-3.3[12]. The first inelastic scattering cross section data of ^{57}Fe will be revised in JENDL-4.
	TOF	Cylinder	1000 mm in diameter 50, 200, 400, 600 mm in thickness	All the calculation results with FENDLs, JENDL-3.3, JEFF-3.1 and ENDF/B-VII.0 represent the measured leakage neutron spectra above 100 keV from 50 and 200 mm thick iron assemblies very well, while they do not always agree with the measured leakage neutron spectra below 1 MeV from 400 and 600 mm thick iron assemblies.
SS 316	*in situ*	Cylinder with reflector as shown in Fig. 6	1200 mm in diameter 1118 mm in thickness	All the calculation results are almost the same. They agree with the measured neutron flux above a few hundred eV, while they overestimate the measured neutron flux below a few hundred eV. One of possible reasons of the overestimation is a molybdenum resonance. All the calculation results for the reaction rate of the ^{115}In(n,n')^{115m}In reaction agree with the measured data within 20%, which is larger than the experimental error but is not so large. The C/Es of the integrated neutron flux from 0.1 to 1 MeV are almost unity up to the depth of 600 mm, while the calculation results with FENDL-1.0, JENDL-3.3 and ENDF/B-VII.0 underestimate the measured data by more than 10% at the depth of 900 mm. All the C/Es of the integrated neutron flux from 10 to 100 eV are 1.3 – 1.4 around the depth of 400 mm as pointed out above. All the calculation results of the gamma-ray heating rate generally agree with the measured data within the larger experimental error up to the depth of 600 mm, but all the calculation results underestimate the measured data by 20 – 30% at the depth of 900 mm. This analysis suggests that all the libraries for nuclei included in SS316 seem to be good except for molybdenum, which may cause some problems for low energy neutrons.
Cu	*in situ*	Quasi cylinder as shown in Fig. 3	630 mm in effective diameter 610 mm in thickness	All the calculation results are almost the same. They agree with the measured neutron flux above 600 eV, while they drastically underestimate the measured neutron flux below 600 eV. One of possible reasons of the underestimation is copper resonance peak. The larger resonance peak around 600 eV of ^{63}Cu may cause the underestimation below 600 eV. The C/Es for the reaction rate of the ^{115}In(n,n')^{115m}In reaction are around 0.9 up to the depth of 400 mm within 15%, while all the calculation results overestimate it by more than 25% at the depth of 500 mm. The C/Es of the integrated neutron flux from 0.1 to 1 MeV largely change at every position. It is considered that all the nuclear data have some problems.
W	*in situ*	Quasi cylinder as shown in Fig. 3	575 mm in effective diameter 507 mm in thickness	The calculation results with FENDL-1.0, -2.1 and ENDF/B-VII.0 slightly overestimate the measured reaction rate of the ^{93}Nb(n,2n)^{92m}Nb reaction at deeper positions than 200 mm, while that with FENDL-2.0 underestimate it by more than 10% at the depth of 380 mm. Those with JENDL-3.3 and JEFF-3.1 show the similar trend with that with FENDL-2.1, but the overestimation is smaller. All the calculation results agree with the measured reaction rate of the ^{115}In(n,n')^{115m}In reaction within 15%. As for low energy neutrons, the calculation result with FENDL-1.0 underestimates by more than 30% at the depth of 380 mm. This tendency of underestimation appears in the calculation results with FENDL-2.1 and ENDF/B-VII.0, but the underestimation is smaller. The calculated gamma ray heating rates agree with the measurement within the large experimental error. This experiment was also carried out with a different experimental assembly at FNG, but the results are different from those at FNS, reasons of which are not found out yet.
Pb	TOF	Quasi cylinder as shown in Fig. 3	100 cm in diameter 51, 203, 406 mm in thickness	The calculation results with FENDL-2.1, JEFF-3.1 and ENDF/B-VII.0 agree with the measured leakage neutron flux very well, while those with FENDL-1.0, -2.0 and JENDL-3.3 disagree with the measured leakage neutron flux.

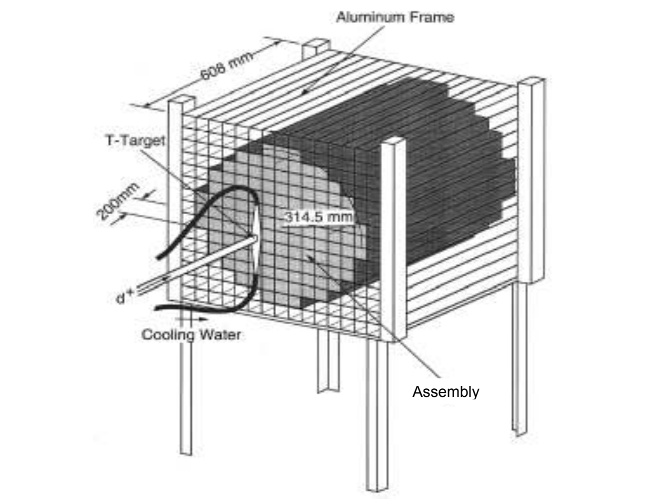

Fig. 3. Quasi cylindrical experimental assembly.

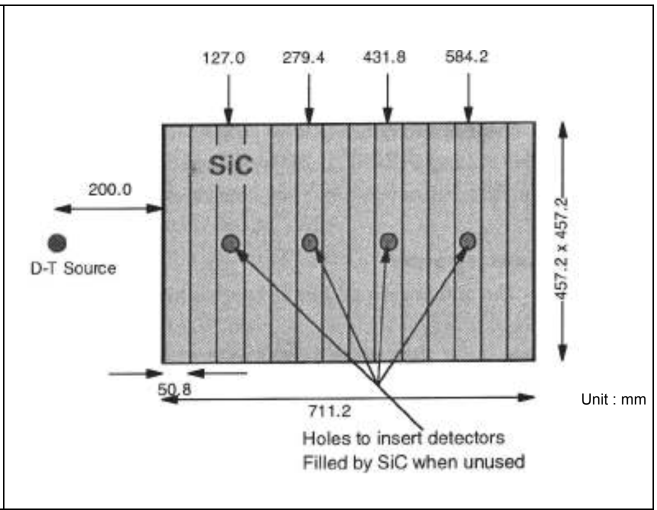

Fig.4. Experimental setup in SiC *in situ* experiment.

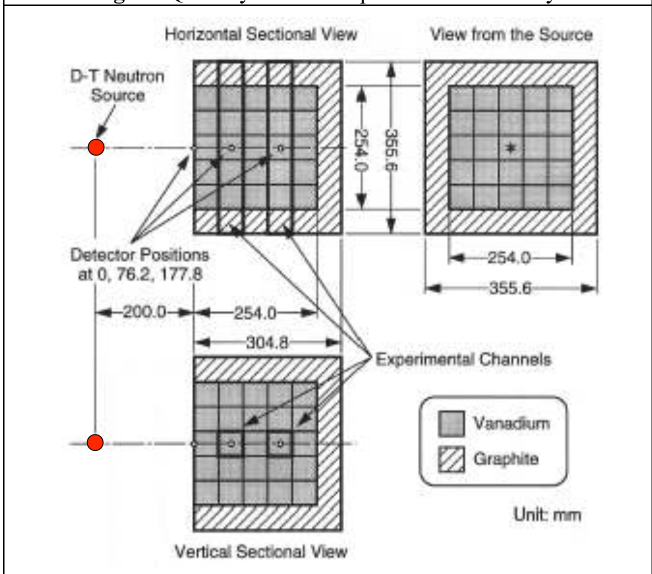

Fig. 5. Experimental setup in V *in situ* experiment.

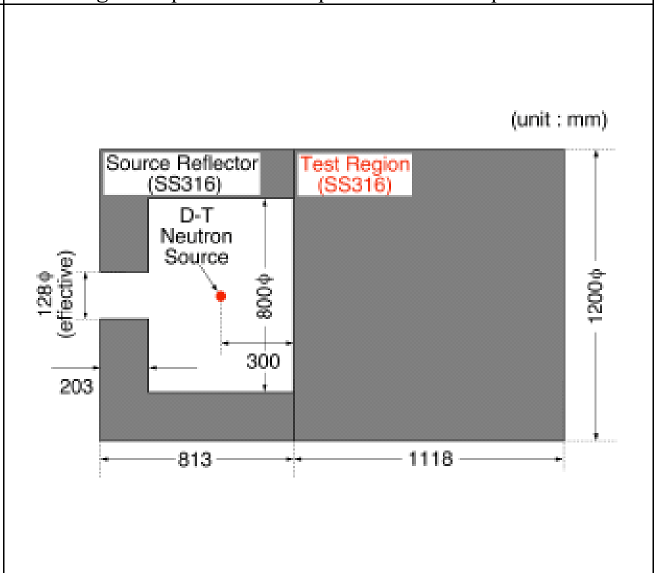

Fig.6. Experimental setup in SS316 *in situ* experiment.

4 Conclusions and recommendations

FENDL-2.1, the reference nuclear data library for ITER design, has been validated using the existing benchmark experiments carried out at the FNG 14 MeV neutron generator at ENEA Frascati and the FNS facility in Japan Atomic Energy Agency. These experiments investigate a wide spectrum of nuclear relevant issues for ITER, including those related to shielding blankets (with and without streaming paths), breeding blanket, vessel, magnets, and divertor. A comparison with previous FENDL version and with the latest nuclear data libraries, JENDL-3.3, JEFF-3.1, ENDF/B-VII.0, is discussed as well.

From these results, the following conclusions are obtained.

1) Li-6, -7 : FENDL-2.1 and all libraries provide good results, in agreement with experiments within total uncertainties.
2) Be-9 : All libraries cause overestimation of low energy neutrons in pure Be assembly. On the other hand, the slow neutron flux is underestimated by about 20% in the HCPB mock-up, where the 5.5-cm-thick Be layer is contained within two layers of neutron absorber (Li_2CO_3).
3) C-12 : FENDL-2.1 and JENDL-3.3 provide better results than ENDF/B-VII.0 and JEFF-3.1.
4) N-14 : FENDL-2.1, JENDL-3.3, ENDF/B-VII.0 and JEFF-3.1 provide results in agreement with experiments, although the angular distributions of some reactions may have some problems.
5) 0-16 : FENDL-2.1, JENDL-3.3, ENDF/B-VII.0 and JEFF-3.1 provide good results, but the angular distributions of some reactions may have some problems.

6) Si-28, 29, 30 : FENDL-2.1 causes the underestimation of neutron flux above 10 MeV. JENDL-3.3 and JEFF-3.1 provide results in better agreement with experiments.
7) V-51: FENDL-1.0 is better than FENDL-2.1. JEFF-3.1 is the best. All the libraries cause underestimation for low energy neutron flux.
8) Fe-54, 56, 57, 58 : FENDL-2.1 provide satisfactory results, although it underestimates the neutron flux at energies > 10 MeV. JEFF-3.1 performs slightly better. ENDF/B-VII also performs better except for the angular distribution of the elastic scattering in ^{56}Fe. (JENDL-3.3 has a problem in ^{57}Fe, but it will be revised in JENDL-4).
9) SS316 : All the libraries for nuclei included in SS316 seem to be good except for molybdenum, which may have some problems for low energy neutrons.
10) SS/Water: The fast neutron fluxes are underestimated by FENDL-2.1 by up to about 10% at 40 cm depth, and by 25-30% at about 1 m depth (outside uncertainty). The gamma ray flux is also underestimated by ~20% at 1 m depth, while a better agreement is found for the thermal neutron flux (within ±10% uncertainty). No significant differences are found between FENDL-1.0 (main materials in stainless steel from ENDF/B-VI), FENDL-2.0 and FENDL-2.1.
11) Cu-63, 65 : All the libraries provide results in disagreement with the experiments, particularly for low energy neutrons.
12) W-182, 183, 184, 186: FENDL-2.1 shows underestimation of the reaction rate of the ^{186}W(n,γ)^{187}W reaction and of ^{197}Au(n,γ)^{198}Au reaction.
13) Pb-206, 208, 207 : FENDL-2.1, JEFF-3.1 and ENDF/B-VII.0 provide results in agreement with the experiments.

It is therefore recommended that Si, Cu, W and Fe data should be improved.

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