Iron Neutron Data Benchmarking for 14 MeV Source

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Abstract. Iron neutron data benchmarking is presented. The iron is widely used as constructional and shields material for nuclear fusion and fission systems. The Iron cross section data from FENDL/MG-1.0 and VITAMIN-B6 were tested by benchmark experiment of integral neutronics' experiment collection implemented by the IAEA Nuclear Data Section. A discouraging bad consistency of calculated and measured results has been obtained for the neutron leakage from iron media. While the discrepancy for the FENDL/MG-1.0 is especially significant in the energy range 0.5-10 MeV for neutron transmission through media with thickness greater than 20 cm, for the VITAMIN-B6 the most inconsistency was obtained in the vicinity of the fusion peak for neutron transmission through lower thickness media. The presented results demonstrate that the tested Iron muligroup cross section data have to be significantly improved so that the neutron transport calculation results to be reliable in order to warrant nuclear safety requirements.

1. Introduction

The IAEA Nuclear Data Section has implemented a computerized collection of data as integral neutronic experiments ("benchmark experiments") suitable to test libraries of evaluated fusion relevant nuclear data. The iron is widely used as constructional and shields material for nuclear fusion and fission systems. The Oyama Y., and Maekawa F. benchmark [1] presenting the results of the 14 MeV neutron transmission through iron slabs of thickness 5, 20, 40 and 60 cm was used for testing of the iron cross section data from FENDL/MG-1.0 and VITAMIN-B6 [2] libraries. Both libraries have the same fine-group VITAMIN-J energy structure above 5.0435 eV (164 groups) and could be presumed as "problem independent" [2].

2. Calculations

Appropriate cross section data sets were especially generated for calculation of the benchmark by discrete ordinate transport codes. A FENDL cross section data set [3] from FENDL/MG-1.0 MATXS format library and the other data set FIOD241 from the VITAMIN-B6 master format library were prepared by corresponding auxiliary codes. Both data sets contain all iron nuclides (Fe54, Fe56, Fe57, Fe58) as well as oxygen data. The order of scattering cross section expansion is P5 for the FENDL and P7 for the FIOD241.

The angular neutron flux leakage spectra from the slabs were calculated by the two-dimensional codes DORT [4] and GRTUNCL [5]. The P5 (FENDL) and P7 (FIOD241) order of scattering expansion and S16 directional quadrature set were applied in the transport calculations. The binary DORT scalar flux and GRTUNCL first collision source output files were processed by the especially created code ANGAVE [3]. The code ANGAVE summarizes uncollided (from GRTUNCL output file) and collided fluxes (from DORT output file) and simulates the space averaging properties of detector response to obtain the values measured in the experiment.

3. Results

The experimental and calculated results of angular neutron flux leakage spectra for different slab thickness and scattering angles are partly presented in Figures 1-6.

As it is seen the consistency of calculated (FENDL) and measured results is very bad for neutron leakage from slabs with thickness greater than 20 cm and especially in the energy range 0.5-10 MeV (Fig. 1-5). This inconsistency was noted previously [3] too. The FIOD241 data set application in this case considerably improved the consistency between the calculation and experimental results.

In the energy range above 10 MeV, the FIOD241 neutron cross section data application results in some additional extension or even oscillations of the spectrum for the energy region close to the 14 MeV peak, while in the experimental data there is no similar spectral dependence for this energy region. This calculation spectrum has such behavior especially in the case when the 14 MeV peak impacts considerably the leakage spectra, i.e. for neutron transmission through slabs of smaller thickness (Fig. 6).

The obtained results show that the FIOD241 data set could not be applied as a problem independent despite of its fine-group structure. The lack of fusion peak weighting application for the VITAMIN-B6 multigroup master library generation [2] could be a possible reason for this discrepancy. The fusion peak has to be taken into account when fusion application multigroup cross section libraries are generated from ENDF/B-VI evaluated data.

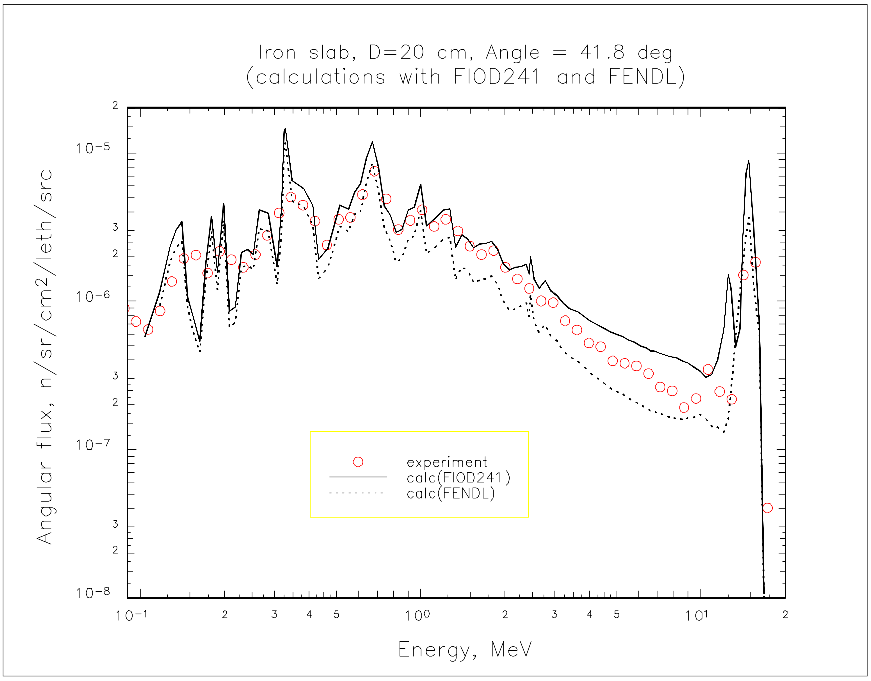

Fig. 1. DORT calculation (FIOD241, FENDL) for FNS Fe slab (20 cm, 41.8 °) TOF-experiment.

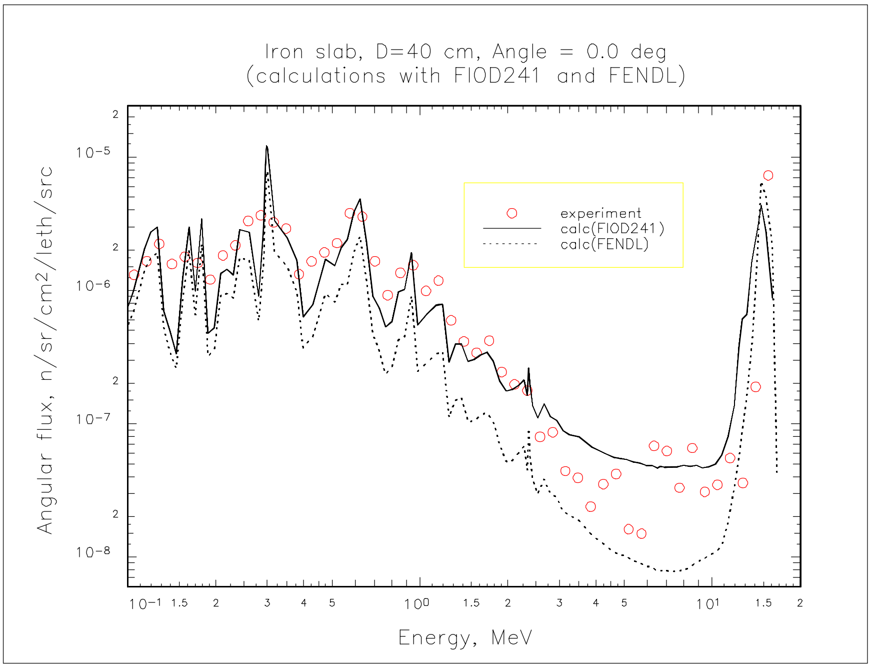

Fig. 2. DORT calculation (FIOD241 and FENDL) for FNS Fe slab (40 cm, 0 °) TOF-experiment.

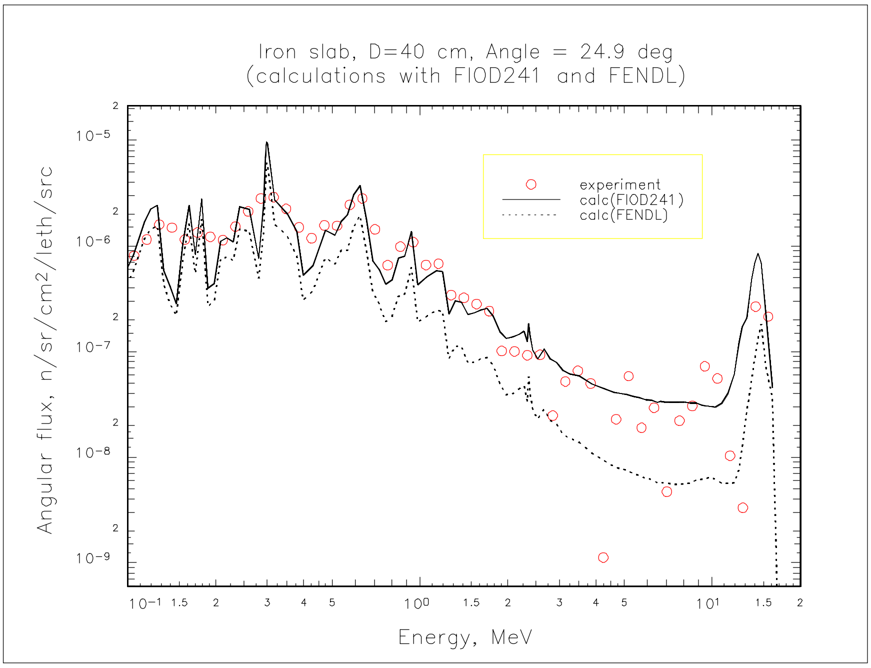

Fig. 3. DORT calculation (FIOD241, FENDL) for FNS Fe slab (40 cm, 24.9 °) TOF-experiment.

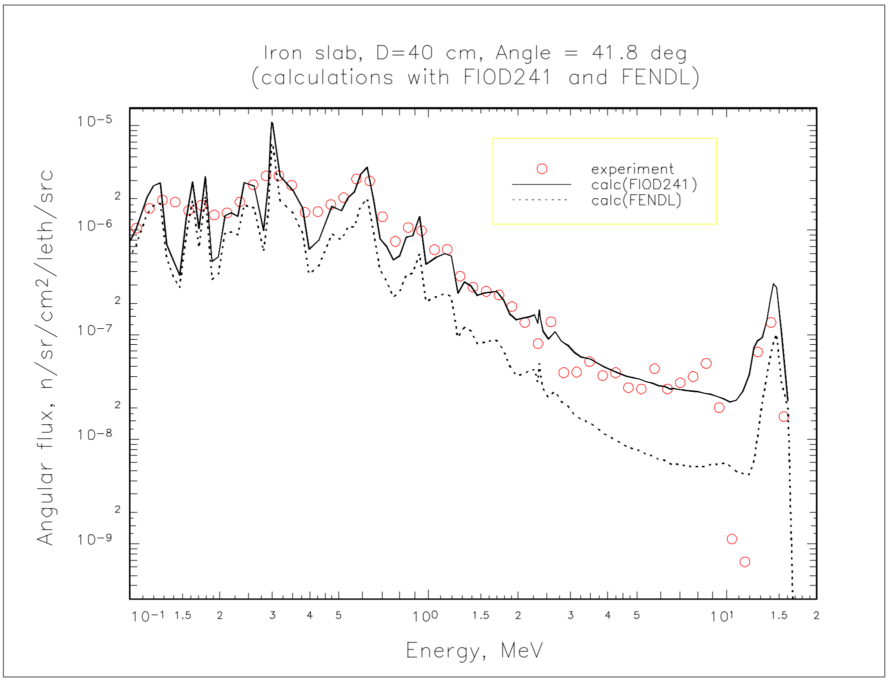

Fig. 4. DORT calculation (FIOD241, FENDL) for FNS Fe slab (40 cm, 41.8 °) TOF-experiment.

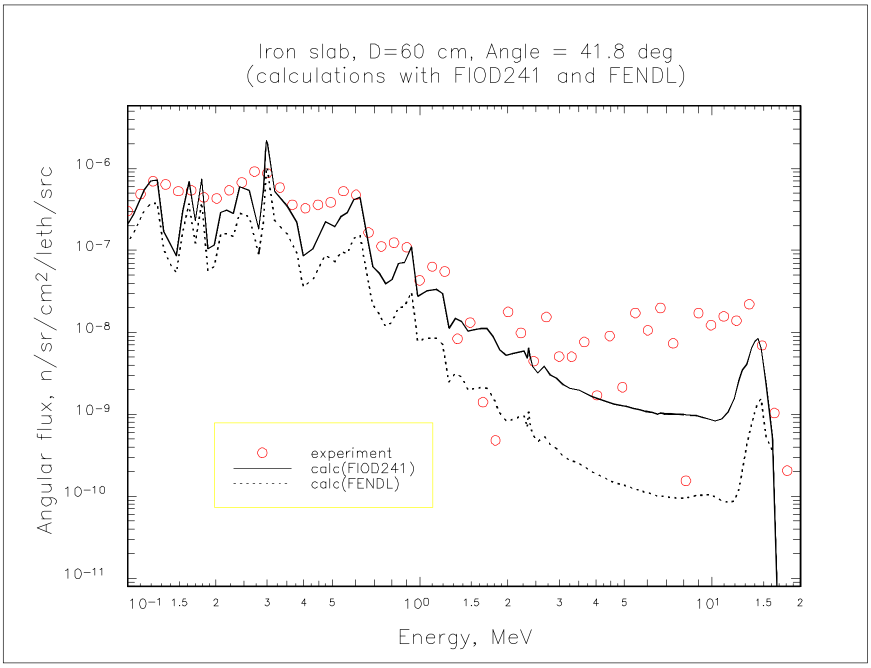

Fig. 5. DORT calculation (FIOD241, FENDL) for FNS Fe slab (60 cm, 41.8 °) TOF-experiment.

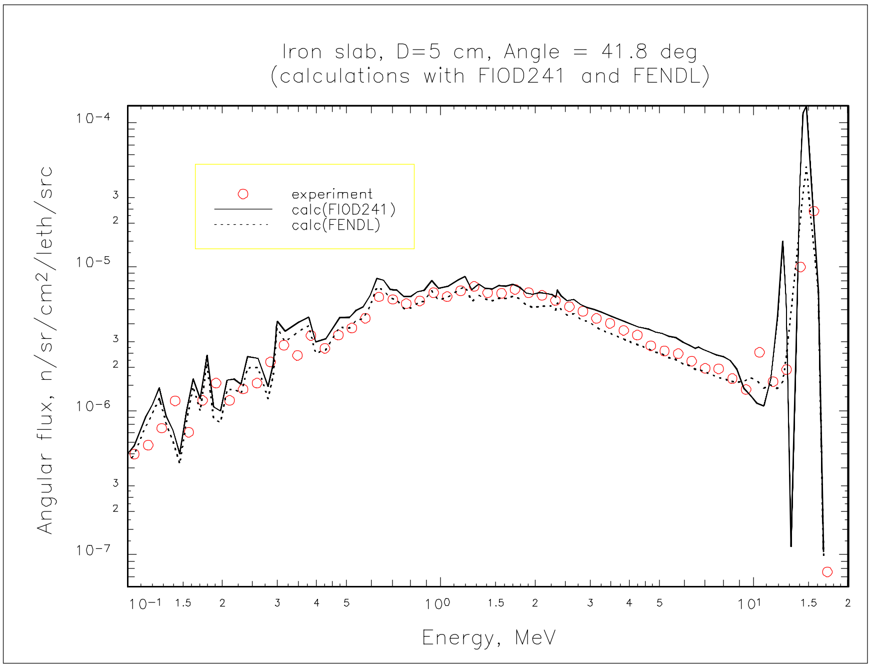

Fig. 6. DORT calculation (FIOD241and FENDL) for FNS Fe slab (5 cm, 41.8 °) TOF-experiment.

4. Conclusion

The multigroup Iron cross sections from FENDL/MG-1.0 and VITAMIN-B6 libraries have to be significantly improved in order to warrant reliable neutron transport calculation results in accordance to nuclear safety requirements.

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