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Transport, Sensitivity and Uncertainty Analysis of FNG 14 MeV Neutron Bulk Shield Experiment

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Bulk Shield experiment representing a mock-up of the ITER inboard blanket and vacuum vessel was completed in 1997 at the 14 MeV Frascati Neutron Generator (FNG) at ENEA Frascati. The objective of the experiment was to verify the calculational methods and nuclear data files used in the design and shielding calculations for fusion reactor. The benchmark experiment was analysed using the discrete ordinates transport code DORT, and the sensitivity and uncertainty analysis code package SUSD3D developed within the EFF project. Neutron cross-sections taken from FENDL-1and FENDL-2 evaluations were used in the transport calculation. The results were compared with the measured reaction rates and with those calculated by the Monte Carlo code MCNP4A using FENDL-1. Good agreement between MCNP and DORT results was observed. No significant difference between FENDL-1 and FENDL-2 results was found, FENDL-2 performing slightly better. The transport calculations were combined with the sensitivity and uncertainty analysis whose application has proven to be very useful in order to obtain information on the importance of various nuclear data in the neutron transport, to explain the discrepancies between the calculation and the measurement, or at least to discard some possible reasons for the discrepancy. The sensitivity analysis coupled with the covariance matrices can guide future nuclear data research and development.

Keywords: fusion, benchmark experiment, transport calculation, discrete ordinates method, cross-sections, sensitivity and uncertainty analysis.

I. Introduction

The recommended way to determine the quality and merit of the data files, including cross sections, radiation sources, etc., as well as of calculational procedures is to test them against some well defined benchmark experiments and shield design problems. The advantage of benchmarks is that the uncertainties, other than those due to the nuclear data, are reduced considerably. They can thus provide indications on needed nuclear data adjustments more accurately, and can serve as test of the computational methods and their ability to meet required standards and safety regulations.

In recent years important progress has been made concerning discrete ordinates and Monte Carlo radiation transport codes and new cross-section evaluations became available. At the same time the testing of these new evaluations required additional experimental work in order to determine if the data can fulfil the new reactor design requirements. Fusion analysis is a typical example where such new needs arise.

The objective of the FNG 14 MeV neutron bulk shield benchmark experiment performed at ENEA Frascati(1) was to verify the calculational methods and nuclear data files used in the design and shielding calculations of the fusion reactor, in particular of the ITER inboard blanket and vacuum vessel (2). The shielding block was made of plates of stainless steel and Perspex of the overall thickness of 95.2 cm. ^{93}Nb(n,2n), ^{58}Ni(n,2n), ^{27}Al(n,α), ^{56}Fe(n,p), ^{58}Ni(n,p), ^{115}In(n,n'), ^{197}Au(n,γ), and ^{55}Mn(n,γ) reaction rates, nuclear heating and fission rates were measured at various positions in the mock-up.

II. Transport Calculations

FNG bulk shield experiment was analysed in the past using the MCNP4A code and the FENDL-1 cross-section library. The results are described in (1). The neutron transport calculations was repeated using this time the two-dimensional discrete ordinates code DORT (3), with the objective to perform in addition the sensitivity and uncertainty analysis.

Several sets of cross-sections were used for the comparison. Two sets of multigroup cross-sections based on FENDL-1 and FENDL-2 evaluations were produced, taking into account the actual self-shielding effects. The starting point were the cross-section data from FENDL-1 and FENDL-2 evaluations, processed at the IAEA using NJOY to obtain 175 neutron/42 gamma cross sections in MATXS format. The processing of the latest ^{56}Fe EFF-3.1 data was performed at IJS using the same procedure. The TRANSX-2.15 code was then used to obtain self-shielded cross-section sets appropriate to the specific configuration of the benchmark.

In the stainless steel the sigma-0 values of around 5 (4 - 5) barn/atom were obtained for ^{56}Fe. Two cross-section sets were obtained, corresponding to the FENDL-1 and FENDL-2 evaluations, both processed in the 175 VITAMIN-J group structure.

The mock-up model was described in R-Z geometry with 24 radial and 117 axial intervals. S-16 and P-5 approximations were used in the DORT calculations.

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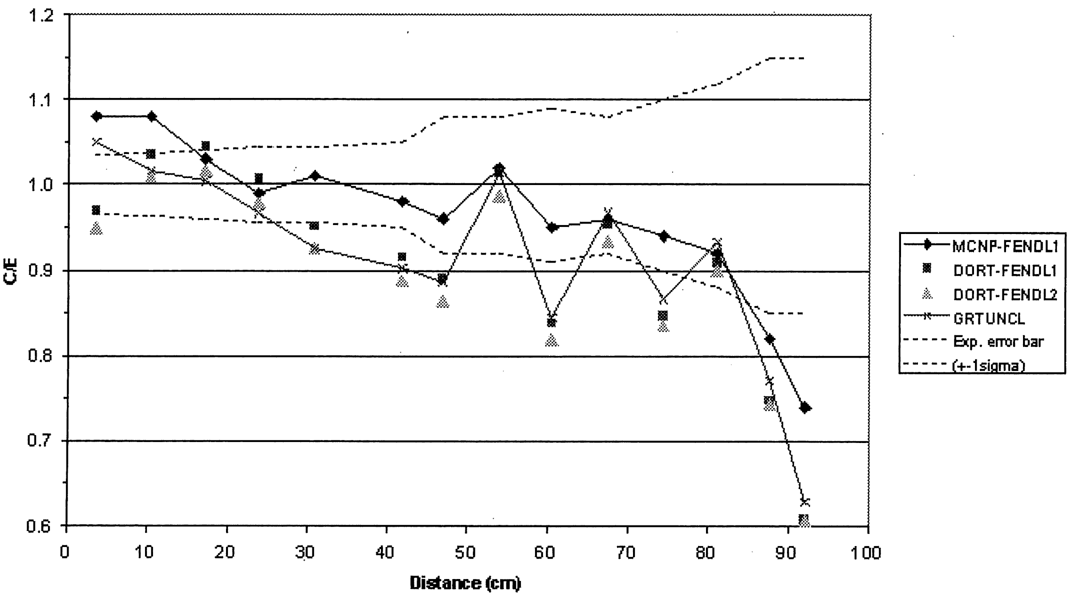

Fig. 1 C/E values for the Ni-58(n,p) reaction rates corresponding to the calculations performed with MCNP/4A (FENDL-1 cross-sections), and with DORT using FENDL-1 and -2 cross-sections. GRTUNCL represents the DORT calculation using the first collision source from the GRTUNCL code.

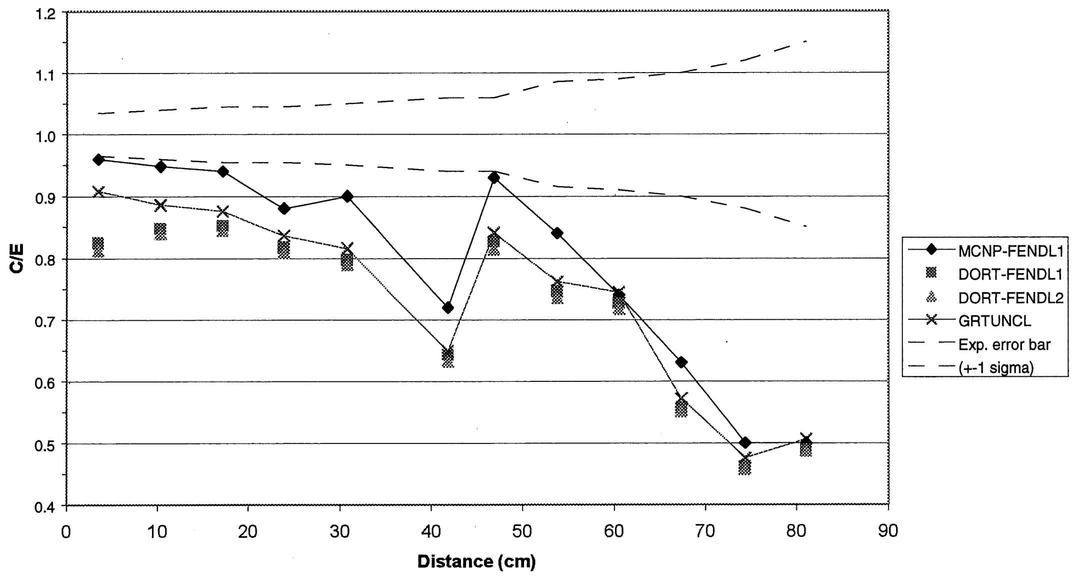

Fig. 2 C/E values for the In-115(n, n') reaction rates corresponding to the calculations performed with the MCNP/4A and DORT codes. FENDL 1 and -2 cross-sections were used. GRTUNCL represents the DORT calculation using the first collision source from the GRTUNCL code.

Ray effect problems were observed due to the source located in the low-scattering media (air). Satisfactory solution was found when the first central radial interval was increased to about 3 cm. Due to this relatively large interval the gradients were of course neglected, which results in slight underestimation of the reaction rates in the front of the shield at the first measurement location (< ~ 15 cm). Although we considered the results satisfactory, another calculation using first collision source method was performed. Uncollided and first collision sources were calculated by the GRTUNCL code.

The response functions taken from the IRDF-90.2 file were used to derive the reaction rates from the calculated neutron fluxes. Examples of the C/E values for the ^{58}Ni(n,p) and ^{115}In(n,n') reaction rates at various depths in the bulk shield are presented in **Figs. 1** and **2**, compared to those calculated by the MCNP4A code using FENDL-1 cross-sections. The agreement between the DORT and MCNP results as well as with the measurement is good, generally within 20 %. Exception are the ^{115}In(n,n') reaction rates, where both for DORT and MCNP calculations the differences of up to a factor of 2 with respect to the experiment were observed. As expected the first collision source calculation indeed improves the C/E values at the

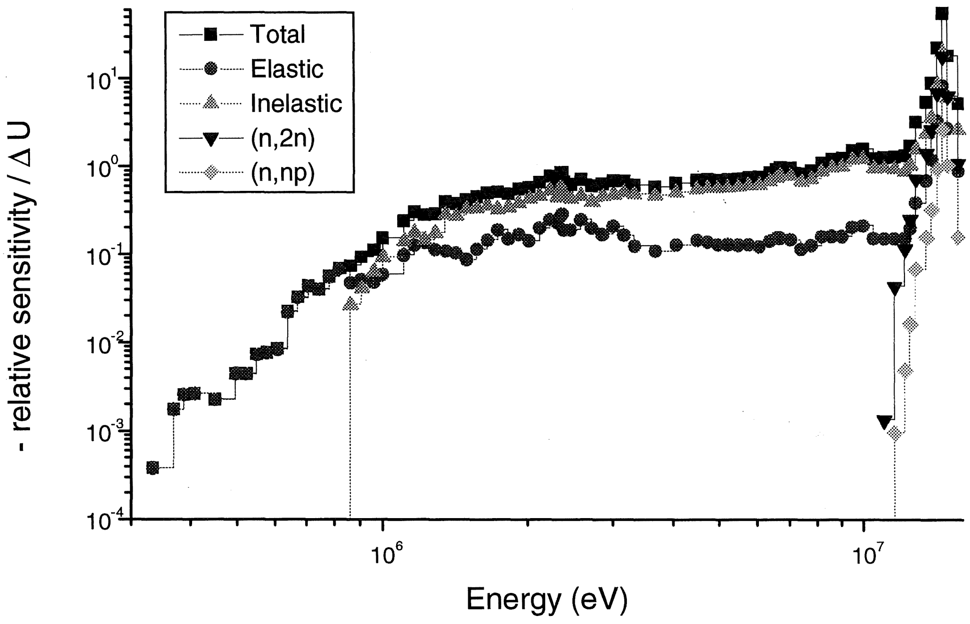

Fig. 3 Sensitivity of In - 115(n,n') reaction rate at z=81cm to different terms of iron cross-sections. The sensitivities are multiplied by (-1).

Table 1 Uncertainties of the detector responses due to the iron cross-section uncertainties. Comparison is done using covariance matrices from different evaluations (EFF-2, EFF-3 and ENDF/B-VI).

Detector	Distance (cm)	Uncertainty [%]		
		ENDF/B-VI	EFF-2	EFF-3
^{115}In(n,n')	81.1	16.8	10.3	5
^{93}Nb(n,2n)	81.1	21	15.3	7
^{55}Mn(n,γ)	81.1	12.3	8	
^{56}Fe(n,p)	74.4	19.1	11	6.4
^{115}In(n,n')	41.85	6.2	4.2	2.5
^{58}Ni(n,2n)	30.8	9.1	5.9	3.6
^{93}Nb(n,2n)	10.32	3	2.5	1.2

locations close to the point source, but does not influence much the results beyond ~15 cm in the block.

For the sensitivity and uncertainty calculations the transport calculations in the adjoint mode are needed as well. They were performed using the same calculational model (with adjoint source, equal to the detector response function) for the following detectors: ^{115}In(n,n'), ^{93}Nb(n,2n), ^{55}Mn(n,g), ^{56}Fe(n,p), ^{58}Ni(n,2n). The distances between 10.32 and 81.1 cm in the shield were considered.

III. Sensitivity and Uncertainty Analysis

The transport methods are combined with sensitivity analysis in order to establish reliable safety margins for the measured and calculated values, as well as to determine to what extent the benchmark experiment is representative of the real nuclear reactor environment.

A procedure based on the SUSD3D code and the VITAMIN-J/COVA covariance matrix library was used [4-7]. The procedure has been extensively used in the sensitivity and uncertainty analysis for the pressure vessel dosimetry, as well as for several benchmark experiment analyses (ASPIS, VENUS-3) and fusion applications. This system can perform one-, two- and three-dimensional sensitivity and uncertainty analysis. The present version of the code can thus use angular flux or flux moment files produced by the discrete ordinates codes ANISN, DOT3.5, ONEDANT, TWODANT, THREEDANT, DORT and TORT. Use of the angular moment files instead of the bulky angular flux files produced by these codes reduces considerably the size of the files required and represents an acceptable approximation for the problem type analysed here.

Sensitivities of different reaction rates with respect to the cross-sections of the main elements constituting the block assembly were calculated for several locations in the bulk shield. Sensitivity to the neutron source description (isotropic vs. forward biased) was also studied.

An example of the results of the sensitivity analysis, based on the direct and adjoint DORT transport calculations described previously, is given on **Fig. 3**. The figure presents the sensitivity profiles with respect to the main iron cross-section components (total, elastic, inelastic, (n,2n), (n,np)).

The uncertainty estimations due to the cross-sections are presented in **Table 1**. Large differences of the estimations based

Table 2 Summary uncertainties in the detector responses due to the cross-section and the neutron source uncertainties (in %).

Detector		Cross-sections						Sum x-sect.	SAD	Resp. funct.	Neut. source	**TOTAL**
Reaction	Posit. (cm)	Fe EFF-3/2	Cr	Ni	C	H	Mn	EFF-3 /EFF-2	Fe-elastic		Magn. /ang. distr.	**EFF-3 /EFF-2**
$^{55}Mn(n,\gamma)$	81.1	-/8.0	1.9	1.3	1.3	0.9	0.8	-/10.7	2.0	8.0	2 / 1	**-/14**
$^{115}In(n,n')$	10.3	0.3/2.0	0.2	0.1	0.1	0.02	0.01	0.4/2.0	0.2	2.0	2 / 1	**3/4**
	41.85	2.5/4.2	1.1	0.8	0.7	0.5	0.2	3.0/4.5	1.1	2.0	2 / 1	**4/6**
	81.1	5/10.3	2.6	1.7	1.7	0.9	0.6	6.2/10.9	2.4	2.0	2 / 1	**7/12**
$^{56}Fe(n,p)$	74.4	6.4/11	3.1	2.1	2.2	2.1	0.6	8.0/11.9	2.8	1.3	2 / 1	**9/13**
$^{93}Nb(n,2n)$	10.3	1.2/2.5	-	-	-	-	-	1/3	-	0.9	2 / 1	**3/4**
	81.1	7/15.3	3.6	2.4	2.7	0.8	0.7	7.3/16.2	3.1	0.9	2 / 1	**8/17**
$^{58}Ni(n,2n)$	30.8	3.6/5.9	1.7	1.2	1.5	0.4	0.3	4.4/6.4	1.6	1.6	2 / 1	**6/7**
$^{197}Au(n,\gamma)$										16.8	2 / 1	
$^{58}Ni(n,p)$										5.5	2 / 1	
$^{27}Al(n,\alpha)$										0.5	2 / 1	

on different covariance matrix evaluations for iron (EFF-2.4, EFF-3, ENDF/B-VI) can be observed.

Table 2 summaries the uncertainties due to the nuclear data and the neutron source for the selected detectors. The table shows also the uncertainties of the detector responses due to the secondary angular distribution (SAD) uncertainties of iron elastic cross-section. Using EFF-3 covariance data it follows that the contribution of SAD uncertainty is superior to the "normal" cross-section uncertainty in the case of the elastic scattering on iron.

Detector response function covariances used to evaluate the uncertainties in the response functions were taken from the IRDF 90.2 file [(7)].

The angular and energy dependence of the neutron source distribution from T(d,n)α reaction in FNG was sampled by MCNP4B code using special subroutine. Several emission angles (from 5 to 60 degrees) and energies were considered. The neutron source was found to be slightly energy-angle dependant. On the other hand the neutron source was described in DORT as isotropic, depending on energy only. To evaluate the effect of this approximation the reaction rates were calculated using the neutron source energy distribution corresponding to different scattering angles. Making use of the adjoint flux, we found that the effect of the angular dependence of the source is small, of the order of ~1 %.

The uncertainty linked to the source intensity was evaluated to about 2 %, and is due to the different methods of its measurement.

IV. Conclusions

Good agreement between the DORT discrete ordinates transport calculations and the measured reaction rates, as well as the values calculated by the MCNP4A M/C code, was found. In general the calculations agree with the measurements within ~20 %, except for $^{115}In(n,n')$ where the C/E values as low as 0.5 were found. No significant differences between the calculations performed using FENDL-1 and FENDL-2 cross-sections were observed. Use of the first collision source method, although not really indispensable, was found to improve the DORT results at the positions close to the neutron source.

The sensitivity and uncertainty studies, which were started at the same time, provide useful supplementary information on the benchmark experiment and measured quantities. The uncertainties due to the source distributions were estimated, and the sensitivity and uncertainty studies with respect to the transport cross-sections and detector response functions were performed.

Large difference between uncertainty estimations based on EFF and ENDF/B-VI cross-section covariance matrices was found, EFF-3 based uncertainties being by about a factor of 3 lower comparing to those based on ENDF/B-VI, and by a factor of ~2 lower than those based on EFF-2.

The total uncertainty in the calculated reaction rates due to nuclear data and neutron source is of the order of ~ 6 % at 10 cm, and of the order of ~10 – 20 % at 80 cm (depending on the detector and the cross-section covariance matrices used) in the bulk shield.

Comparing these uncertainties with the actually observed C/E values the quality of the covariance information can be judged and possibly explain the differences between the measurements and the calculations. For most detector responses, except for $^{115}In(n,n')$, the observed C/E values are situated within the measured and calculated uncertainty intervals.

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