# Advancement of Nuclear Analysis Method for ITER

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**Abstract**. Advancement of nuclear analyses, which means improvement and validation of radiation transport codes and nuclear data libraries, is a key issue for ITER licensing. Thus we have developed an automatic conversion system in order to easily make precise geometry input data of radiation transport codes from CAD data even for complicated ITER structures. We also applied this system to the ITER 40 degree benchmark model for validation and compared our results with those by other ITER participating parities (China, EU, US). We also performed a narrow slit streaming experiment for analysis validation on streaming effects at JAEA/FNS and verified the prediction accuracy of neutron transport calculations with the MCNP4C, TORT and Attila codes.

## 1. Introduction

Nuclear analysis is carried out with neutron and gamma transport code and nuclear data libraries. The Monte Carlo code MCNP [1] and fusion evaluated nuclear data library FENDL [2] have been established as the standard transport code and nuclear data library for ITER, respectively. Fusion reactors such as ITER have much more complicated structures than fission reactors. The MCNP code can treat such complicated geometry precisely, but it takes incredibly much time to make precise geometry input data for MCNP, which is a large problem to be solved for ITER licensing. Therefore we developed an automatic conversion system from three-dimensional CAD drawing data to geometry input data under the ITER/ITA Task.

Slits between the vacuum vessel port walls and the port plugs in ITER provide possible radiation streaming paths. The dimensions of the slits are ~2 cm in width and ~200 cm in depth and offset geometries are arranged on the middle of slits to mitigate the streaming effect. Experimental verification of nuclear analyses for such slit streaming is not enough. Thus we have carried out a narrow slit streaming experiment at the Fusion Neutronics Source (FNS) facility in JAEA under the ITER/ITA Task.

## 2. Development of automatic conversion system from CAD data to MCNP input data

### 2.1. Automatic conversion system

MCNP requires void data. No void is expressed in CAD data though there are solid region data such as blankets in CAD data. This system consists of a void creation program (CrtVoid) and a conversion program (GEOMIT) from CAD drawing data to geometry input data of MCNP. First CrtVoid creates void region data. The void region data is very large and complicated geometry. CrtVoid automatically divides the overall region to many small cubes, and the void region data can be created in each cube. Next GEOMIT generates surface data from CAD data including the void data generated with CrtVoid. These surface data are connected, and cell data are generated. By using this system, geometry data in MCNP inputs are automatically produced, which leads drastic time- and labor saving.

## 2.2 Validation of automatic conversion system

We applied this system to the ITER 40 degree benchmark model as shown in Fig. 1 for validation. The calculation with the generated input gave adequate neutron flux and nuclear heating distributions as shown in Fig. 2. This figure also includes results with similar systems developed by other ITER participating parities (China, EU, US) for reference [3].
We also applied this system to nuclear analysis of simplified Japanese water-cooled solid breeder test blanket module (TBM) for ITER. Fig. 3 shows cross sectional view of the CAD model and the contour map of calculated neutron flux in the TBM normalized by neutron wall loading of 1 MW/m$^2$.

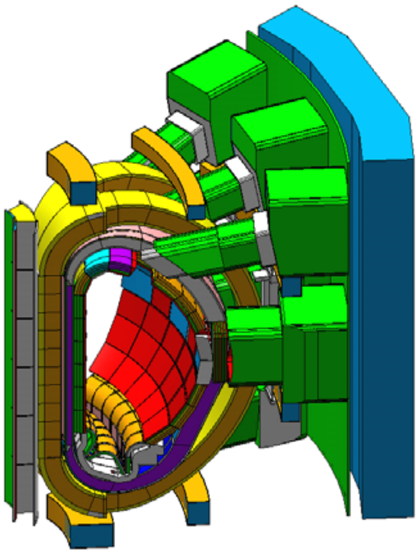

*FIG.1. ITER 40 degree benchmark model.*

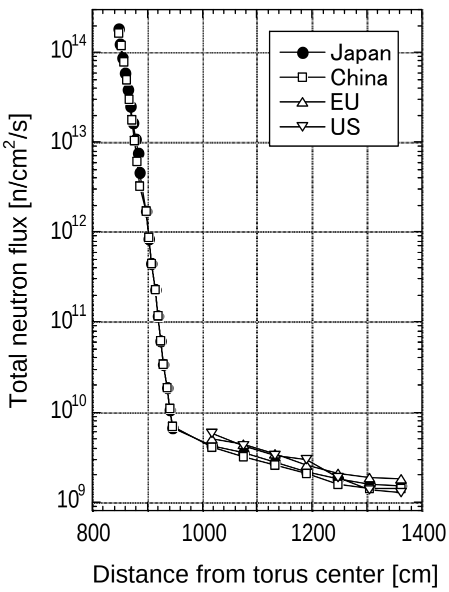


*FIG.2. Neutron flux in and behind port.*

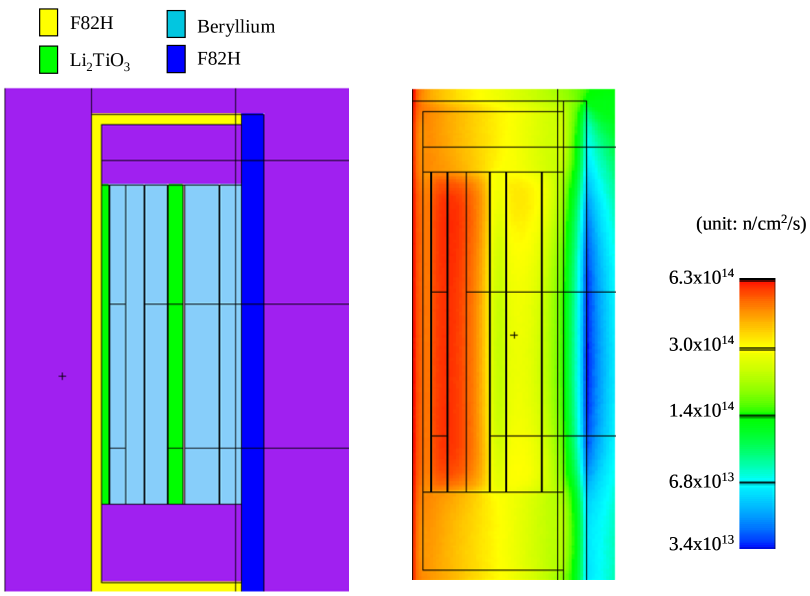


*FIG. 3. CAD model (left) and the contour map of calculated neutron flux in TBM (right).*

## 3. Narrow slit streaming experiment

### 3.1. Experiment

Horizontal and vertical cross sectional views between the experimental assembly and the D-T neutron source are shown in Fig. 4. This experimental assembly has a slit of 2 cm in width, 195 cm in depth and 3-cm offset at 56-cm depth from the surface, which was constructed with iron blocks. It was located 20cm from the D-T neutron source. D-T neutrons were generated by bombarding 350-keV and 1-mA deuteron beam to a tritiated titanium target. The intensity was about 1.5 x $10^{11}$ neutrons/sec.
U-238 and U-235 Micro-Fission Chambers (MFC) and Nb, Al and In activation foils were utilized as detectors to measure fission rates and reaction rates along the slit as a function of the depth. The $^{238}$U(n,fission), $^{93}$Nb(n,2n)$^{92m}$Nb, $^{27}$Al(n,$\alpha$)$^{24}$Na and $^{115}$In(n,n')$^{115m}$In reasctions are sensitive to fast neutrons. Therefore, the reaction and fission rates show the relative MeV neutron fluence. On the other hand, the $^{235}$U(n,fission) reaction is sensitive to slow neutrons, especially neutrons from thermal energy to 1 keV, and the U-235 fission rate is a good index of slow neutrons. The experimental errors of the fission and reaction rates were within 10 %.

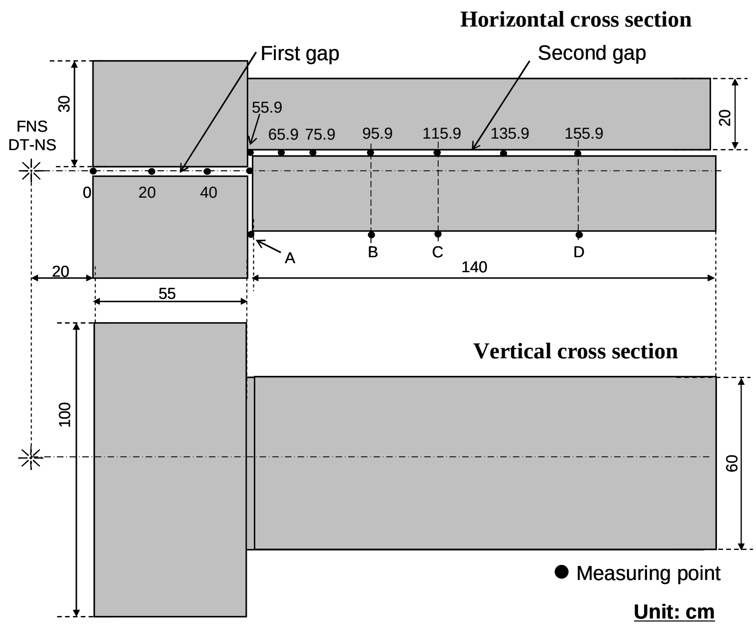


*FIG.4. Horizontal and vertical cross sectional views between experimental iron assembly and D-T neutron source.*

### 3.2. Analysis

This experiment was analyzed by using the Monte Carlo code MCNP-4C [1] with the nuclear data libraries FENDL-2.1 [2] and other recently released nuclear data libraries (JENDL-3.3, JEFF-3.1 and ENDF/B-VII.0) [4-6]. The importance was increased by a factor of 2 every 5cm. The tally of 2 x 2 x 2 $cm^3$ was set along the slit.
It is generally pointed out that Sn codes cannot always represent radiation streaming behavior such as this streaming experiment well. In order to investigate this issue, we performed analyses with the Sn codes TORT [7] and Attila [8]. A multigroup library of 175 neutron groups was generated from the matxs file FENDL/MG-2.1 with the TRANSX2.15 [9] code.

This library was reduced to the 21 neutron groups in the Attila. The $P_5S_{30}$ approximation and last collided source calculation with forward bias were used in Attila, while $P_5$U315(upward biased angular quadrature set) approximation was adopted in TORT. The total mesh number in the Attila and TORT calculations was around 24,000.
The data for $^{235}$U(n,fission), $^{238}$U(n,fission), $^{93}$Nb(n,2n)$^{92m}$Nb, $^{27}$Al(n,α)$^{24}$Na and $^{115}$In(n,n')$^{115m}$In reaction cross sections were cited from JENDL Dosimetry File 91 [10].

### 3.3 Comparison between calculation and experiment

The calculation results with MCNP-4C agreed well with the measured ones, not depending on nuclear data libraries as shown in Fig. 5.
Figure 6 shows the measured and calculated U-238 fission rate. The same results of activation foils, Nb and In, are also shown in Figs. 7 and 8, respectively. Furthermore, the U-235 fission rate is shown in Fig. 9. It is shown in Fig. 6 that the calculated U-238 fission rates up to 95.9 cm in depth are close to the measured one. Especially, the calculation with MCNP well agrees with the measurement. The calculated reaction rates with Nb and In also correspond with the measured ones up to 95.9 cm in depth. From the results, it is found that the evaluation of fast neutron transport by using MCNP, TORT and Attila is adequate up to about 100 cm in depth.

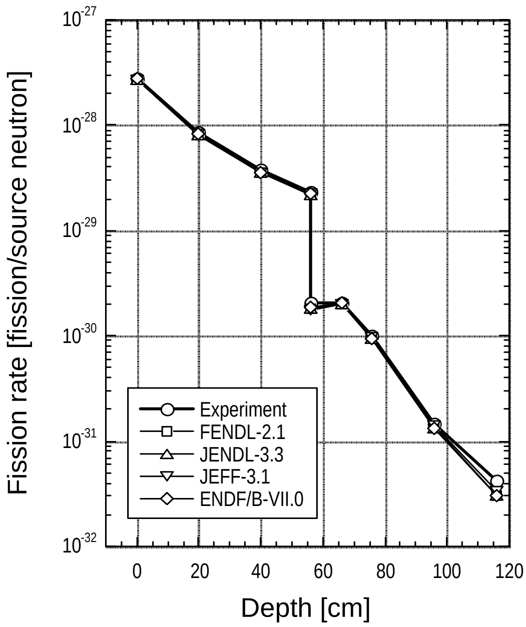


*FIG. 5. Calculations of fission rate distribution of U-238 with l nuclear data.*

It is noted that the considerable underestimation appears at the region between 136 cm and 176 cm in depth. The measured U-238 fission rates were about 5 x $10^{-32}$ at the points of B, C and D outside the assembly (see Fig. 4), which included lots of neutrons scattered in the experimental room. About one tenth of the number of fast neutrons near the points of B, C

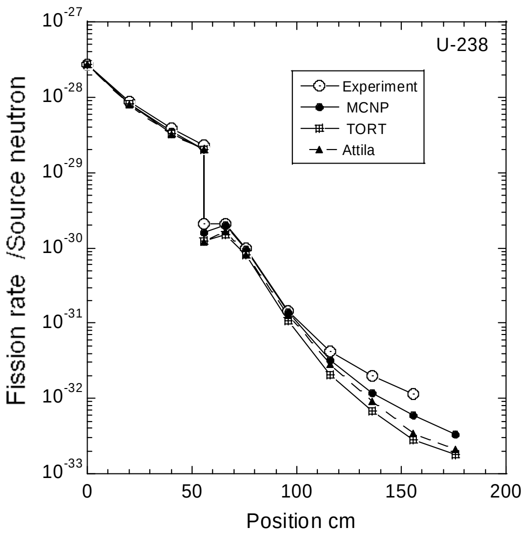


*FIG. 6.Measured and calculated $^{238}$U fission rates.*

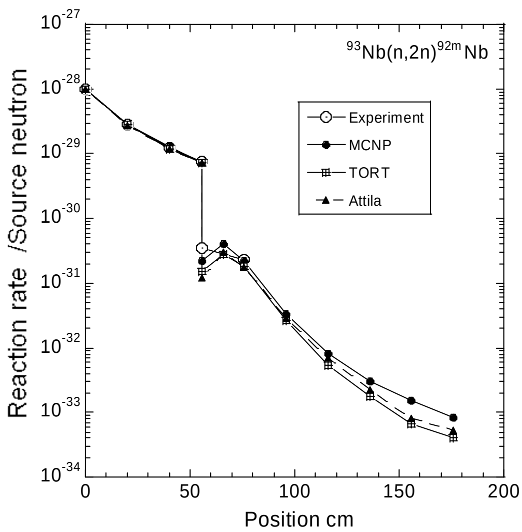


*FIG. 7. Measured and calculated $^{93}$Nb(n,2n)$^{92m}$Nb reaction rates.*

and/or D may penetrate to the area of slit through the 20 cm thick iron. Thus it is considered that the scattered neutrons cause the considerable underestimation at the deeper region than 115.9 cm.

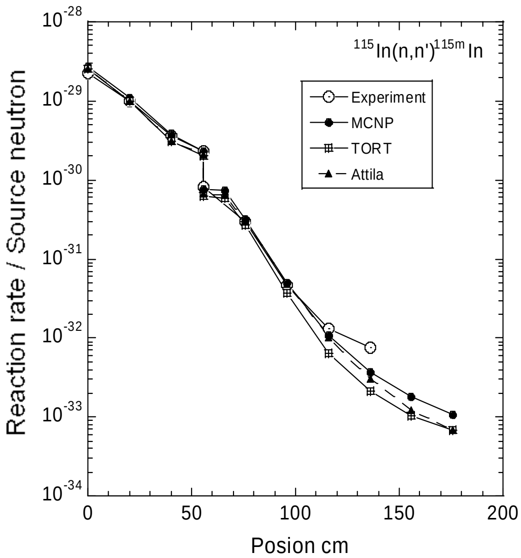


*FIG. 8. Measured and calculated $^{115}In(n,n')^{115m}In$ reaction rates.*

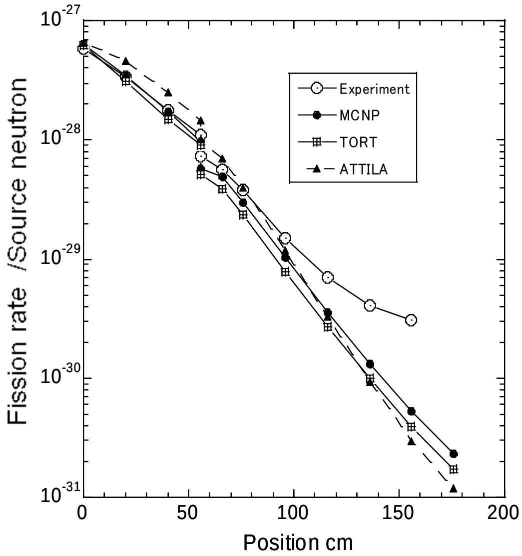


*FIG. 9. Measured and calculated $^{235}U$ fission rates.*

## 4. Summary

We developed an automatic conversion system from the three-dimensional CAD drawing data to MCNP input data. This system consists of a void creation program (CrtVoid) and a conversion program (GEOMIT) from CAD drawing data to MCNP input data. The developed system was applied to the ITER 40 degree sector model, and neutron fluxes and nuclear heating were calculated using the generated MCNP input data. The calculation results agreed well with those provided by other parties. Using by the automatic conversion system, we also successfully obtained neutron flux distribution in the TBM.

We have conducted a neutron streaming experiment with the JAEA/FNS neutron source and an experimental assembly modeled the slits of the boundaries between ITER vacuum vessel and equatorial port plugs. By using U-238 and U-235 micro fission chambers and some activation foils, we measured the depth profiles of fission rates and reaction rates along the slit and performed neutron transport calculations with the MCNP, TORT and Attila codes. From our measurements and calculations, the following facts were found: (1) It is confirmed that the present ITER nuclear analysis method can also give accurate results for such slit streaming.; (2) In case of such a narrow and deep slit structure, the calculation methods with MCNP, TORT and Attila codes and FENDL-2.1 is sufficient to predict fast neutron field inside the slit up to about 100 cm.; (3) By neutrons scattered in the experimental room, experimental background noise considerably increased at the deeper region than 100 cm.; (4) It was pointed out it was necessary to use upward biased sets and last collided source calculation in TORT and Attila.

**References**

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