Determination of the neutron irradiation field at the research reactor TRIGA Mainz

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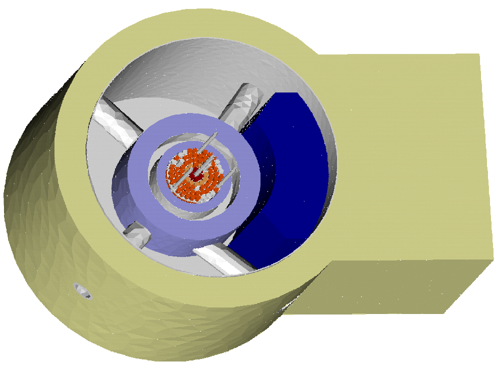
The thermal column of the research reactor TRIGA Mainz shall be reconstructed to allow the irradiation of an organ with thermal and epithermal neutrons to establish the Boron Neutron Capture Therapy (BNCT) at the reactor facility [1].

The basic characteristics of the radiation field in the thermal column like beam geometry, neutron and gamma ray energies and angular distributions, neutron flux and absorbed gamma and neutron doses must be determined in a reproducible way. Therefore, calculations are carried out using the General Monte Carlo N-Particle Transport Code (MCNP, Version 5) as well as the transport code ATTILA.

Attila is a 3D multi-group Discrete Ordinate particle transport code which was first developed at Los Alamos National Laboratory (LANL). It is used for neutral, charged and coupled particle transport including the capability to calculate eigenvalues for critical systems. ATTILA is designed to solve the linearized Boltzmann transport equation for a variety of radiation transport applications as burn up, shielding, criticality, radiation protection and dosimetry, radiography and medical physics.

The reactor model has been developed using MCNP as well as ATTILA (Fig. 1). Results of simulations are shown in Fig. 2.

Fig. 1 Computational model of the TRIGA reactor in Mainz using the ATTILA program.



The calculations are validated measuring the neutron and gamma flux in the middle channel of the thermal column using gold activation foils and Thermoluminiszenz dosimeter (TLD). Cadmium covered gold foils were used to separate the thermal and epithermal neutron flux. The irradiated foils were analysed using a standard high-purity germanium (HPGe) gamma spectrometry system (Canberra /Genie^{TM}). The system efficiencies based on calibrations of the spectrometer using the PTB standards and a mixed calibration source. The comparison between the calculations and measurements in the thermal column are shown in Fig.3.

The Attila and MCNP program will be used for further simulations of the TRIGA Mainz to calculate the optimal irradiation geometry for the explantated organ and the estimations of the different BNCT dose compounds.

Fig. 2: Neutron flux distribution at a horizontal cross section in n/(cm² s) of the TRIGA Mainz at a power of 100 kW calculated with ATTILA.

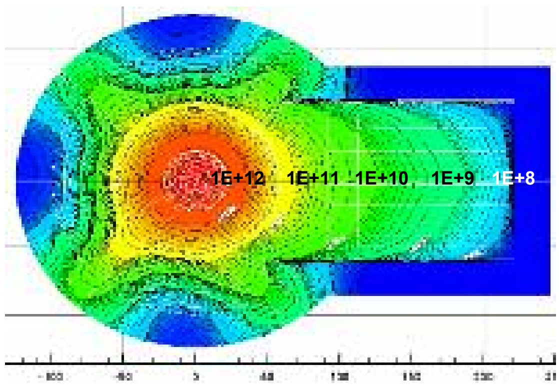
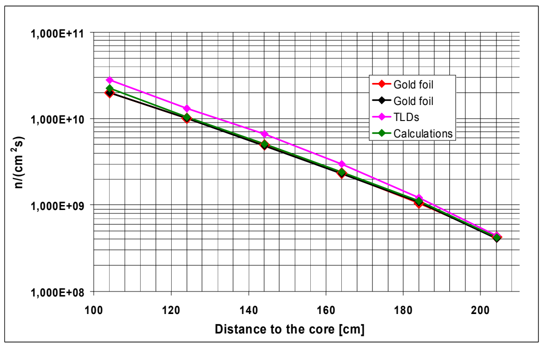


Fig. 3: Comparison between simulation and measurements using gold foils and TLD in the thermal column



References

[1] T. Pinelli, et al. *TAOrMINA: From the First Idea to the Application to the Human Liver", in: Research and Development in Neutron Capture Therapy.* Proceedings of the 10th Internatioal Congress on Neutron Capture Therapy, Monduzzi editore, Bologna pp. 1065-1072 (2002)

[2] MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, Los Alamos National Laboratory, (2003)

[3] ATTILA – 3D multi-group Sn particle transport code, Version 6, Transpire, Inc. (2005)