

MCNP AND VISUALIZATION OF NEUTRON FLUX AND POWER DISTRIBUTIONS

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ABSTRACT

The visualization of neutron flux and power distributions in two nuclear reactors (TRIGA type research reactor and typical PWR) and one thermonuclear reactor (tokamak type) are treated in the paper. The distributions are calculated with MCNP computer code and presented using Amira and Voxler software. The results in the form of figures are presented in the paper together with comments qualitatively explaining the figures. The remembrance of most of the people is better, if they visualize a process. Therefore a representation of the reactor and neutron transport parameters is a convenient modern educational tool for (nuclear power plant) operators, nuclear engineers, students and specialists involved in reactor operation and design. Visualization is applicable not only in education and training, but also as a tool for core and irradiation planning.

Key Words: MCNP, Monte Carlo, visualization, neutron flux distribution, particle tracks

1. INTRODUCTION

Modern Monte Carlo computer codes (e.g. MCNP [1]) for neutron transport allow calculation of detailed neutron flux and power distribution in complex geometries with resolution of ~1 mm. Moreover they enable the calculation of individual particle tracks, scattering and absorption events. With the use of advanced software for 3D visualization (e.g. Amira [2], Voxler [3], etc.) one can create and present neutron flux and power distribution in a 'user friendly' way convenient for educational purposes. One can view axial, radial or any other spatial distribution of the neutron flux and power distribution in a nuclear reactor from various perspectives and in various modalities of presentation. By visualizing the distribution of scattering and absorption events and individual particle tracks one can visualize neutron transport parameters (mean free path, diffusion length, macroscopic cross section, up-scattering, thermalization, etc.) from elementary point of view. In addition one can use special features of the code to visualize the geometry of the computational model.

The paper describes the calculation procedure to generate data for 3D visualization and explains how to combine MCNP and visualization tools to obtain interactive 3D figures. The procedure for presenting the components from the MCNP computational model will also be presented and explained. The results (neutron flux and power distributions, particle tracks in moderating media and geometry of computational model) are presented at the end.

In the paper, the visualization is presented for Jožef Stefan Institute (JSI) TRIGA reactor, for a typical PWR core, and for a typical toroidal fusion reactor - tokamak, however, it can be easily applied to any other reactor as well.

2. CALCULATION METHOD

MCNP computer code [1] was used in the calculations. MCNP is a general-purpose, continuous-energy, Monte Carlo transport code, capable of treating very complex three-dimensional configurations. The calculations reported in this paper were performed with version 5.1.40 of the code.

Three reactor cores were modeled; two fission reactors, a research and a power reactor, and a toroidal thermonuclear fusion reactor, tokamak.

2.1 Neutron flux and power distribution

The fission density and neutron flux distribution were calculated in the following way. A mesh of cells, in which the neutron flux was calculated, was superimposed on the reactor core. The fission density or any other reaction rate density is obtained by simply multiplying the tallied neutron flux by the corresponding reaction cross section. In MCNP a mesh of cells is applied by using the FMESH tally card. For example in TRIGA core calculation we used $500 \times 500 \times 50$ (length \times width \times height), that is 1.25×10^7 , cells. Use of larger number of cells was not possible due to limited amount of computer memory that could be used by the code and large statistical uncertainty due to small number of neutrons in each cell.

The output files containing information on neutron flux in individual cells of the mesh are usually quite large, on the order of several GB, if they are in ASCII format. Hence it is convenient to convert them to binary format and reduce their size for approximately 15 times. The data is then imported into one of the many codes for 3D visualization. Amira and Voxler were used to visualize 3 dimensional neutron flux and power distributions. We used the height field option to present the neutron and fission density distribution data. In this option the height (and the color) of the surface is proportional to the value of the parameter under investigation, e.g. neutron flux or power. The second option we used is the iso-surface, which plots a surface over all points with the same value of a parameter under investigation.

The particle tracks, scattering and absorption events were calculated using the PTRACK option in MCNP, which gives the information about the neutron tracks (energy, direction, location of the scattering, absorption and other events) and the results were plotted with Voxler software package.

2.2 Visualizing the MCNP geometry

In addition to the visualization of fluxes it is often desirable to plot the geometry of the computational model in the same figure. One of the unconventional options, however very effective, is to run the MCNP program to map the regions in space where individual components in the model are situated.

In order to export the shape of individual components, belonging to the MCNP geometry model, again we use the MCNP's capability of applying a 3D mesh over the geometry and tallying particle flux in the mesh cells. For this purpose the flux has to be made uniform in all cells,

belonging to the specific component, and be different – preferably equal to zero – in other parts of the geometry. We need to perform a run with the geometric model, which is to be visualized, and defining the source and material cross-sections in the model in such a way, that the flux is uniform in the desired components and zero elsewhere.

This can be done in several ways from which two will be described.

2.2.1 Using the source subroutine

The MCNP code includes a source subroutine which enables the user to define arbitrary source shapes. This option is usually used when the built in options of the general source are not sufficient. By rewriting the FORTRAN source subroutine it is possible to define sources of arbitrary shapes; with a small effort also a uniform volume neutron source, limited to the cells of interest can be modelled. Further the original model is slightly modified by changing the densities of the materials inside all defined cells to a high value, e.g. 10 g/cm^3 , in order to make the neutron mean free path short in comparison to the cell size. It is especially important that no void cells are present in the model and that all possible fissile nuclides are removed from it. The starting energy of neutrons in the source subroutine should be relatively low, e.g. 10 eV and a lower energy cut-off (defined in the input file) should be slightly below the source energy, e.g. 9.9 eV. By setting the energy cut-off just below the source energy, the neutron history is practically terminated after its first scattering event.

Subsequently the run is performed with the input file prepared with the changed materials, lower neutron energy cut-off and inclusion of the desired mesh tally the run is performed. The produced mesh tally is then exported into a visualization program, where the flux (which has the shape of the component of interest) is visualized by using one of the many visualization modes.

2.2.1 Using dilute materials

The other possibility for visualization of model components is by changing the original MCNP model making all cells void except the cells, belonging to the components, which are to be visualized. Those cells are then filled with a material of such a low density that the mean free path is much smaller than the cell size. It is important that the materials are so dilute, that most of the neutrons do not experience any interactions with the material. The geometry is then flooded with neutrons, and one of the nuclear reactions (scattering, absorption, fission, etc.) is tallied using a mesh tally. The reaction rate density distribution corresponds to the shape of the component of interest and can be plotted by using one of the visualization software. One option to visualize the component is by using the iso-surface option which draws the outer boundary of the component. The other option is to use direct volume rendering, which is a common method for visualizing 3D scalar fields. Each point in a data volume is assumed to emit and absorb light. The amount and color of emitted light and the amount of absorption is determined from the scalar data by using the user-defined color map. Detailed information on individual visualization techniques can be found in Amira [2] and Voxler [3] help and user's manuals.

3. VISUALIZATION

3.1. TRIGA Mark II

The reactor core model was very similar to the one used for benchmark evaluation of TRIGA Mark II reactor [4], the main difference being in irradiation channels and the number of fuel elements. The fuel element was modeled exactly, meaning that Zr rod, stainless steel cladding, air gaps and Mo supporting disc were modeled explicitly. The control rods, supporting grid, graphite reflector with rotary groove and central irradiation channel in the core were also explicitly modeled. The main components of the TRIGA computational model are visualized by the described source subroutine method and are presented in Fig. 1.

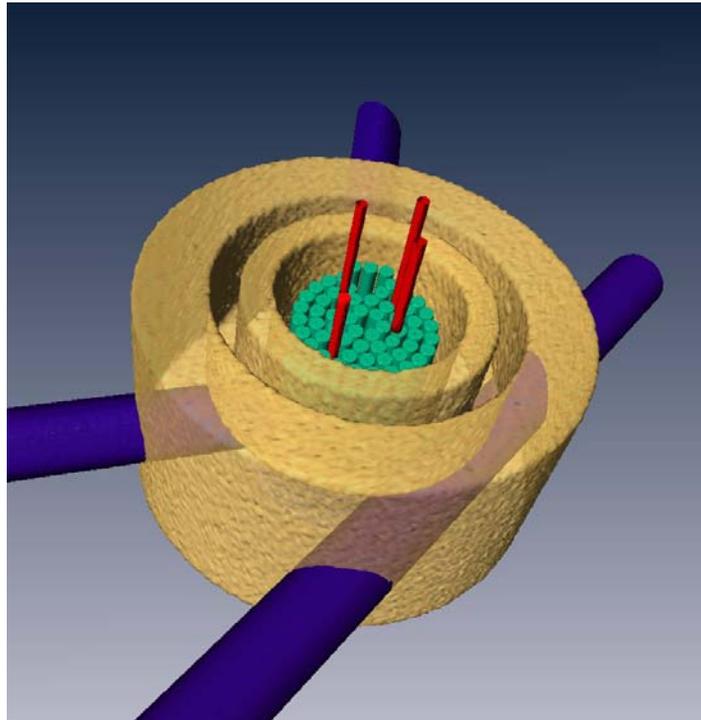


Figure 1: Components of the computational model of the JSI TRIGA reactor: radial and tangential irradiation channels (violet), graphite reflector (yellow and semi-transparent), fuel rods (green) and control rods (red).

Thermal ($E < 0.625$ eV) flux radial distribution in TRIGA core is presented in Fig.2. Large depressions of thermal flux inside the fuel elements are clearly seen. Local thermal flux peakings inside the water gaps between the fuel elements are also visible. Increase of thermal flux in the outermost ring (F) is observed, due to empty positions filled with water. The regions with flat flux distribution correspond to the irradiation channels, i.e. aluminum tubes filled with air. The purple region with zero flux value corresponds to the control rod inserted into the reactor core. The axial thermal flux distribution is presented in Fig. 3. The completely depressed thermal flux in the control rods and slightly depressed thermal flux in the fuel rods are clearly seen. The chopped cosine shape of axial neutron flux distribution can also be observed in the F ring and central irradiation channel.

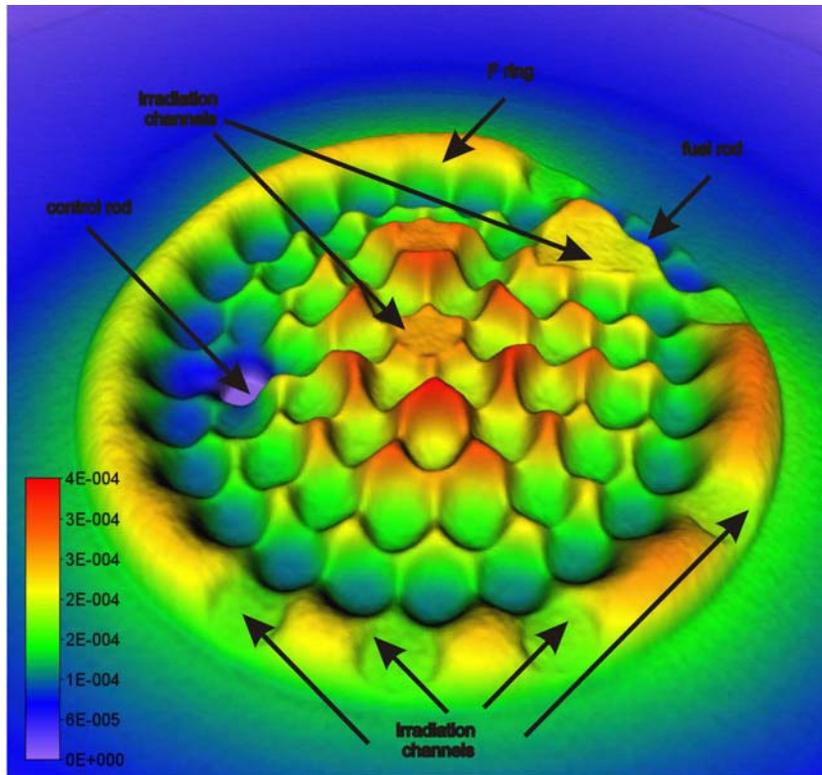


Figure 2: Radial thermal flux distribution (rel. units) at the core midplane of the JSI TRIGA Mark II core no. 189.

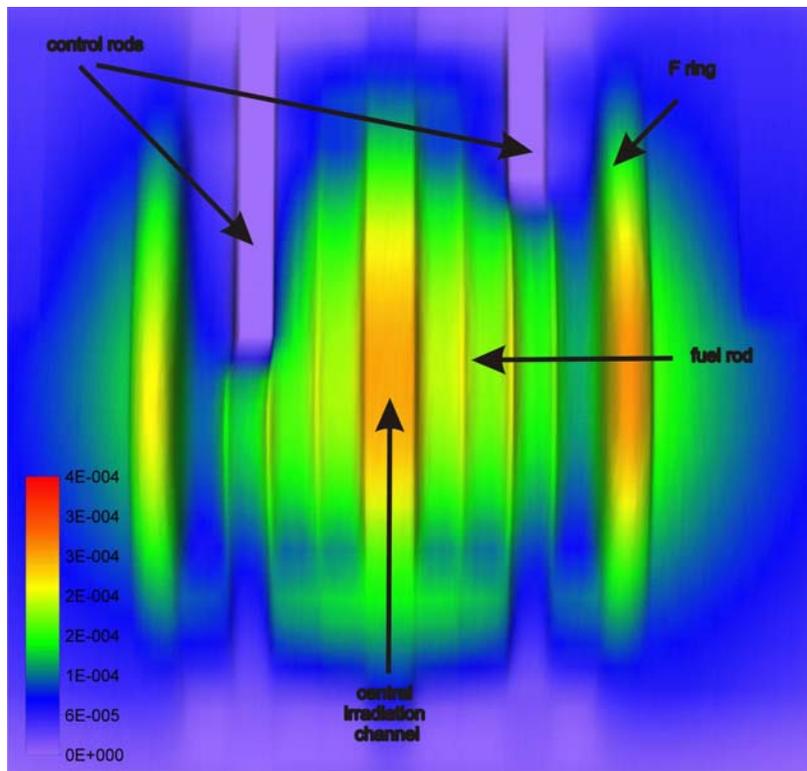


Figure 3: Axial thermal flux distribution (rel. units) in JSI TRIGA Mark II core 189.

The thermal flux distribution visualized in three dimensional isosurface (isoflux) mode is presented in Fig. 4. This figure clearly shows the effect of control rod insertion (red) on the thermal flux distribution as it can be seen that the thermal neutron flux is strongly depressed in the vicinity of the control rods.

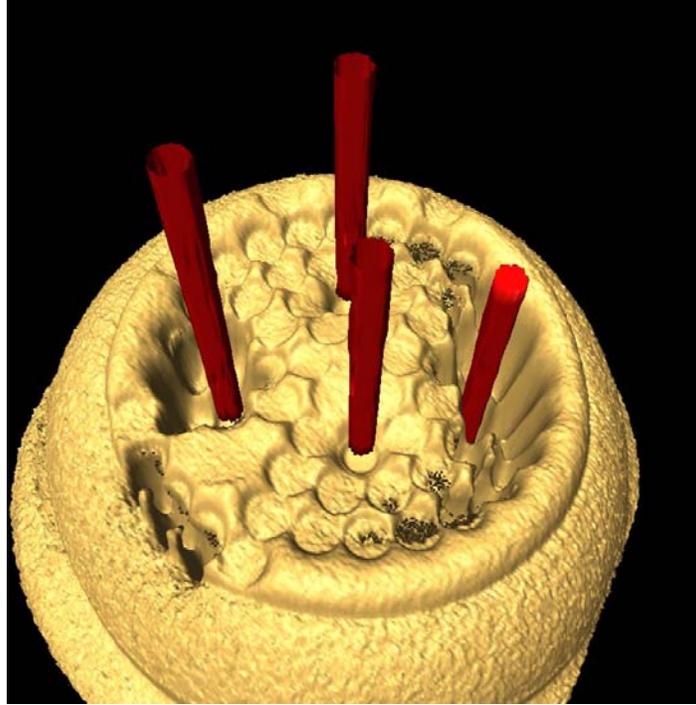


Figure 4: Thermal flux distribution in TRIGA reactor presented in isosurface mode (yellow) and control rods (red).

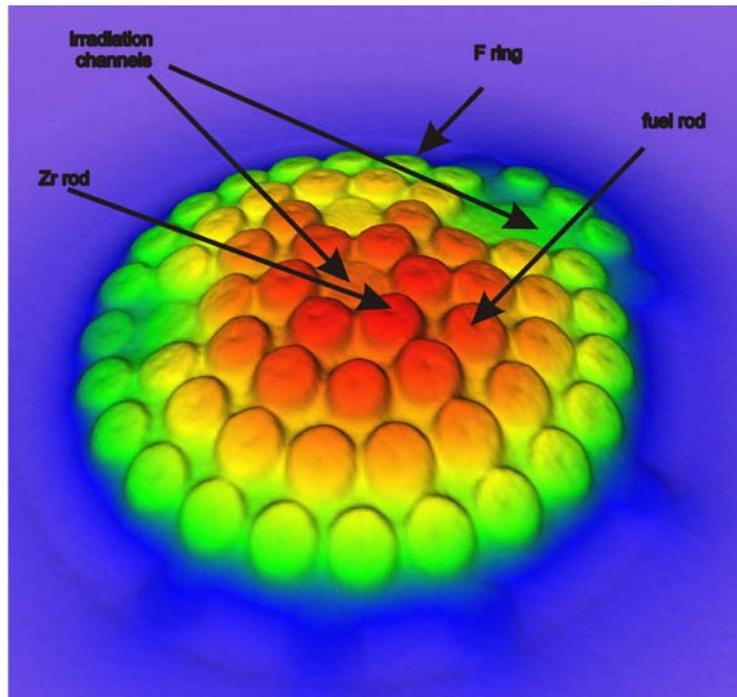


Figure 5: Radial fast flux distribution (rel. units) in JSI TRIGA Mark II core 189.

The radial fast ($E > 100$ keV) neutron flux distribution is presented in Fig. 5. It can be observed that the fast flux is the highest in the fuel elements. It is interesting that one can observe even slight depressions of fast flux in the Zr rod in the centre of a fuel rod, meaning the fission neutrons slow down on Zr.

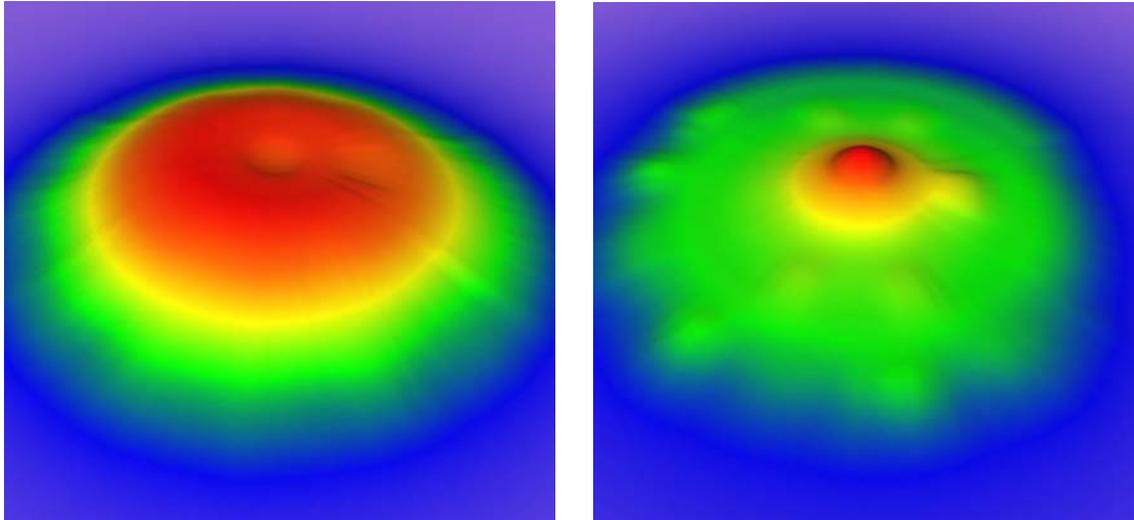


Figure 6: Radial fast (left) and thermal (right) flux distribution (rel. units) in JSI TRIGA Mark II core 189 calculated with TRIGLAV code [5].

It is important to note that visualization of the fluxes calculated in diffusion approximation lacks the educational aspect as all small scale flux distribution effects are smeared in diffusion approximation, while they are clearly visible in this model. Diffusion approximation fluxes for the same reactor conditions as in Figs. 2 and 4 are presented in Fig. 6 for comparison.

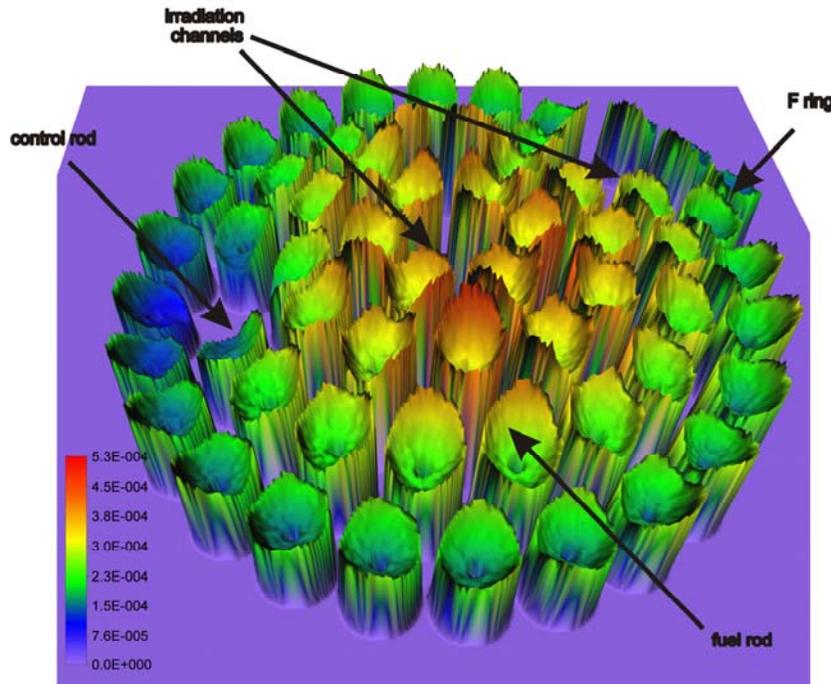


Figure 7: Radial power density distribution (rel. units) in JSI TRIGA Mark II core 189.

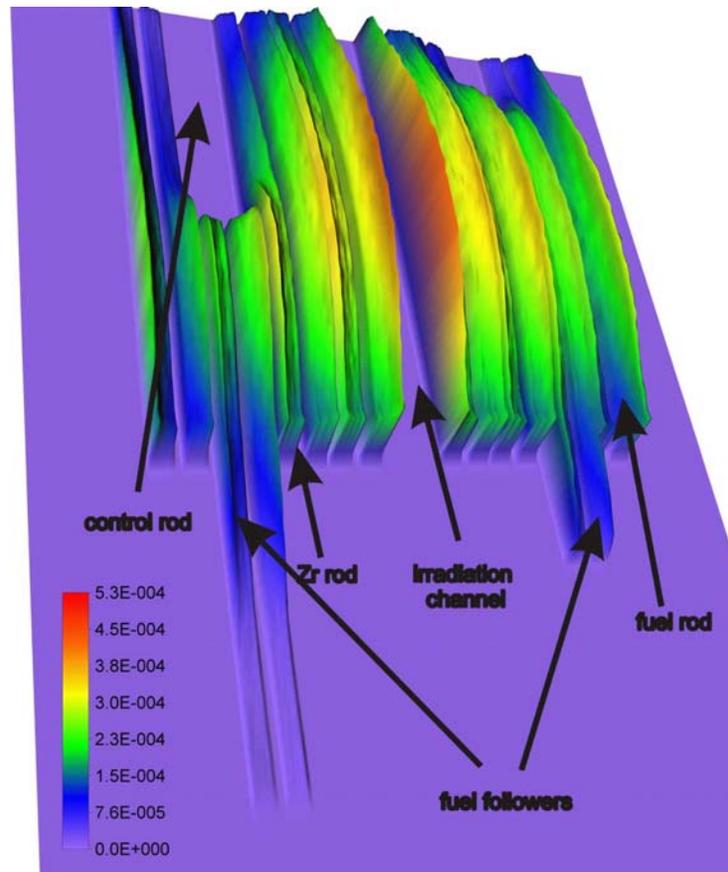


Figure 8: Axial power density distribution (rel. units) in JSI TRIGA Mark II core 189.

MCNP calculated power density radial distribution is presented in Figure 7. In-rod power density gradients due to the heterogeneities in the core such as water gaps and control rods can be clearly seen. Local fission density peakings at the edges of the fuel rods and the depressions near the control rod are also noticeable. In the fuel rod centre the fission density is zero as there is a Zr rod which contains no fissile material. Axial power density distribution is presented in Figure 8. We can observe that control rods are inserted deeply into the core leading to very low power density at the end of the fuelled followers.

3.2. Typical PWR core

The PWR core model is based on the typical PWR containing 121 fuel elements of 16×16 type. The fuel elements were modeled exactly, including all the gaps, fuel pellets and Zircaloy cladding. We modeled the core at the beginning of life (BOL). There are three types of fuel elements in the core; fuel elements containing 3.1 % enriched uranium are at the core periphery, fuel elements containing 2.6 % and 2.1 % enriched uranium are positioned in a chessboard-like pattern and are located in the centre of the core. Some elements contain 8, 12 or 16 rods with burnable absorber. The calculations were performed for the full core as well as for the single fuel element.

Radial thermal flux distribution, in a typical PWR core with burnable absorbers, is presented in Fig. 9. The areas with relatively high thermal flux inside the fuel element are the control rod and instrumentation channels filled with water at operating conditions. The fuel rods containing burnable absorbers are also clearly seen, as the thermal neutron flux is strongly depressed in those positions. Small peakings of the neutron flux appear also outside the core, in the reflector, where fast neutrons thermalize and therefore raise the thermal flux. The radial total flux distribution is presented in Fig. 9 (right). It can be seen that the distribution is relatively flat, meaning that core design was performed efficiently. Again we can observe strong flux depressions at the position of burnable absorber rods. At the core periphery the neutron flux falls down due to neutron attenuation.

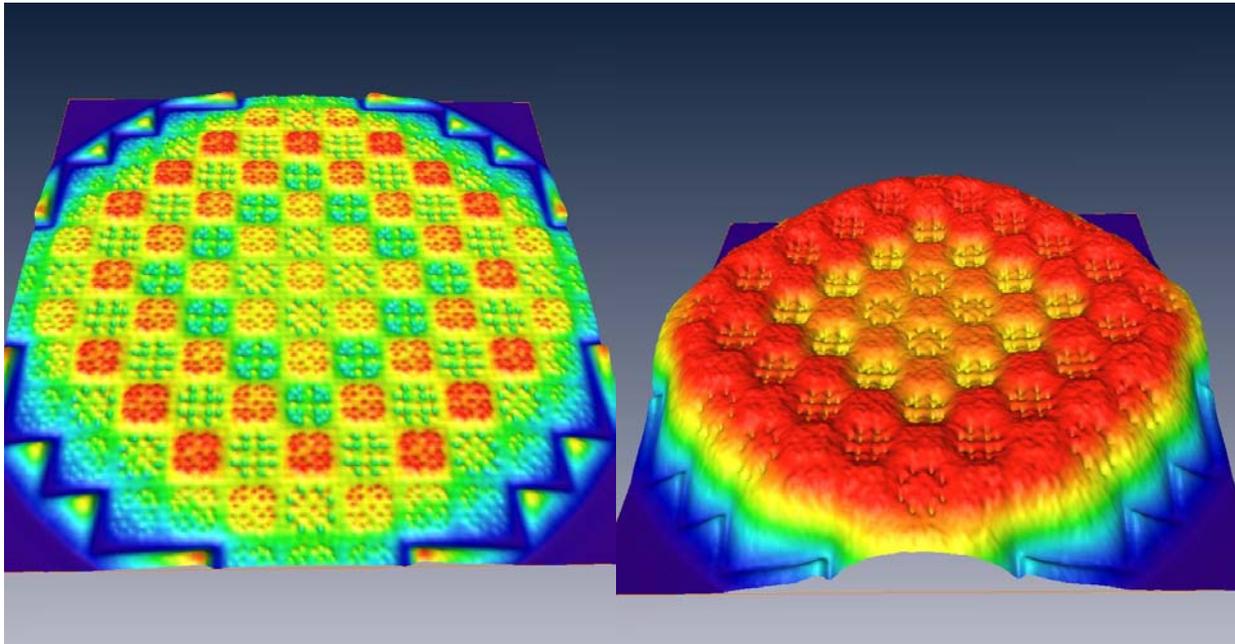


Figure 9: Radial thermal flux (left) and total flux (right) distribution in a typical PWR core with burnable absorbers.

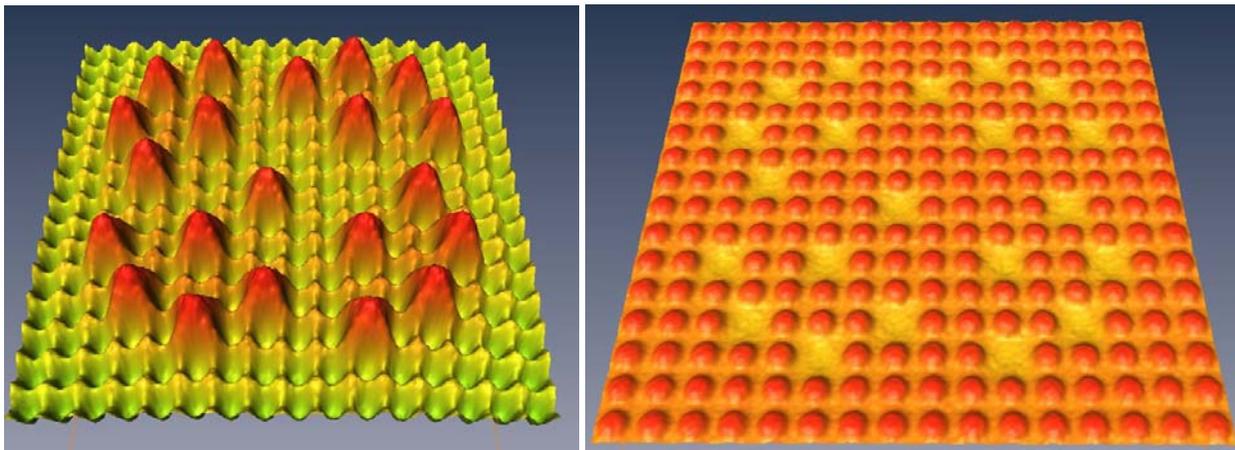


Figure 10: Radial thermal flux (left) and fast flux (right) distribution in a typical 16×16 PWR fuel element without burnable absorbers.

Radial thermal and fast flux distribution in a typical PWR fuel element are presented in Fig. 10. Areas with high thermal flux in control rod and instrumentation channels (filled with water) and thermal neutron flux depressions inside the fuel pellets are noticeable. In the Fig. 10 (right) one can clearly observe areas with high fast flux inside the fuel pellets, where fission neutrons are born, and areas with flat fast flux inside the water surrounding the pellets and in control rod and instrumentation channels.

3.3. Typical tokamak

The model of a fusion reactor is based on the tokamak design and includes practically all components inside the vacuum vessel, from divertor to first wall and blanket; in Fig. 11. (left) some of the inner components of a fusion reactor are visualized. Total neutron flux distribution in the poloidal plane is presented in Fig. 11. (right). It can be observed that neutron flux is the highest in the centre of the tokamak, where plasma is the hottest and most dense, leading to higher DT fusion rate and neutron production. The neutron flux then decreases towards the first wall and then attenuates in the blanket and shield.

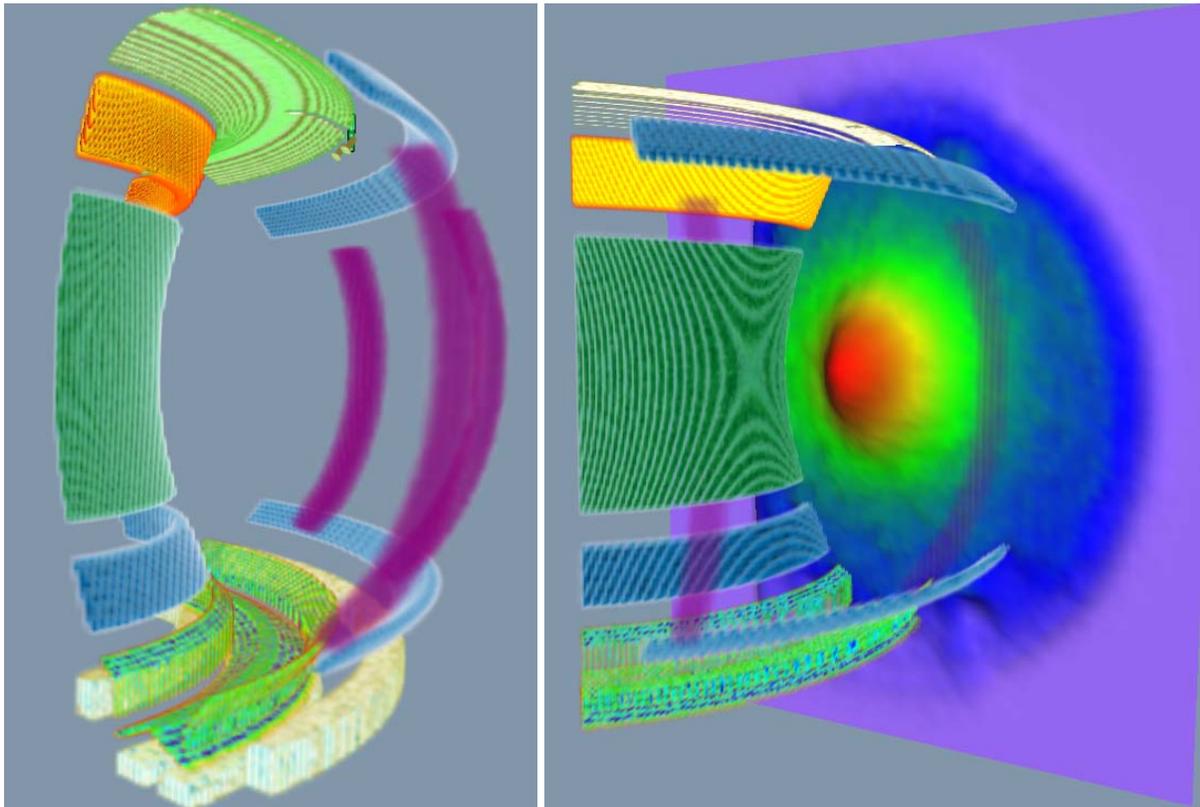


Figure 11: Inner components of a fusion reactor, produced from the MCNP model using the source subroutine technique (left). Total neutron flux distribution inside a toroidal fusion reactor (tokamak).

3.4. Neutron tracks

Neutron tracks (lines) and scattering events (points) from the monoenergetic 3 MeV plane source in graphite are presented in Fig. 12. The moderation process can be presented by coloring the particles by their energy. When the neutrons slow down to thermal energies ($\sim 1/40$ eV \rightarrow blue color) they scatter several times until they are absorbed. The upscattering events at thermal energies can also be observed where blue lines change the color to turquoise.

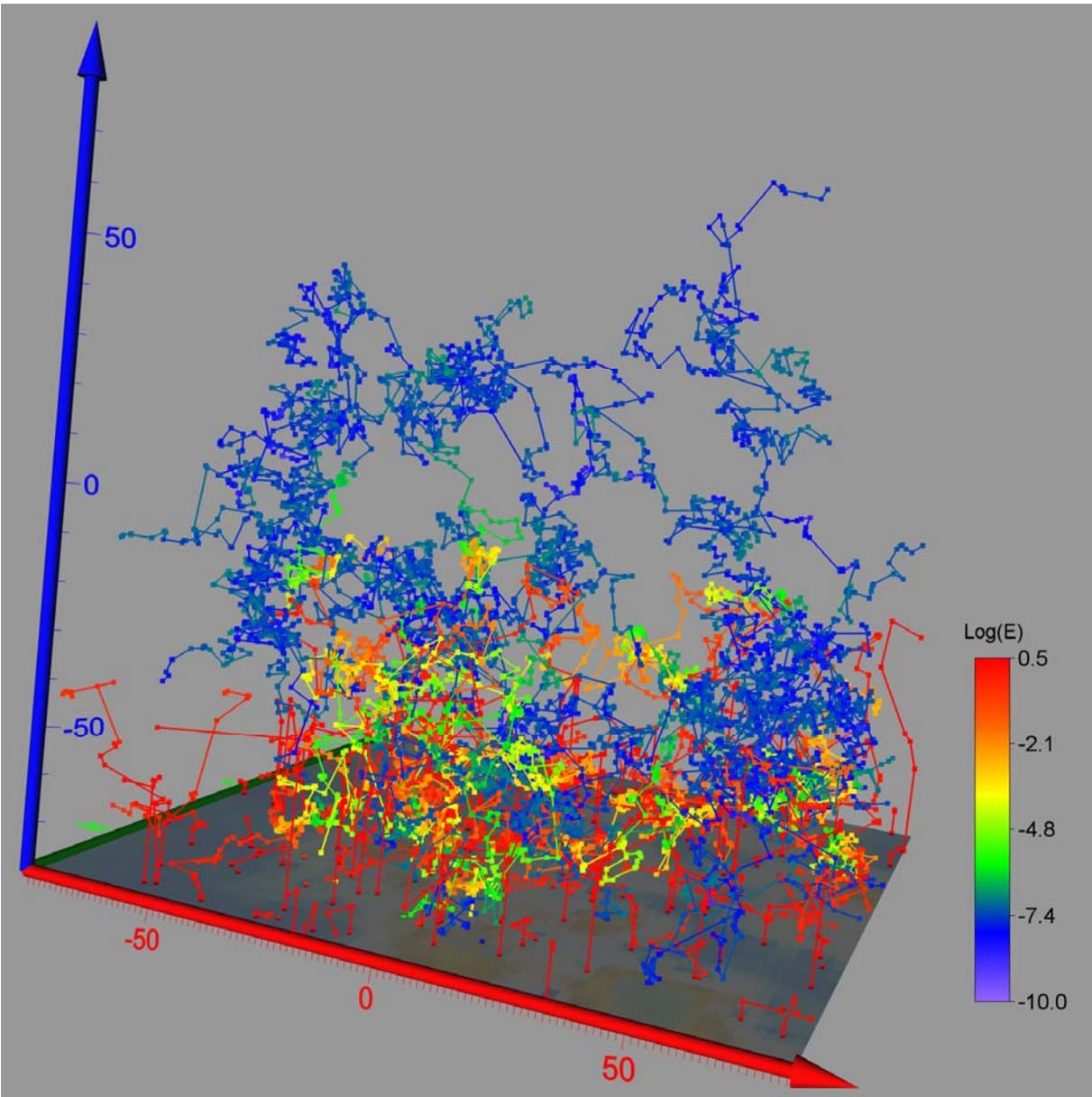


Figure 12: Particle tracks (lines) and scattering events (points) from the plane source emitting 3MeV neutrons perpendicular to the source plane into graphite. The tracks are coloured with logarithmic scale of energy (E/E_0), where $E_0=3$ MeV.

5. CONCLUSIONS

Using combination of advanced Monte Carlo codes such as MCNP and modern 3D data visualization tools like Voxler or Amira we can produce educative figures of neutron flux and power distributions in fission and fusion nuclear reactors. Moreover we can also visualize particle tracks, which is a very effective tool for explaining the basic principles of neutron transport.

The remembrance of most of the people is better, if they visualize a process. Therefore a representation of the reactor and neutron transport parameters is a convenient modern educational tool for (nuclear power plant) operators, nuclear engineers, students and specialists involved in reactor operation and design. Visualization is applicable not only in education and training, but also as a tool for core and irradiation planning [6].

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