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The applications of MCNP in nuclear data measurement and analysis

The nuclear transport calculations are crucial for the optimization of the reactor design parameters. The nuclear data from the different nuclear data libraries are used for such transport calculations. There are several nuclear transport codes are available such as MCNP, FLUKA, GEANT4, etc. The Monte Carlo based N Particle code i.e., MCNP code is a worldwide used for such particle transportation. It can transport neutron, photon, electron, proton, and many other particles. Its application in shielding design is remarkable and International Thermonuclear Experimental Reactor (ITER) shielding has been designed using MCNP code.
The code is useful for the various parameter estimation and optimization in the experiment. The detector efficiency is one of the sensitive parameter for the experiments based on activation analysis. In such experiment the source used for efficiency calibration is most probably a point source where the actual samples are volume sample. For such case MCNP modeling is useful to calculate the exact efficiency of the sample geometry. A sample detector geometry effect in cross section estimation will be discussed. The neutron spectra from the accelerator based neutron source is also a key parameter for the accurate calculation of the cross section of a nuclear reaction. In pelletron facilities the 7Li(p,n) reaction is most commonly used for the neutron production. The neutron energy spectra can be measured using the Time of Flight set up. The alternative is to produce it using particle transport which is done using MCNP. The generation of such neutron spectra will be discussed. These two work shows the importance of MCNP in nuclear experiments. Further, the different shielding materials are used for neutrons and gamma shielding. Concrete is a common reactor shielding material for neutron and high energy gamma. The amount of concrete used for the shielding requires high amount, which need more space and cost effective. In present work we have used different composition of concrete to increase its shielding capacity. We mainly changed the amount of WC and B4C to enhance the shielding of gamma and neutrons respectively. So the application of MCNP in different area of experiment and radiation protection simulation will be discussed.

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