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A preliminary assessment of MCNP unstructured mesh integration in the ITER neutronic model

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The design of nuclear fusion devices like ITER requires the execution of complex multi-physics simulations, involving different analysis disciplines such as mechanical, thermal-hydraulic and neutronics. Nowadays, thanks to the novel implementation of unstructured mesh capability into MCNP6, nuclear responses can be computed over meshes conformal with the components tackling the problematic load transference between different codes and, at the same time, improving the modelling methodology.
Initially a sensitivity analysis over type, order and number of elements was performed to evaluate the hybrid geometry performances in terms of memory demands, time of loading and the different MCNP implementation methodologies.
Subsequently, a limited section of the ITER vacuum vessel was modelled with unstructured meshes by means of the HyperMesh code. This region was chosen as one of the examples where the explicit structured mesh tally scoring can produce unphysical results near the cells boundaries where the mesh tally voxels cross two (or more) cells with different material properties and relative cross sections. Results are compared with the standard Constructive Solid Geometry (CSG) ITER model and discussed. Samplings of the unstructured mesh tallies were performed and locally assessed against the structured tallies. The particle fluxes and nuclear heating maps obtained show an overall agreement on the average resulting values, however significant local deviations were observed. In particular, unstructured mesh local results showed higher degree of correlation to the underlying materials and avoided the presence of unphysical peaks. However, the added complexity and the time required for meshing large regions, which include many complicated parts, makes an analysis of the benefits necessary, and it limits the application of hybrid geometries to particular cases.

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