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The analysis of shielding performance for toroidal field coils of CFETR

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Chinese Fusion Engineering Testing Reactor (CFETR) is an ITER-like superconducting TOKAMAK aiming to bridge the gap between ITER and future fusion power plant. Superconducting coils of CFETR provide high-intensity magnetic field to confine the core plasma. Ports are used for RH (Remote Handing) maintenance, plasma diagnose and other measuring equipment. Neutrons leaking from the ports will activate the material and deposit nuclear heat on coils, which may cause coils losing superconductivity. Among superconducting coils, toroidal field coils (TFCs) are closest to the core plasma and suffer the neutronic radiation damage more easily. In order to ensure stability of CFETR operation, shielding analyses of TFCs need to be estimated detailedly.

In this paper, the upper ports and equatorial ports are selected to analyze the influence for TFCs by using the Monte Carlo transport code MCNP with nuclear data library FENDL-2.1. The software McCad which is published by Karlsruhe Institute of Technology is used to convert the CAD model of CFETR into the MCNP input file. The nuclear heat and radiation dose of TFCs are calculated to evaluate the shielding performance. And the influence of the different schemes of ports for TFCs is analyzed, which can obtain an optimized port schemes for the shielding performance. The results can be used as some references for CFETR design.

Eligible for student paper award?

No

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