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FTP/P7-36: Neutronics Design of Helical Type DEMO Reactor FFHR-d1

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Neutronics design study has been performed in a newly started conceptual design activity for a helical type DEMO reactor FFHR-d1. Features of the FFHR-d1 design are enlargement of the basic configurations of reactor components and extrapolation of plasma parameters from those of the helical type plasma experimental machine Large Helical Device (LHD) to achieve the highest feasibility. From the neutronics point of view, a blanket space of FFHR-d1 is severely limited at the inboard of the torus. This is due to the core plasma position shifting to the inboard side under the confinement condition extrapolated from LHD.
The first step of the neutronics investigation using the MCNP code has been performed with a simple torus model simulating thin inboard blanket space. A Flibe+Be/Ferritic steel breeding blanket showed preferable performances for both tritium breeding and shielding, and has been adapted as a reference blanket system for FFHR-d1. The investigations indicate that a combination of a 15 cm thick breeding blanket, 55 cm thick WC+B4C shield, i.e. the blanket space of 70 cm, could suppress the fast neutron flux and nuclear heating in the helical coils to the design targets for the neutron wall loading of 1.5 MW/m2. Since the outboard side can provide a large space for a 60 cm thick breeding blanket, a fully-covered tritium breeding ratio (TBR) of 1.31 has been obtained in the simple torus model.
The neutronics design study has proceeded to the second step using a 3-D helical reactor model. The most important issue in the 3-D neutronics design is a compatibility with the helical divertor design. To achieve a higher TBR and shielding performance, the core plasma has to be covered by the breeding blanket layers as possible. However, the dimensions of the blanket layers are limited by magnetic field lines connecting an edge of the core plasma and divertor pumping ports. After repeating modification of the blanket configuration, the global TBR of 1.08 has been obtained. Sufficient radiation shielding for the helical coils has also been confirmed in the 3-D model.
The neutronics design studies have been conducted while keeping compatibilities also with in-vessel design, core plasma distribution etc. Optimization of the helical divertor configuration without significant degradation of shielding performance for coil systems is being investigated at present.

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