§18. Investigation of Neutronics Design of FFHR-d1

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A new design study of the helical type DEMO reactor FFHR-d1 has been started aiming for early demonstration of power generation.[1)] The features of the FFHR-d1 design are enlargement of component configurations and extrapolation of operation parameters from those of the present plasma experimental machine Large Helical Device (LHD) in NIFS to achieve the feasibility. Under the plasma confinement conditions in FFHR-d1, the core plasma position shifts to the inboard side of the torus. Therefore, the reduction of the inboard tritium breeding blanket and radiation shield thicknesses is required while keeping tritium fuel breeding and radiation shield performances. This neutronics issue in the FFHR-d1 design is being investigated by neutron and gamma-ray transport calculations.

The neutron and gamma-ray transport calculations in the present study has been performed by using the Monte Carlo transport code MCNP5 and nuclear data library JENDL-3.3. In the first stage of the neutronics study, the possible minimum thicknesses of the inboard breeding blanket and radiation shield have been investigated with a simple torus calculation model shown in Fig. 1. The volumetric ratio of the inboard part where reduction of the thicknesses is required has been estimated to be ~15 % and the averaged neutron wall loading is 1.5 MW/m^2 in FFHR-d1.

As to the breeding blanket for FFHR-d1, the tritium breeding and neutron shielding performances of Flibe, Flinak, Li, LiPb and LiSn cooled blanket systems have been compared with the same calculation model and the Flibe cooled blanket system has been selected as the first candidate from the balance of the performances. In the investigation of the shielding material, a superior neutron shielding performance of tungsten carbide (WC) has been confirmed as other reactor designs[2)] and selected for the main inboard shielding material also in FFHR-d1. After optimizing the balance between thicknesses of the tritium breeding blanket and radiation shield, the configuration of the inboard blanket and shield has been fixed to a 15 cm thick Flibe cooled blanket, 50 cm thick WC shield (WC: 85 vol.%, ferritic steel: 10 vol. %, helium coolant: 5 vol.%) and 5 cm thick B_4C shield (B_4C: 85 vol.%, 10 vol.%, helium coolant: 5 vol.%).[3)] The combination is estimated to suppress the fast neutron flux to ~3x10^{10} n/cm^2/s and nuclear heating to ~0.5 mW/cm^3 in the superconducting magnet region. The magnitude of the fast neutron flux indicates that FFHR-d1 would operate longer than 10 years from the view point of neutron irradiation damage on the superconducting metal wire.

The main shielding materials except the inboard region are ferric steel (70 vol.%) and B_4C (30 vol.%) as adopted in previous FFHR designs. The thickness of the shield is > 50cm. By attaching 60 cm thick Flibe cooled blanket at the outboard region, the fully-covered tritium breeding ratio (TBR) of 1.31 has been obtained in the simple torus model.

At present, the neutornics investigation has proceeded into the second stage using the three-dimensional calculation model shown in Fig. 2.[4)] After improvement of blanket and shielding dimensions, the global TBR of 1.08 has been obtained. This indicates that the coverage ratio of blanket layers is ~80 %. The performance of the WC radiation shield at the inboard region has also been confirmed.

The next major neutronics issue is suppression of neutron leakage through the divertor pumping ports to protect the poloidal coils (IV and OV coils in Fig. 2) from irradiation damages. The position and shape of the pumping ports is required to be optimized by keeping the compatibility between the divertor pumping performance and the neutron shielding performance. The neutornics performances of FFHR-d1with the Flinak, Li, LiPb and LiSn blankets are also being investigated at present.

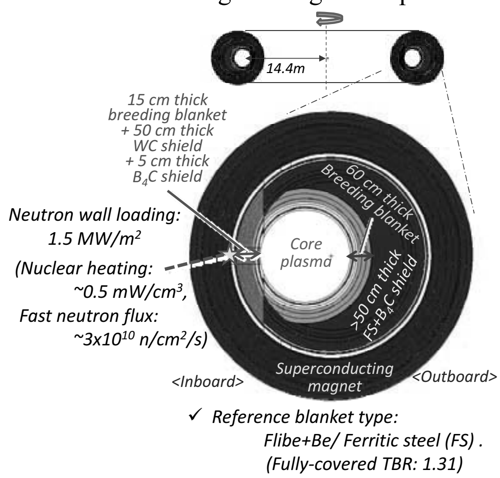

Fig. 1 Neutronics investigation with Simple torus model.

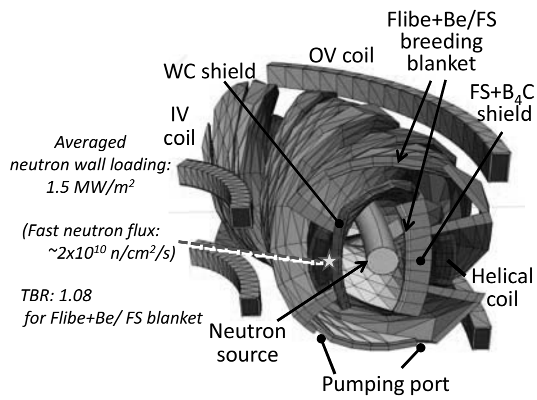

Fig. 2 Neutronics investigation with three-dimensional model.

1) Sagara, A. et al.: Fusion Eng. Des., in press.
2) El-Guebaly,L.A. et al.: Fusion Eng. Des. 51-52(2000)325.
3) Tanaka, T. et al.: Fusion Eng. Des., in press.
4) Tanaka, T. et al.: submitted to Plasma Fusion Research.