## §14. Neutronics Evaluation of FFHR-d1 with 3-D Calculation Model

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Three-dimensional (3-D) neutron and gamma-ray transport calculations are performed in the FFHR-d1 design study to evaluate tritium breeding and radiation shielding performances of a blanket system, distribution of neutron wall loading on first walls, distribution of nuclear heating and irradiation damages in reactor components, etc. A 3-D calculation model used in the neutronics evaluation is prepared by keeping consistency between core plasma, magnet, blanket and in-vessel component designs, and modified according to progress in the design study.

The Monte-Carlo transport code MCNP5[1] and nuclear data library JENDL-3.3[2] were used in the neutronics calculations. Figure 1 shows the present 3-D calculation model [3] whose blanket shapes have been modified from those of the last year [4] to suppress the neutron streaming to divertor spaces. A Flibe cooled blanket system has been adopted as the primary candidate for FFHR-d1. Inboard and outboard shield materials are tungsten carbide (WC) and ferritic steel + $B_4C$, respectively. Neutron wall loading, a tritium breeding ratio and radiation shielding performance were evaluated with the modified calculation model. A helical neutron source was simulated assuming that the ion temperature and density are proportional to $1\text{-}r^2$ and the fusion output is 3 GW. Here, r is normalized distance from the center axis of the core plasma.

Distribution of neutron wall loading (NWL) on the first walls is shown in Fig. 2. Since the core plasma of FFHR-d1 shifts to the inboard side, the maximum point appeared also on the inboard first wall. The maximum magnitude of NWL was 2.0 MW/m$^2$. Comparing with the average NWL of 1.5 MW/m$^2$ of the FFHR-d1 design parameter, the peaking factor is ~1.3.

The tritium breeding ratio (TBR) for the present calculation model was 1.08 for a Flibe blanket system. While the thin breeding blanket is installed to the inboard side, the Flibe+Be/ferritic steel blanket with $^6$Li enrichment to 90 % would achieve sufficient fuel breeding performance.

The fast neutron flux (> 0.1 MeV) distribution in the horizontal plane is shown in Fig. 3. In the coil winding region of the helical coils, the fast neutron flux was ~2 x $10^{10}$ n/cm$^2$/s at the inboard side. This magnitude of flux indicates that critical degradation of superconducting wires could be suppressed for more than 10 years. The combination of the 15 cm thick Flibe blanket and 55 cm thick tungsten carbide (WC) radiation shield attenuates the neutrons effectively in the 3-D geometry. The maximum nuclear heating was suppressed to ~0.6 mW/cm$^3$ in the coil winding region. This is almost the maximum acceptable level for cooling of the helical coils. Further design efforts for radiation shielding are required in the progress of detailed reactor design.

The fast flux distribution in Fig. 3 indicates that the divertors can be placed behind the radiation shield in FFHR-d1. The fast neutron flux was ~4 x $10^{12}$ n/cm$^2$/s at the divertor space. Compared with the flux of ~1-4 x $10^{14}$ n/cm$^2$/s at the blanket first walls, the magnitudes of fluxes at divertors could be suppressed by more than one order. Detailed analyses of irradiation damages on candidate divertor materials and impact on radioactivation are being performed at present.

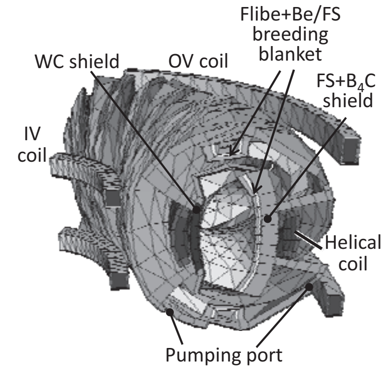


Fig. 1 Present 3-D neutronics calculation model of FFHR-d1 [3].

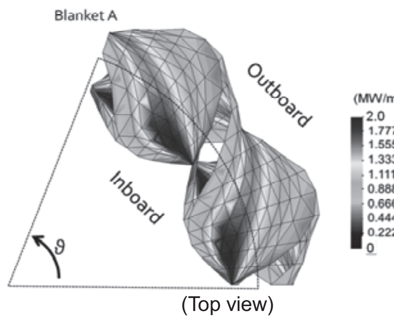


Fig. 2 Neutron wall loading distribution on blanket first walls [3].

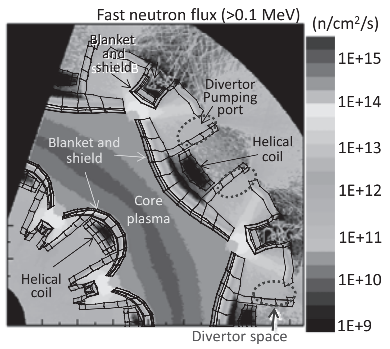


Fig. 3. Distribution of fast neutron (>0.1 MeV) flux [3].

[1] Briesmester,J.F.,Los Alamos National Laboratory Report LA-12625-M (2000).
[2] Shibata, K., et al., Journal of Nuclear Science and Technology, 39 (2002) 1125-1136.
[3] Tanaka, T., et al., FTP/P7-36, 24th IAEA Fusion Energy Conference, 08-13 October, 2012, San Diego, USA.
[4] Tanaka,T., et al., annual report of national institute for fusion science, April 2011- March 2012, 250.