§6. Evaluation of Neutron Environment at Divertors in FFHR-d1

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In the helical reactors, the magnetic field lines generated by helical coils are extracted from a core plasma to back sides of radiation shields without crossing breeding blankets nor radiation shields. This indicates that divertors can be placed behind radiation shields to suppress the radiation damages drastically. The integrated design study of FFHR-d1 is being conducted by keeping consistencies between design factors of a core plasma, magnet system, support structure, divertor, blanket etc. In the present study, neutron environment at divertors in FFHR-d1 has been evaluated by neutron transport calculations [1].

Figure 1 shows the 3-D neutronics calculation model of FFHR-d1. The positions of the divertors were tentatively decided according to calculated magnetic field lines. Divertor components were simulated by 2 m wide tungsten and ferritic steel layers. The thicknesses were 5 cm and 10 cm, respectively. The neutron transport calculation was performed with the MCNP5 code and JENDL-3.3 nuclear data library. Neutron spectra at center positions on the divertor surfaces were calculated and the irradiation damages were evaluated by multiplying dpa (displacement per atom) cross sections [2] based on JENDL-3 nuclear data library. The dpa cross sections assume the displacement energy of 40 eV. The fusion output of FFHR-d1 is 3 GW.

Figures 2 (a)-(c) show fast neutron flux distributions of >0.1 MeV. Since the spaces of divertor areas are significantly smaller at the inboard side in the present in-vessel configuration of FFHR-d1, the fast neutron fluxes at the inboard divertor areas are also significantly higher compared with those at the outboard divertor areas. The calculated damages at the center position of the blanket first wall (Position (A) in Figs. 2 (a)-(c)) and divertor surface (Position (B)) are shown in Table 1. The damages of first walls are evaluated with Fe which is a major element in a low activation ferritic steel. The damages at divertor surfaces which are evaluated for Fe, Cu and W are suppressed to ~1/10 and ~1/100 for the inboard and outboard divertors, respectively. Damages in Cr, Zr, Al and O, which are contained in Cu based materials such as Cu-0.3 wt% Al_2O_3, Cu-(0.25- 0.65)wt%Cr-(0.08-0.20)wt% Zr etc., are almost same level as those in the three materials.

Assuming 100 dpa as the criteria for replacement, the inboard blanket has to be replaced after 6 years operation. As to the outboard divertors, the damages in Cu materials could be suppressed <~1 dpa even after 10 years operation. This indicates that degradation in the elongation property is suppressed and Cu coolant pipes could be possible to be used. If Cu based material can be used up to ~10 dpa, the inboard divertors with Cu coolant pipes could be adopted without further reduction of the reactor availability.

In the divertor areas, the position with the lowest fast neutron flux is considered to be at the surface of the helical coil shield (C). If a thinner divertor unit is developed and attached on the shield surface, further suppression of damages by 30-40 % could be possible.

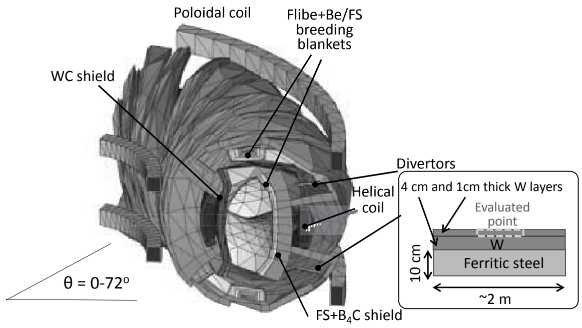

Fig.1 Present 3-D calculation model of FFHR-d1.

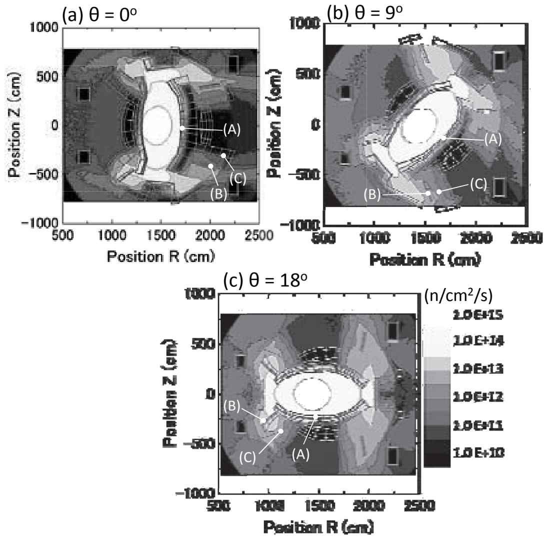

Fig. 2 Calculated fast neutron (>0.1 MeV) flux distributions in the poroidal cross sections.

Table 1 Calculated irradiation damages at the first wall and divertor area. Positions (A)-(C) are rotating in the poloidal cross section with the toroidal angle θ.

(Unit: dpa/year)

	Inboard	Outboard
First wall (A)	Fe: ~ 16	Fe: ~ 12
Divertor (B)	Fe: ~1.4 Cu: ~ 1.6 W: ~ 0.9	Fe: ~0.08 Cu: ~ 0.1 W: ~ 0.05
Shield surface in divertor area (C)	Fe: ~1.0 Cu: ~ 1.0 W: ~ 0.6	Fe: ~0.05 Cu: ~ 0.06 W: ~ 0.03

[1] T. Tanaka et al., Fusion Eng. Des. (2014), in press, http://dx.doi.org /10.1016 /j.fusengdes.2014.02.071.

[2] K. Maki et al., JAERI-Data/Code 97-002.