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Impact of fusion power and impurity radiation on DEMO divertor power handling
原型炉ダイバータの熱制御に対する核融合出力及び不純物放射の影響

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The power handling in the divertor is one of the most crucial issues for a fusion reactor design. In the previous study of development of the power handling scenario for a compact DEMO reactor, SlimCS, further reduction of the target heat load was required even in the case where more than 90% of the exhausted power from the core plasma was radiated by the argon impurity gas seeding. In this study, the impacts of the fusion power and the impurity radiation on the power handling in the DEMO divertor have been investigated by using an integrated divertor code SONIC. With decreasing the fusion power, the divertor plasma detachment is extended and the target heat load decreases. At the fusion power less than 2 GW, the target heat load become less than 6 MW/m^2, which is possibly handled by a tungsten mono-block divertor with a ferritic steel water-cooling pipe. It is also shown that the impurity radiation fraction on the exhausted power can be reduced to 80% at the fusion power of 2 GW when a copper-alloy is utilizable for divertor water-cooling tube.

1. Introduction

Huge power handling in the SOL/divertor region is one of the crucial issues for a tokamak fusion DEMO reactor. In a DEMO reactor, the desirable heat load on the divertor target is less than or comparable to the ITER design of 10 MW/m^2 because the available material is restricted by the strong neutron irradiation environment. The power removal capability is evaluated to be 5-7 MW/m^2 for a mono-block target with tungsten and reduced activation ferritic martensitic steel (RAFM) as the structure and the water-cooling pipe materials [1]. The primary technique for reduction of the divertor heat load to such a desirable level is enhancement of the radiation loss by impurity gas seeding.

Power handling scenario for a compact DEMO reactor (SlimCS: major radius R of 5.5 m, a fusion power P_{fus} of 3 GW) has been investigated [2]. The target heat load q_{target} was 16 MW/m^2 even in the case where more than 90 % of the exhausted power from the core plasma (P_{out}) was radiated by the argon (Ar) impurity gas seeding. The effects of the seeded impurity species (neon and krypton) and the divertor geometry on the divertor performance have been also investigated. Due to such effects, q_{target} could be decreased but it was still larger than the heat removal capability.

One of the possible solutions for further reduction of q_{target} is an advanced divertor concept, such as short super-X divertor [3]. Another possible solution is change of machine specifications, such as fusion power, machine size, etc.

In this study, the impacts of fusion power and the impurity radiation on the divertor power handling are investigated by using a suite of integrated divertor codes SONIC [4] with the impurity backflow model [5].

2. Impact of the fusion power on the divertor power handling

The input parameters for the SONIC simulation are as follows: the fixed fuel ion density of 7×10^{19} m^{-3} at the core side boundary of r/a = 0.95, and the spatially uniform diffusion coefficient for the particle and heat of D = 0.3 m^2/s and χ = 1 m^2/s, respectively. The power flux across the core side boundary (= P_{out}) is given corresponding to P_{fus}.

In the case of P_{fus} = 3 GW (P_{out} = 500 MW), large impurity radiation fraction on P_{out} (f_{rad}) of 92 % is achieved by the Ar impurity gas puff of 14.5 Pa m^3/s and the partially detached divertor plasma is obtained. Figure 1 shows the radial profile of q_{target} on the outer target. The plasma heat

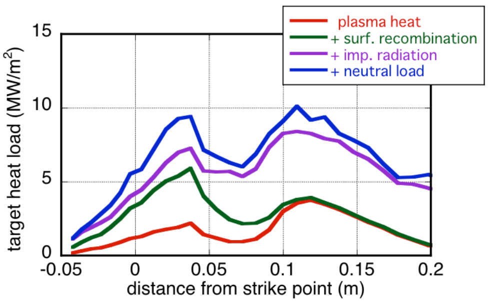

Fig. 1 The radial profile of the heat load on the outer target for the P_{fus} = 3 GW case. Heat loads by plasma heat flux, surface recombination, radiation power, and neutral flux are stacked.

load along the field line is decreased to less than 4 MW/m^2 by the partial detachment. The heat load due to the surface recombination of the ion flux is ~ 4 MW/m^2 near the strike point. The outer SOL region (r_{sep} > 7 cm) is still attached state. Therefore, the large impurity radiation power is in front of the target and the impurity radiation load is high. The neutral heat load of less than 3 MW/m^2 is widely distributed on the target. Consequently, the peak of the total heat load is about 10MW/m^2, which is larger than the heat removal capability of the target with the RAFM pipe.

In the P_{fus} = 2 GW case (P_{out} =320 MW), due to the low P_{out} with f_{rad} = 92 %, the detached region (T_i, T_e < 2 eV) is radially extended to 12 cm from 7 cm for the P_{fus} = 3 GW case. As shown in Fig. 2, sum of the plasma heat load and the surface recombination of the ion flux is less than 4 MW/m^2 due to the extension of the detachment region. In addition, the impurity radiation load decreases to less than 2 MW/m^2 because the impurity radiation moves upstream due to decrease in T_e. Consequently, peak of the total heat load becomes ~ 6 MW/m^2, which is possibly handled even in the case of the RAFM pipe.

3. Impact of the impurity seeding

In above case, f_{rad} = 92 % was assumed but it is difficult to extrapolate such large f_{rad} from the present experiments accompanied by the energy confinement degradation and fuel dilution appropriate for the fusion reactor plasma. Therefore, the impact of the reduced f_{rad} on the divertor performance is investigated at P_{fus} = 2 GW. With decreasing f_{rad} from 92 % to 80 %, the Ar impurity radiation power in the upstream, i.e., the SOL and edge region (r/a > 0.95) decreases from 115 MW for the f_{rad} = 92 % case to 81 MW. The electron heat flux through the divertor throat is increased and then the area of the low T_e less than 2 eV becomes narrow, i.e., the detachment becomes weak. The

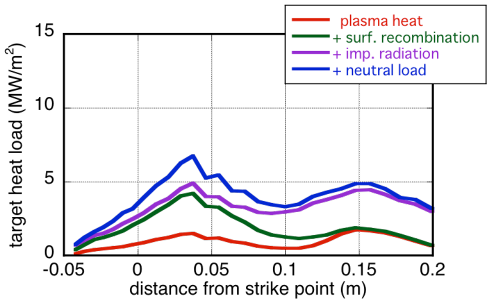

Fig. 2 The radial profile of the target heat load for the P_{fus} = 2 GW case.

heat load due to plasma heat transport and the surface recombination increases to 6 MW/m^2. In addition, the impurity radiation region moves to the target and radially expands due to increase in T_e near the strike point and the radiation load on the target is increased. Consequently, peak of the total target heat load increases to ~ 10 MW/m^2. When the copper-alloy cooling tube is utilizable for the divertor target, the heat removal capability increases to ~ 10 MW/m^2. In such a case, q_{target} in the f_{rad} = 80 % case may be handled.

4. Summary

In this study, some design points in the viewpoint of the divertor power handling were shown by the SONIC stimulation. However, f_{rad} is still high compared with the present experiments with high performance core plasma. In addition, the high SOL density and the impurity concentration of a few %, which are obtained in the simulation, may be inconsistent with the plasma physics design. To explore the design window consistent with the burning core plasma performance, impact of other key parameters, such as edge density profile, the impurity concentration, a major radius, etc., on power handling will be investigated.

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References

[1] K. Tobita, et al., Nucl. Fusion 49 (2009) 075029.
[2] N. Asakura, et al., Nucl. Fusion 53 (2013) 123013.
[3] N. Asakura, et al., Trans. Fus. Sci. Tech. 63 (2013) 70.
[4] H. Kawashima, et al., Plasma Fus. Res. 1 (2006) 031.
[5] K. Hoshino, et al., Contrib. Plasma Phys. 54 (2014) 404.