Divertor Design and Physics Issues of Huge Power Handling for SlimCS Demo Reactor

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ABSTRACT:
Power exhaust scenario for a 3 GW class fusion reactor with the ITER-size plasma has been developed with enhancing the radiation loss from seeding impurity. Impurity transport and the plasma detachment were simulated self-consistently, for the first time, under the Demo divertor condition using an integrated divertor code SONIC. Power handling with different seeding impurities showed that the total heat loading, including the plasma transport ($q_{\text{pl}}$) and radiation ($q_{\text{rad}}$), was reduced from 15 MW/m$^2$ to $\sim$10 MW/m$^2$ for the higher Z (Kr) case, where both heat load components become comparable, which $q_{\text{target}}$ extended over the wide area accompanying with impurity recycling. Geometry effects of the long-leg divertor was examined. Full detachment was obtained, and peak $q_{\text{target}}$ was decreased to 12 MW/m$^2$, where neutral heat load became comparable to $q_{\text{pl}}$ and $q_{\text{rad}}$, maybe due to small flux expansion. Finally, plasma transport significantly affected the performance of the plasma detachment. Peak $q_{\text{target}}$ was significantly reduced to 5 MW/m$^2$ well below engineering design level, with uniformly increasing the diffusion coefficient by the factor of two. Effects of the divertor geometry were also investigated.

1. Introduction
Handling of a huge exhausted power from the core plasma is the most important issue for the fusion reactor. Power handling scenario for a 3 GW class Demo fusion reactor with the ITER-size plasma has been developed in order to reduce the target heat load with enhancing the radiation loss from seeding impurity [1, 2, 3]. SlimCS is a conceptual Demo design of a low aspect ratio tokamak ($R/a = 2.6$) with the core dimension similar to those in ITER, and power generation capability of a giga-watt level [4], where the exhausted thermal power from the core plasma ($P_{\text{out}}$) is expected to about 500 MW (subtracting blemstrahlung and synchrotron radiations). On the other hand, the heat handling capacity of the divertor target is determined by the engineering design of 10 MW/m$^2$ or less under the Demo operation condition. Therefore, the radiative divertor scenario should be developed, where the total radiation loss fraction ($P_{\text{rad}\, tot}/P_{\text{out}}$) of larger than 90 % is required in the edge and divertor. In the recent studies, effects of the divertor geometry and impurity transport on the radiation distribution and plasma detachment have been quantitatively simulated in order to improve the divertor design and the plasma operation.

Divertor plasma simulation code, SONIC, has been developed for SlimCS in order to investigate formation of the plasma detachment ($T_e$ less than a few eV). Basic design of the divertor geometry with the V-shaped corner was proposed [1], where a simple radiation model for seeding argon (Ar) impurity (non-coronal equilibrium with a constant density ratio of $n_{\text{Ar}}/n_e$) was used. In the highly radiative plasma, the radiation power distribution in the divertor significantly affects the target heat load profile. Thus, impurity transport and the plasma detachment were simulated self-consistently, for the first time, under the Demo divertor condition [2], using an integrated SONIC code, where impurity Monte-Carlo (MC)
code IMPMC was incorporated and has advantages for impurity modelling since most kinetic effects on the impurity ions such as thermal and friction forces along the magnetic field in original formula.

Recent progress in the divertor design and understanding of the divertor performance are summarized. Development of the SONIC code and improvement of the modelling are explained in Sec. 2. Three important physics parameters to control the impurity and plasma transport are discussed. Seeding impurity gases from low Z (10: neon) to relatively high Z (36: krypton) are investigated in Sec. 3. Effects of the long leg divertor (geometry) and the plasma radial diffusion in SOL (plasma transport) are determined in Sec. 4 and Sec. 5, respectively. Conclusions of these studies are given in Sec. 6.

2. Divertor Simulation and Developments

A common calculation mesh for neutral and impurity transport is shown in Fig. 1. Core plasma boundary for the plasma calculation is set at $r/a = 0.95$, where $P_{\text{out}} = 500$ MW is exhausted and $n_e = 7 \times 10^{19}$ m$^{-3}$ is fixed. Total fuel gas puff of $\Gamma_{\text{puff}} = 1 \times 10^{23}$ atm/s ($\sim 200$ Pa-m$^3$/s) is injected at the outer divertor and midplane. Thermal diffusivities of electron and ion, $\chi_e = \chi_i = 1$ m$^2$/s, and particle and impurity diffusion coefficients, $D_e = 0.3$ m$^2$/s, are constant in the edge and SOL similar to those in the ITER simulation [5]. The fuel pumping speed ($\Delta_{\text{pump}} = 200$ m$^3$/s) is specified at the exhaust port of the divertor, i.e. right end of the calculation mesh. Seeding impurity (Ar for the most cases) is injected from the outer divertor until $P_{\text{rad tot}}$ is achieved and the puffing rate becomes small by the feedback.

Development of the SONIC simulation has been in progress in order to improve conversion of the plasma solution, in particular, under the DEMO divertor condition, where the parallel heat flux in SOL is about 5 times larger than ITER, and its characteristic width ($\lambda_3^{\text{SOL}}$) of 2-3 mm is smaller due to $T_e$ and $T_i$ higher. Following techniques and corrections of modelling are incorporated, (1) distribution of Ar atom under the divertor (exhaust route) is assumed by a result calculated up to a steady-state condition. Here, Ar atom is injected from the exhaust slot to the divertor region like a backflow ($\Gamma_{\text{Ar back}}$), where particle balance of Ar atom is considered. (2) Neutral source terms near the target are determined by average values from several time steps in order to reduce the MC noise and perturbation. (3) Thermal force term for long mean-free-path (i.e. high plasma temperature at upstream of the target) is corrected to be smaller.

Distribution of the radiation power density, and profiles of the divertor plasma and heat load for the case of $P_{\text{rad tot}} = 460$ MW ($P_{\text{rad div}}/P_{\text{out}} = 0.92$) are demonstrated in Figs 2 and 3. The total radiation loss at the divertor ($P_{\text{rad div}}$) and the main plasma edge plus SOL ($P_{\text{rad edge&SOL}}$) are 307 MW ($P_{\text{rad div}}/P_{\text{out}} = 0.61$) and 153 MW ($P_{\text{rad edge&SOL}}/P_{\text{out}} = 0.31$), respectively. Ar puff at the divertor is $1.45 \times 10^{21}$ Ar/s. Ar backflow $\Gamma_{\text{Ar back}}$ is handled as gas puff from the exhaust slots: $\Gamma_{\text{Ar back(in)}} = 8.8 \times 10^{21}$ Ar/s and $\Gamma_{\text{Ar back(out)}} = 13.8 \times 10^{21}$ Ar/s, which are determined to balance in Ar puff and its net pumping fluxes.

Fig.1 Divertor geometry and calculation mesh for SlimCS. Locations of impurity (Ar) and fuel gas injections and exhaust route are shown.
Generally, plasma heat loading at the outer divertor is severer than that at the inner divertor, in particular, lower aspect tokamaks. Figures 2(a) and 3(a) show that, in the inner divertor, significant radiation area is shifted to the upstream and full detachment is observed widely over the target. On the other hand, in the outer divertor, the partial detachment is extended from the outer strike-point to the upper region in Fig. 2(b), and low $T_e$ and $T_i$ (less than 2 eV) are observed at $\Delta_{\text{div}} < 6$ cm in Fig. 3(c). Noted that at the outer flux surfaces ($\Delta_{\text{div}} > 6$ cm), $T_e$ and $T_i$ are increased due to low $n_e$ and collisionality.

The total heat load, $q_{\text{target}}$, is evaluated including radiation power load ($q_{\text{target}}^{\text{rad}}$) and neutral load ($q_{\text{target}}^{\text{ne}}$), in addition to the plasma heat load along the field line ($q_{\text{target}}^{\text{plasm}} = \gamma \cdot n_d \cdot C_{sd} \cdot T_d + n_d \cdot C_{sd} \cdot E_{\text{ion}}$), where $\gamma$, $C_{sd}$, $n_d$, $T_d$, $E_{\text{ion}}$ are sheath transmission, plasma sound velocity, density and temperature at the divertor sheath, recombination energy, respectively. At the outer target, peak $q_{\text{target}}$ of 16 MW/m$^2$ is seen at the boundary of the detachment, where $q_{\text{target}}^{\text{plasm}} \sim 9$ MW/m$^2$. At the same time, $q_{\text{target}}^{\text{ne}} \sim 6$ MW/m$^2$ is large since the significant radiation volume is near above the strike point. Large $q_{\text{target}}^{\text{rad}}$ of 4-6 MW/m$^2$ is extended over the wide target area. As a result, the peak $q_{\text{target}}$ is larger than the engineering design level of 10 MW/m$^2$. On the other hand, at the inner target, peak $q_{\text{target}}$ is 7 MW/m$^2$ attributed to surface recombination by low energy ions, while $q_{\text{target}}^{\text{rad}}$ is large component (4 MW/m$^2$) over the wide target area. Consequently, the power exhaust result for $P_{\text{rad}}^{\text{tot}}/P_{\text{out}} = 0.92$ with the Ar seeding, the peak $q_{\text{target}}$ at the outer target was larger than the
engineering design level. Improvement of the power handling is necessary to reduce the peak heat load.

3. Radiation and impurity control in the divertor

Impurity seeding such as such as nitrogen (N$_2$), neon (Ne) and argon (Ar) gas puff has been carried out in the improved confinement plasmas of the existing tokamaks. Large radiation loss over the wide plasma region such as in the main SOL and edge as well as the divertor is required in the Demo reactor. Higher Z impurity will be preferable since radiation loss rate coefficient for higher Z impurity is increased at high $T_e > 50$ eV [6]. Effects of seeding impurity species (Ne, Ar and Kr) on detachment and radiation distribution are compared.

Distribution of the radiation power density, and profiles of the divertor plasma and heat load are shown in Figs 4 and 5 for $P_{\text{rad}}^{\text{tot}} = 460$ MW (the same as Ar). For the low Z (Ne), strong radiation is seen along the separatrix, and the plasma detachment is produced in the relatively narrow region near the strike-point ($\Delta_{\text{div}} < 5$ cm). With increasing Z (Ar and Kr), width of the intense radiation is gradually increased. The peak $q_{\text{target}}$ of 15-16 MW/m$^2$ is comparable for Ne and Ar, where $q_{\text{target}}$ corresponds to ~10 MW/m$^2$.

On the other hand, for the case of Kr seeding, the plasma detachment widely extends above and along the target, and the peak $q_{\text{target}}$ is decreased to 11 MW/m$^2$, due to the reduction in $q_{\text{target}}^{\text{plasma}}$ (~5 MW/m$^2$) as well as $T_e$ and $T_i$. At the same time, $q_{\text{target}}^{\text{rad}}$ is increased over the wide area. The peak $q_{\text{target}}$ is slightly larger than the engineering design level.

**TABLE I: Radiation power distribution**

<table>
<thead>
<tr>
<th>Ne</th>
<th>Ar</th>
<th>Kr</th>
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</thead>
<tbody>
<tr>
<td>204(41)</td>
<td>170(34)</td>
<td>86(17)</td>
</tr>
<tr>
<td>180(36)</td>
<td>127(25)</td>
<td>153(31)</td>
</tr>
<tr>
<td>154(31)</td>
<td>76(15)</td>
<td>230(46)</td>
</tr>
</tbody>
</table>

The total radiation loss at the divertor and the main edge are summarized in TABLE I: for Ne, $P_{\text{rad}}^{\text{div}} = 374$ MW (75%) is significantly large, while $P_{\text{rad}}^{\text{div}} = 230$ MA (46%) is comparable to $P_{\text{rad}}^{\text{edge\&sol}}$ for Kr. Reduction in $P_{\text{rad}}^{\text{div-in}}$ with higher Z is
caused by shifting the large radiation region upstream of the full detached divertor. In order to achieve $P_{\text{rad}}^{\text{tot}} = 460 \text{ MW}$, impurity gas puff rates from the outer divertor are $3.8 \times 10^{21}$ (Ne), $1.5 \times 10^{21}$ (Ar) and $0.92 \times 10^{21}$ atm/s (Kr), and back flows from the divertor slots are also decreased.

Consequently, distribution of the radiation power can be controlled by selecting the impurity gas. With increasing $Z$, $P_{\text{rad}}^{\text{edge & sol}}$ becomes comparable to $P_{\text{rad}}^{\text{div}}$, thus the wider plasma detachment and lower peak $q_{\text{target}}$ can be produced in the outer divertor. Even for the case of Kr, improvement of the power handling is necessary. At the same time, restrictions of $P_{\text{rad}}^{\text{edge & sol}}$ and $P_{\text{rad}}^{\text{div}}$ would be determined by the edge plasma performance such as confinement degradation and by the thermal stability at the inner and outer detachment conditions.

4. Improvement of the divertor geometry: Long-Leg Divertor

From the previous result, improvement of the divertor geometry is important in order to reduce the peak heat load. V-shaped corner is efficient to enhance particle and impurity recycling near the strike-point, and to produce the partial detachment [1]. On the other hand, further reduction of the peak $q_{\text{target}}$ less than 10 MW/m² is required for the Demo divertor. Recently, effect of “longer leg divertor” was investigated for SlimCS [3], using a simple impurity model with non-coronal equilibrium for Ar with a constant ratio of $n_{\text{Ar}}/n_i$, where the divertor plasma temperature $T_{\text{div}}$ is expected to decrease because of longer connection length to the target ($L_{\parallel}^{\text{div}} = L_p^{\text{div}} B_{\parallel}/B_p$), i.e. $T_{\text{div}} \propto q_{\parallel}^{10/7}/n_i^{4/7} L_{\parallel}^{4/7}$ ($q_{\parallel}$ and $n_i$ are parallel heat flux and upstream plasma density, respectively) from a simple 2-point model. TABLE II compares key geometry parameters of the SlimCS and ITER divertors. From the viewpoint of the divertor design, the divertor size, i.e. poloidal length from the divertor null to the target ($L_p^{\text{div}}$), is increased from 1.72 to 2.42 m, and the magnetic flux expansion at the target ($f_{\text{exp}}^{\text{div}}$) is reduced to 60%. Study of the divertor geometry effect on the detachment profile and radiation distribution has been progressing, using the integrated SONIC code.

The radiation distribution and heat load profile are shown in Figs 6 and 7. It was found that the large radiation region is shifted upstream...
from just above the target due to a reduction in $T_e$ along the field line, and that $T_i$, $T_e$ and $n_e$ at the divertor decrease. As a result, peak $q_{\text{target}}$ decreases to 12 MW/m², where a reduction in heat loading by the plasma transport and radiation is caused by full detachment over the outer divertor. On the other hand, power loading due to neutral flux is increased, which is produced by compression of neutrals and low energy ion flux near the target.

For the long-leg divertor, $P_{\text{rad}}^{\text{div}}$ is increased to 400 MW (80%), where, $P_{\text{rad}}^{\text{div-out}}$ is significantly enhanced to 323 MW (65%) since particle and impurity recycling is enhanced in the long-leg divertor. It is noted that radiation heat load on the PFC along the long leg is relatively small (less than 3 MW/m²). Here, $P_{\text{rad}}^{\text{div-in}}$ and $P_{\text{rad}}^{\text{edge&sol}}$ decrease to 77 MW (15%) and 60 MW (12%), respectively.

Consequently, effect of the long leg divertor is efficient for a reduction in the peak $q_{\text{target}}$. On the other hand, relatively large contribution of the neutral flux is caused by effect of small magnetic expansion even when the full detachment is produced. It was also found that radiation distribution at main edge and divertor can be controlled by the divertor geometry as well as the seeding impurity species.

5. Effects of Plasma Transport on Detachment

Plasma transport affects the performance of the plasma detachment and its control. Although profiles of the heat and particle fluxes are important parameters [7, 8], application of radial diffusions of energy and particles to the reactor is not well known. Uniform heat and particle diffusion coefficients of $\chi_e$, $\chi_i$ and $D_e$ are generally used in the simulation. Effects of the plasma transport, in particular, enhancement of the plasma diffusion, on the divertor plasma are investigated. For the reference case of $\chi_e = \chi_i = 1$ m²/s, $D_e = 0.3$ m²/s as is used for ITER, the plasma detachment in the outer divertor was obtained near the strike-point (< 6 cm) for the Ar seeding case, and the peak $q_{\text{target}}$ was 16 MW/m² in Sec. 2. For the Demo edge plasma, $\lambda_q^{\text{SOL}}$ of 2.2 mm is smaller than ITER (3.6 mm) [9] due to $T_e$ and $T_i$ higher. Two representative examples (1) $\chi$ and $D_e$ are increased to 2 and 0.6 m²/s (2 times), and (2) $\chi$ and $D_e$ are enhanced 5 times only at the outer SOL of $r_{\text{mid}} > 1.5$ cm, are investigated. Profiles of $n_e$, $T_e$ and $T_i$ at the outer midplane are shown in Fig. 8. For Case (1), $\lambda_q^{\text{SOL}}$ increases slightly from 2.2 to 2.7 mm. Case (2) is a simulation of enhanced radial diffusion, i.e. “non-diffusive transport”, which was generally observed in most tokamak experiments [10, 11], although radial profiles of effective diffusion coefficients for the plasma energy was not well determined.

![Fig. 7. Profiles of (a) $n_e$, $T_e$, $T_i$ and (b) heat load, at the long leg divertor target.](image)

![Fig. 8. Profiles of $n_e$, $T_e$, $T_i$ and $\chi$ and $D_e$. (a) uniform in SOL, (b) enhanced 5 times at the outer flux surfaces of $r_{\text{mid}} > 1.5$ cm.](image)
Figure 9 show distribution of the radiation power, the plasma parameters and heat load at the outer target for the case (1). Plasma detachment extends in wider area (<15 cm) and almost full detachment was obtained. The peak $q_{\text{target}}$ is significantly reduced to 5.5 MW/m² in Fig. 9 (a, b); both $q_{\text{target}}\text{plasma}$ and $q_{\text{target}}\text{rad}$ are decreased to 3.3 and 1 MW/m², respectively. This is caused by the radiation volume shifting upstream and the plasma detachment extending. Flat $q_{\text{target}}$ profile of 3 – 5 MW/m² is seen over a wide area at the target, which can be handled by the Demo engineering design.

Figure 10 show distribution of the radiation power, the plasma parameters and heat load at the outer target for the case (2), where the outer SOL of $r_{\text{mid}}^{\text{mid}} >1.5$ cm corresponds to the divertor target of $r_{\text{div}}^{\text{div}} >15$ cm. Partial plasma detachment is extended to 10 cm. Reductions in the peak $q_{\text{target}}$ to 12 MW/m² and $q_{\text{plasma}}$ to 7 MW/m² are seen at $r_{\text{div}}^{\text{div}} -3$ cm. Although detachment is narrower and the reduction in the peak $q_{\text{target}}$ is smaller, this fact suggests that enhancement of the local $\chi$ and $D_\perp$ for the outer flux surfaces affects a reduction in the peak $q_{\text{target}}$ near the separatrix. From the two examples, it is found that the radial diffusion is an important parameter to determine the divertor detachment and heat load profiles in the divertor.

6. Summary and Conclusion

Power exhaust scenario for a 3 GW class fusion reactor with the ITER-size plasma was investigated with enhancing the radiation loss from seeding impurity. Impurity transport and the plasma detachment were simulated self-consistently, for the first time, under the Demo divertor condition using an integrated divertor code SONIC. Conversion time and stability of
the solutions were improved by incorporating the following numerical techniques and corrections of modelling: (1) Ar atom transport under the exhaust route is assumed by a typical calculated result. (2) Neutral source terms near the target are determined by average values in order to reduce the MC perturbation. (3) Thermal force term for long mean-free-path (i.e. high plasma temperature at upstream of the target) is corrected to be smaller.

Key physics issues to control the impurity and radiation distributions are investigated. Generally, plasma heat loading at the outer divertor is severer than that at the inner divertor, and performance of the outer divertor plasma is rather emphasized, where reduction in peak $q_{\text{target}}$ is required to the engineering design of 10 MW/m$^2$ or less under the Demo operation condition.

First, power handling with different seeding impurities from low (10: neon) to relatively high Z (36: krypton) was studied. The peak $q_{\text{target}}$ is decreased from 15-16 MW/m$^2$ for Ne and Ar to 11 MW/m$^2$ for Kr, where the local $q_{\text{target}}^{\text{ave}}$ was decreased from 10 MW/m$^2$ to 5 MW/m$^2$ due to extending the detachment and reduction in $T_e$ and $T_i$. At the same time, $q_{\text{target}}^{\text{ave}}$ is comparable to $q_{\text{target}}^{\text{ave}}$ in the radiative divertor, and extended over the wide area accompanying with impurity recycling. Full detachment was not observed at the outer divertor.

Second, long-leg divertor was examined to investigate effect of longer connection length and divertor chamber. Particle and impurity recycling was enhanced in the long-leg divertor, and the full detachment was obtained. The peak $q_{\text{target}}$ was decreased to 12 MW/m$^2$ due to a reduction of $q_{\text{target}}^{\text{ave}}$ and $q_{\text{target}}^{\text{out}}$. On the other hand, contribution of the neutral flux was increased due to small magnetic expansion even in the full detachment. From the first and second studies, it was found that radiation distribution at edge and divertor is controlled by the divertor geometry as well as selection of the seeding impurity species.

Third, plasma transport affects the performance of the plasma detachment and its control, effects of heat and particle diffusion coefficient were studies in two examples: enhancement of $\chi$ and $D$, (1) uniformly by the factor of 2, and (2) only at the outer flux surfaces ($\rho_{\text{mid}} > 1.5$ cm). It was found that the radial diffusion is an important key to determine the divertor detachment and radiation distribution in the divertor. For the former case, peak $q_{\text{target}}$ was significantly reduced to 5 MW/m$^2$ well below engineering design level. For the latter case, peak $q_{\text{target}}$ was reduced to 12 MW/m$^2$, which suggested that enhancement of the local $\chi$ and $D$ for the outer flux surfaces affects a reduction in the peak $q_{\text{target}}$ near the separatrix.

References