High Temperature Divertor Plasma Operation

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Abstract

High temperature divertor plasma operation has been proposed, which is expected to enhance the core energy confinement and eliminates the heat removal problem of a reactor grade device. In this approach, the heat flux is guided through divertor channels to remote areas with large target surface, to reduce the heat load on the target plate substantially. This allows pumping of the particles recycled at the target plate and hence maintaining of high divertor plasma temperature, comparable to the core temperature. The energy confinement is then determined by the diffusion coefficient of the core plasma.

Keywords:

tokamak, stellarator, divertor, bundle divertor, high temperature divertor plasma, improvement of energy confinement.

1. Introduction

Recent progress in the tokamak experimental research has led to design studies of reactor grade devices such as ITER¹⁾. These studies have identified two critical issues, which have to be resolved before committing construction of these devices. They are: (1) deterioration of the energy confinement with increasing input power, (2) difficulty in handling the heat removal. The latter issue has been regarded as an engineering issue in the present research. But a significant modification of the boundary magnetic structure may be needed to resolve this. In this paper, we propose high temperature divertor plasma operation which may resolve these key issues simultaneously. In section 2, we review the tokamak divertor from these considerations and discuss possible scenarios to resolve these issues and the discussions there lead to the high temperature divertor plasma operation. In section 3, the basic idea of the high temperature divertor operation is described.

2. Existing Boundary (Divertor) Scenarios

The ITER design study has identified that high heat flux on the divertor tiles causes many engineering problems such as erosion of the carbon tiles and corrosion of the cooling pipe. These problems are further complicated since the tiles are subject to occasional enormous heat fluxes during the plasma disruption. Frequent replacement of these tiles will be an engineering nightmare. One approach to mitigate these problems is to convert the heat flux from the confining region into the impurity radiative power at the boundary (so called radiative boundary). This necessitates large volume of the cold boundary, which can be formed by large expansion of the edge flux surface region or divertor channel (expanded boundary concept²⁾, as

Fig. 1. Expanded boundary divertor proposed in ref. 2 (Fig. 4).

Fig. 2. Radiative ergodic boundary.

illustrated in Fig. 1. The proposal of the expanded boundary poloidal divertor led to the Doublet III experiment, which demonstrated that such a concept works at least under certain present experimental conditions. In a reactor grade device, the input power increases, the thickness of the scrape-off layer decreases due to the expected higher temperature and the acceptable level of the impurity concentration in the core region is low. Considering these factors, this approach may not function unless a significant expansion is achieved by an elaborated poloidal coil system, or there exists a favorable impurity transport, which leads to accumulation of the impurity in the edge. In this context, expansion of the edge flux surfaces by the resonant magnetic field (ergodic boundary) is more promising (Fig. 2)²⁾ and is implemented easily because of small required coil currents. The pioneering work of the ergodic boundary on TEXT has demonstrated that the edge temperature can be reduced by the ergodic field structure, while the core plasma being unaffected³⁾. The cold boundary plasma will be an effective radiator. These approaches may contradict with existence of a stable H-mode⁴⁾. Since H-mode discharges exhibit significant improvement in the particle confinement, the plasma particles including the impurities accumulated in the edge of the radiative boundary will be absorbed in the confining region and thus the core plasma density and impurity radiation may become uncontrollable.

Bundle divertor⁵⁾ were once considered to be a promising heat removable concept, as illustrated in Fig. 3. The outward heat flux is guided to the outside of the toroidal coil cage with very large expansion of the divertor channel. This feature is very attractive in terms of heat removal at the divertor plate and maintenance of the tiles. This approach can also provide a good radiative boundary with large volume, but compatibility with H-mode may be problematic, as discussed above.

Fig. 3. Bundle divertor proposed for INTOR (M. F. A. Harrison *et al.*, J. Nucl. Mater. 93&94 (1980) 454).

Drawbacks of this approach are (1) a significant portion of the outer closed surface region is destroyed and (2) canceling toroidal field strength locally requires a large coil current for the bundle divertor coils.

These drawbacks of the bundle divertor may be mitigated in the configuration in Fig. 4.

Fig. 4. An example of the tokamak divertor configuration, which allows high temperature divertor operation.

In this configuration, a D-shaped tokamak plasma are connected to two small, horizontally elongated plasmas (doublet), which serves to guide the outward particle and heat fluxes near the outer toroidal coil legs. The toroidal coils are larger compared with the conventional tokamak design to accommodate the guiding doublet plasma and the required number of the toroidal coils, however is much smaller. This allows enough space between the toroidal coil legs to accommodate a bundle divertor which guides the outward particle and heat fluxes from the doublet plasma region into the outside of the toroidal coil cage, where low magnetic field allows large expansion of the divertor flux tube. The magnetic configuration is rather complicated and large, but if the energy confinement is enhanced, as described below, then the required size of the plasma becomes smaller, compensating for the geometrical disadvantage of the coil system. Shaping of such a configuration has already been established in GA Doublet program ⁶⁾ and can be feasible. Plasma current in the guiding doublet is driven by RF waves. The density there can be very low and thus efficiency of the current drive can be high.

Such a bundle divertor concept may also be applied to helical devices. Fig. 5 is an existing helical device with many divertor bundles which guide the heat and particle fluxes from the core to the outside of the coil cage. With appropriate modification, the bundle divertor flux tubes may be expanded and the number of the bundles may be reduced. For the helical case (torsatron/heliatron type), bundling and expansion of the divertor flux tube may be easier since the separatrix of the inherent helical divertor geometry extends beyond the coil system compared with tokamak. Presently we are making efforts to find such configurations.

3. High Temperature Divertor Operation

A high temperature divertor plasma is achieved by efficient pumping of the neutrals recycled at the divertor plate and hence by avoiding cooling of the divertor plasma by the cold particles, generated at the target plate. This is not a new concept of the divertor operation, rather a classical one. Its many advantages have been neglected because the pumping is not easy

Top view of the coil assembly of the IMS.

Diverted magnetic field line flux (from a 5 mm launch distance) at the coil minor radius for the IMS magnetic field case with no externally applied magnetic fields.

Fig. 5. IMS device, a helical device with many bundle diverted flux tubes (D. T. Anderson *et al.*, IEEE Trans. Plasma Sci, PS-9 (1981) 212.).

Fig. 6. Two distinctly different operations of the divertor.

for the conventional toroidal devices. The most important advantage is that it may overcome the deterioration of the energy confinement. To explain how the energy confinement can improve with this operation, we first describe two extreme divertor operations, high and low divertor temperature operational modes. Fig. 6 (a) is the low temperature mode, which is the present experimental operational mode. The temperature is low at the edge and thus the plasma energy is confined by maintaining ∇T . In this operational mode, the thermal diffusivity (χ) determines the energy confinement. Since the particle source is located at the edge, the density profile is generally flat and the diffusivity (D) does not affect the energy confinement. Unfortunately, it has been observed that χ increases with increasing input power, causing severe deterioration of the energy confinement. Fig. 6 (b) is the high temperature divertor operation mode, in which the edge temperature is kept high by the pumping and the peaked density profile is maintained by a combination of deep fuelling such as pellet or neutral beam injection and pumping of the recycled neutrals. In this case, D and hence particle confinement determines the energy confinement (for simplicity, we assume that χ is infinite). Even though D has been observed to increase with increasing input power similarly as χ , the observed values of D are generally much lower than those of χ . As shown later in Eq(2), if $D < \chi/\gamma$ (γ : the total transmission coefficient at the sheath boundary), then the energy confinement in such an operational mode is higher than those of conventional low edge temperature discharges. Recently T. Ohkawa⁸⁾ has also emphasized importance of the reduction of parallel electron heat loss in the scraped-off layer for overall confinement improvement. He has proposed a thermal dike which prevents the influx of cold electrons from entering the scraped-off layer surrounding the main plasma, thereby avoiding the plasma cooling there. The thermal dike is created by RF fields.

In a torsatron/Heliotron device, the neoclassical loss due to helical ripples is also a concern. The expected parameter space for a helical reactor plasma is that the ions are in the ν regime where the poloidal rotation due to the radial electric field reduces the ripple loss significantly. The electrons are in the $1/\nu$ (ripple loss dominant) regime and hence the electron neoclassical thermal diffusivity (χ) is high, thereby limiting achievable temperature. However, the diffusion process is ambipolar and thus determined by the diffusion of slower species (in this case, the ions) and hence the effective D is substantially low.

To estimate the divertor temperature, we consider the following power and particle balances in the divertor:

The temperature is given by the power balance at the divertor plate.

$$Q = \gamma \cdot \Gamma \cdot T \rightarrow T = Q/(\gamma \cdot \Gamma) = Q \tau_p / (\gamma \cdot n \cdot V) \quad (1)$$

where Q and Γ are the total heat flux and particle flux on the divertor plate, respectively, n and V are the average density and the volume of the main plasma respectively and τ_p (particle confinement) is defined as $\tau_p = nV/\Gamma$.

The energy confinement (τ_E) with $\chi = \infty$ is given as

$$\tau_E = 3 nTV/Q = 3 \tau_p / \gamma \quad (2)$$

From these simple expressions, τ_p determines the energy confinement and hence $D(r)$ and $S(r)$ (the spacial distribution of the particle source) are the key parameters for the energy confinement. The neutral particles recycled at the divertor plate are needed to be pumped out and be fuelled by deep particle penetration methods such as pellet or neutral beam injection to enhance τ_p and hence τ_E .

For a simple example, we consider a steady-state discharge which is heated and fuelled by neutral beam alone. We assumed that the pumping efficiency of the divertor is η , i. e., a fraction (η) of the particles reaching the divertor plates are pumped out and the same amount of the particles are fuelled by the neutral beam injection. The rest of them are ionized in the very edge region and we assume that they do not contribute to fuelling of the core plasma. The temperature for this case is $T = W \cdot \eta / \gamma$, where W is the beam energy. The energy confinement is $3 \tau_p^* \eta / \gamma$ where τ_p^* is the core particle confinement. For a parameter set ($W \sim 200$ keV, $\gamma \sim 10$, $\eta \sim 0.5$), T becomes as high as 10 keV.

The edge temperature may not need to be raised to a very high value, comparable to the central temperature, as illustrated in Fig. 6 (b) if relatively high edge temperature leads to improvement of the core confinement as seen in the improved confinement (H-mode) discharges. In the H-mode discharges, both energy and particle confinements in the core gradually improve after pedestal formation in the temperature and density profiles. Typical temperatures at the pedestal of the H-mode discharges are between 200 eV and 1 keV. Such a range of the temperature can be achieved relatively easily even outside the outermost closed surface provided that the natural recycling be avoided by efficient pumping.

4. Key Issues of High Temperature Operation

In this high temperature divertor operation, plasma-wall interaction is very critical. The expansion of the divertor channel is the most important in solving many conceivable problems. The required expansion may be ~ 500 or so, reducing the heat flux down to 5 MWm^{-2} , a level of the heat flux which can be handled reliably by the present technology. The expansion also reduces the density in front of the divertor plate down to below 10^9 cm^{-3} . Then the ionization mean free path becomes quite long $\sim 1000\text{m}$, reducing the probability of the ionization in the divertor channel significantly. Pumping is essential for this high temperature divertor operation. The long guiding structure between the edge of the main plasma and the divertor target plate make the pumping relatively easy and thus the pumping efficiency of $\sim 80\%$ may be achievable.

There are other concerns in some of wall plasma interactions.

(1) Secondary Electron Emission: The secondary electrons emitted from the divertor plate are also a source of the cold particles, which lower the divertor plasma temperature. This effect can be included in γ and γ is 7.8 without the secondary electron emission and is ~ 10 when the secondary emission rate is 0.7. But when it exceeds 0.7, γ increases rapidly and saturates at $\gamma = 23$ due to space charge limit of the secondary electron emission⁸⁾. At high incident ion energy, the secondary electron emission by ion becomes high and influences the sheath boundary condition. It is typically ~ 1.0 at the incident ion energy of 10 keV. For this case, the emission rate by electron must be below 0.4 for γ to be below 10. Selection of the divertor plate material could be important.

(2) Physical Sputtering: Generally sputtering yield of material by D^+ ion increases with incident energy and reaches a mild peak at a certain value ($\sim 300\text{eV}$ for carbon, $\sim 1000\text{eV}$ for stainless steel) and then decreases gradually. Considering acceleration of the incident ion at the sheath potential ($3.5 T_e$), plasma with temperature of $30 \sim 300\text{eV}$ has a maximum sputtering yield for the carbon target. The temperature in the present divertor operation is within this temperature range and thus discharges are operated with nearly maximum sputtering yield condition (however, the temperature for high density divertor operation is typically below 10eV and the sputtering yield is substantially low). For the high temperature divertor operation, the temperature is beyond 1 keV and thus sputtering yield of the wall material is even lower, compared with conventional non-high density divertor operations. Furthermore the total flux to the divertor plate is more than two orders of magnitude lower, thus reducing the erosion rate by sputtering substantially.

(3) Unipolar Arcing : It has been believed that unipolar arcing occurs, generating copious impurities from the divertor plate when the temperature becomes a few hundred eV. It is not clear that it occurs for the parameters of our interest, very low particle and energy fluxes, compared with the conventional divertor design. Recent JET results (although preliminary)⁹⁾ are encouraging for our proposed operation. For a low density JET H-mode discharge, the ion and electron temperatures beyond the separatrix are found to be very high, ~ 3 keV and ~ 300 eV, respectively. This suggests that unipolar arking do not occur in high temperature divertor operation, at least at this temperature range. The observed significant difference between the electron and ion temperatures appears to be due to that in heat transmission coefficient. The ion heat transmission coefficient is ~ 2 and the electron transmission coefficient may increase up to ~ 20 due to secondary electron emission.

Because of the low density due to expansion of the divertor channel, the Debye length is ~ 10 cm and with an elaborate target system, the problems mentioned above could be ameliorated and even some of the power can be recovered by a direct conversion system.

In summary, we have proposed high temperature divertor plasma operation, using a modified bundle divertor. The proposed operation may eliminate the heat removal problem and enhance the energy confinement of the core plasma.

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REFERENCES

- 1) ITER Conceptual Design : Interim Report, International Atomic Energy Agency, Vienna, 1990.
- 2) N. Ohyabu : Nucl. Fusion **21** (1981) 519.
- 3) N. Ohyabu : J.S. DeGrassie, N. Brooks, T. Taylor and H. Ikeji : Nucl. Fusion **25** (1985) 1684.
- 4) F. Wagner, G. Becker, K. Behringer, D. Campbell, and A. Eberhagen *et al.* : Phys. Rev. Lett. **49** (1982) 1408.
- 5) P.E. Scott, C.M. Wilson and A. Gibson : Nucl. Fusion **17** (1977) 481.
- 6) J.C. Wesley, T. Angel, C.J. Armentrout, D.R. Baker and F.P. Blau *et al.* : in *Plasma Physics and Controlled Nuclear Fusion Research (Proc. 8th Int. Conf. Brussels, 1980)* Vol. 1, IAEA, Vienna (1981) 35.
- 7) T. Ohkawa : "Thermal Dike Concept for Plasma Confinement" , Kakuyugo Kenkyu **64** (1990) 305.
- 8) G.D. Hobbs and J.A. Wessen : Plasma Physics **9** (1967) 85; A summary on this subject is given in review paper by P.C. Stangeby and G.M. McCracken (Nucl. Fusion **7** (1990) 1252).
- 9) P. Thomas : "Edge Plasma Measurement in JET H-modes", presented at H-mode workshop 10-12 June, 1991, at JET.