ISSN 0915-633X

NATIONAL INSTITUTE FOR FUSION SCIENCE

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(Received – Jan. 16, 1991)

NIFS-77

Feb. 1991

RESEARCH REPORT
NIFS Series

NAGOYA, JAPAN

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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. In this approach, the heat flux is guided through divertor channel to a remote area with a large target surface, resulting in low heat load on the target plate. This allows pumping of the particles escaping from the core and hence maintaining of the high divertor temperature, which is comparable to the core temperature. The energy confinement is then determined by the diffusion coefficient of the core plasma, which has been observed to be much lower than the thermal diffusivity.

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 the reactor grade devices such as ITER [1]. These studies have identified two critical issues, which have to be resolved before committing construction of these devices. They are (1) problem of the deterioration of the energy confinement with increasing input power and (2) that of heat removal. The latter issue has been regarded as an engneering 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 a high temperature divertor plasma operation which may resolve these key issues simultenuously. 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 high temperature diverter plasma operation. In section 3, the basic idea of the high temperature divertor operation is described.

2. Existing boundary (divertor) scenarios

ITER design study has identified that high heat flux on the divertor tiles causes many engneering 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 flux during the plasma disruption. Frequent replacement of these tiles will be an engneering 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 concept [2]) ,as 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 device, the input power increases, the thickness of the scrape-off layer decreases due to the expected high temperature and the acceptable level of the impurity concentration in the core region is low. Considering these factors, this approach may not function unless the significant expansion is achieved by an elaborated poloidal coil system, or there exists a favorable impurity transport, which causes accumulation of the impurity at the edge. In this context, expansion of the edge flux surfaces by the resonant magnetic field(ergodic boundary) is promising (Fig.2) [2] and is implemented easily because of small required coil currents. The pioneering work of the ergodic boundary on TEXT tokamak has demonstrated that the edge temperature can be reduced by the ergodic field structure, while the core plasma being unaffected [3]. The cold boundary will be an effective radiator. These approach may contradict with existence of stable H-mode [4]. Since the H-mode discharge exhibits significant improvement in the particle confinement, the plasma particles including 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 divertor [5] 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 coils with very large expansion of the divertor channel. This feature is very attractive in terms of the heat load on the divertor plate and its maintenance.This approach can also provide a good radiative boundary with large volume, but compatibility with H-mode may be problematic, as discussed above. 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. 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 to near the outer toroidal coil legs. The toroidal coils becomes larger compared with the convential tokamak design to accomodate the guiding doublet plasma and the required number of the toroidal coils, however becomes smaller. This allows enough space between the toroidal coil legs to accomadate a bundle divertor which guides the outward particle and heat fluxes from the doublet plasma region into outside 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 descibed below, then the required size of the device becomes smaller , compensating the geometrical disadvantage. Shaping of such a configuration has already established in GA Doublet program [6] and can be feasible. The plasma current of the guiding doublet is drived by RF waves. The density there can be very low and the efficiency of the current drive can be high.

Such a bundle divertor concept can be applied to the helical device. Fig.5 is a existing helical device with many divertor bundles which guide the heat and particle fluxes from the core outside the coil cage. With appropriate modification, the bundle divertor flux tubes may be expanded and number of the bundles may be reduced. For helical case(torsatron/heliotron 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.

3. High temperature divertor operation

High temperature divertor plasma is achieved by efficient pumping of the neutrals recycled at the divertor plate and hence by avoiding cooling 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 for the conventional toroidal devices. The most important advantage is that it may overcome deterioration of the energy confinement. To explain how the energy confinement can improve with this operation, we first describe two extreme operations, high and low divertor temperature modes. Fig.6(a) is the low temperature mode ,which is close to the present experimental operation mode. The temperature is low at the edge and thus the plasma energy is confined by maintaining ∇T. The thermal difffusivity(χ) determines the energy confinement. Since the particle source is located at the edge, the density profile is generally flat and value of the diffusivity 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 the pumping. 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 . In a torsatron/Heliotron devise, neoclassical loss due to helical ripple 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 and the electrons are in the $1/\nu$(ripple loss dominant) regime. Thus the electron neoclassical thermal diffusivity (χ) is high, limiting achievable temperature. However, the diffusion process is ambiplolar and thus determined by the diffusion of slower species(in this case, the ions) and hence the effective D is substantially low.

The power and particle balances in the high temperature divertor operation is as follows:

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

$$Q = \gamma \,.\, \Gamma \,.\, T \qquad \rightarrow \quad T = Q \,/\, (\gamma \,.\, \Gamma)$$

$$= Q\, \tau_p \,/\, (\gamma \,.\, n \,.\, V)$$

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

The energy confinement (τ_E) is given as

$$\tau_E = 3\ n\ T\ V \,/\, Q = 3\ \tau_p \,/\, \gamma$$

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 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 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 so that we assume that they do not contribute to fuelling of the core plasma. The temperature for this case is $T = W. \eta / \gamma$,where W is the beam energy. The energy confinement is $3 \tau_p \eta / \gamma$ where τ_p is the particle confinement which takes into account the beam feulling. For a parameter set (W ~ 200Kev, γ ~ 10 , η ~ 0.5), T becomes as high as 10keV.

In this high temperature divertor operation , the wall plasma interaction is very critical. The expansion of the divertor channel is the most important to handle the various problems. The required expansion may be ~500 or so ,reducing the heat flux down to 1 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 ~1000 m , reducing the probability of the ionization in the divertor channel significantly. The 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 more than 90 % pumping efficiency may be achieved.

There are concerns in some of the wall plasma interactions.

(1) secondary electron emission: The secondary electrons emitted from the divertor plate are 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 emssion rate is 0.7. But when it exceeds 0.7, γ increases rapidly [7]. At high incident ion energy, the secondary electron emission by the 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 the electron must be below 0.4 for γ to be below 10. Selection of the divertor plate material could be important.

(2) Sputtering : Since the incident energy is high, the sputtering rate of the wall material is high , compared with the case of the low temperature divertor operation. But the total flux to the divertor plate is more than two order of the magnitude lower, reducing the erosion rate by sputtering substantially. Furthermore, the divertor plates are located far away from the core plasma region and thus replacement of the plates will be easier.

(3) Unipolar arking: It has been suggested that unipolar arking occurs, generating copious impurities from the divertor plate when the temperature becomes high. It is not clear that it occurs for the parameters of our interest, very low density and low particle flux, compared with the conventional divertor.

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

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

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Fig. 1 Expanded boundary divertor proposed in ref. 2 (Fig. 4).1010

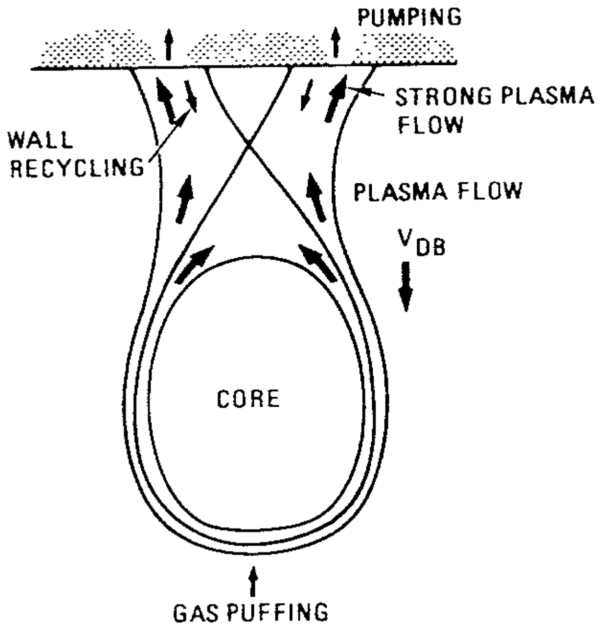
Fig. 2 Radiative ergodic boundary.

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

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

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

Fig. 6 Two distinctly different operations of the divertor.



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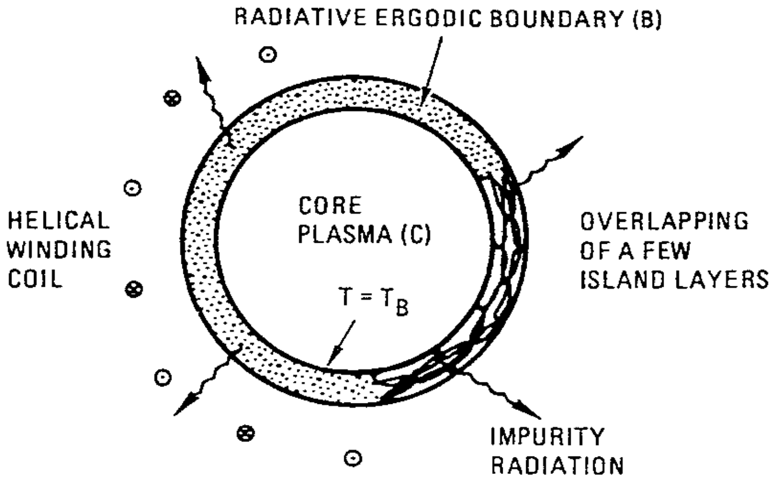
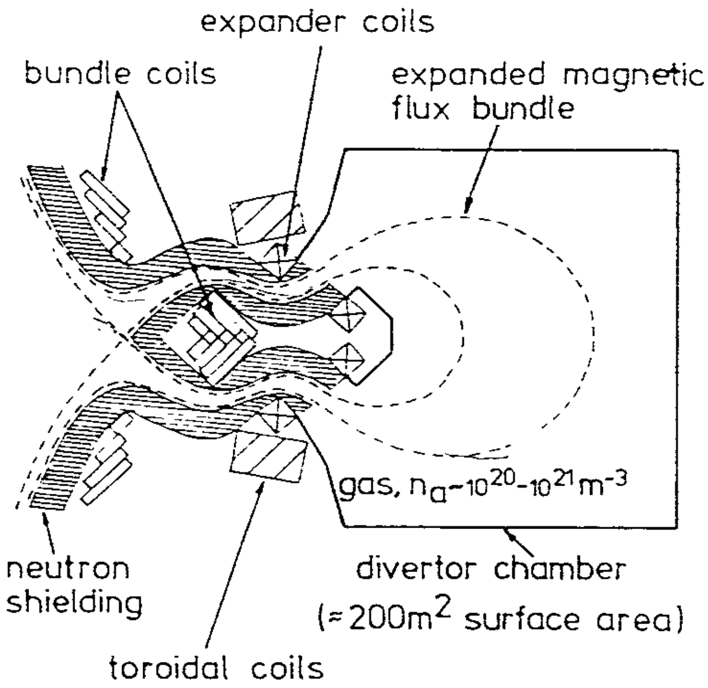


Fig. 2 Radiative ergodic boundary.



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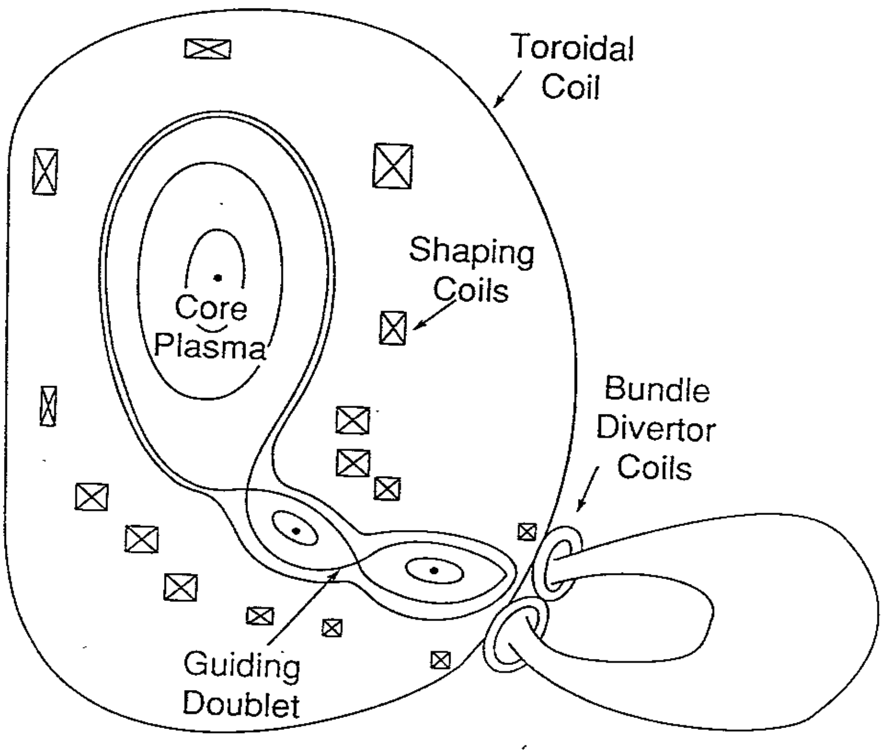

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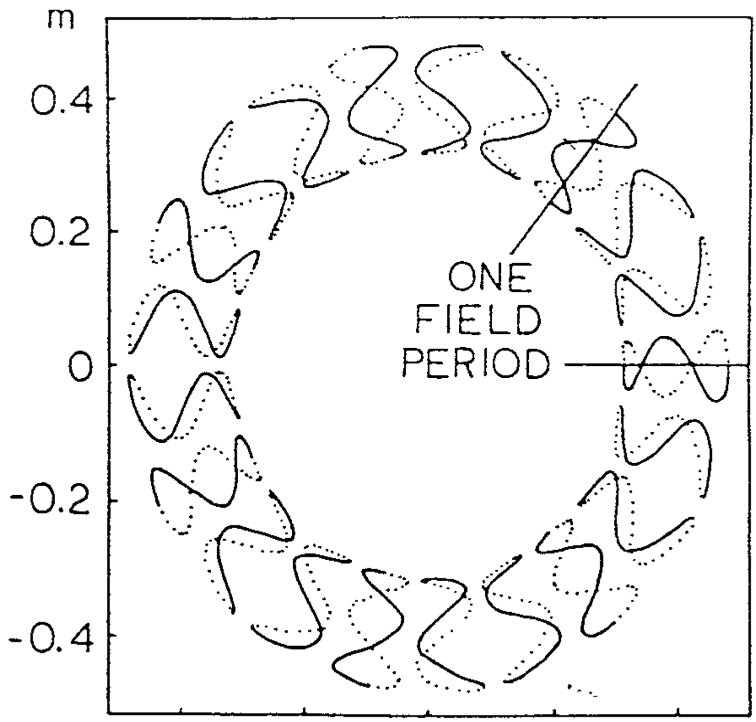
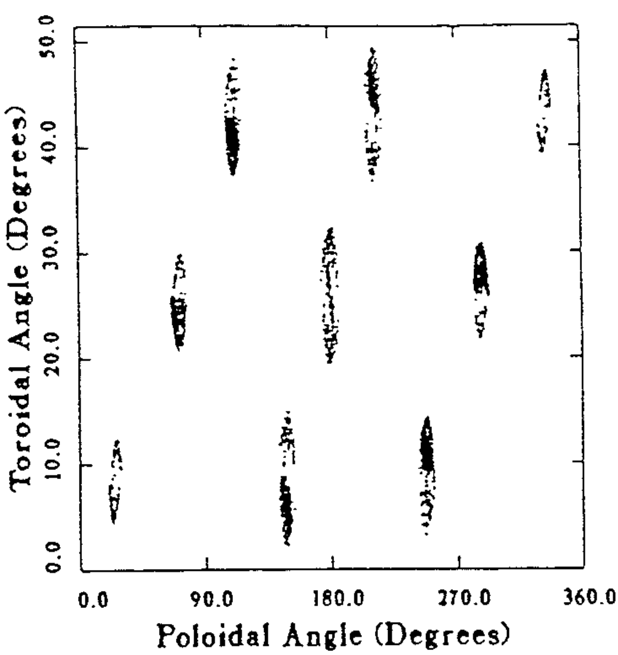


Top view of the coil assembly of the IMS.



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 (Anderson, D.T., et al., IEEE Trans. Plasma Sci. PS-9 (1981) 212.).

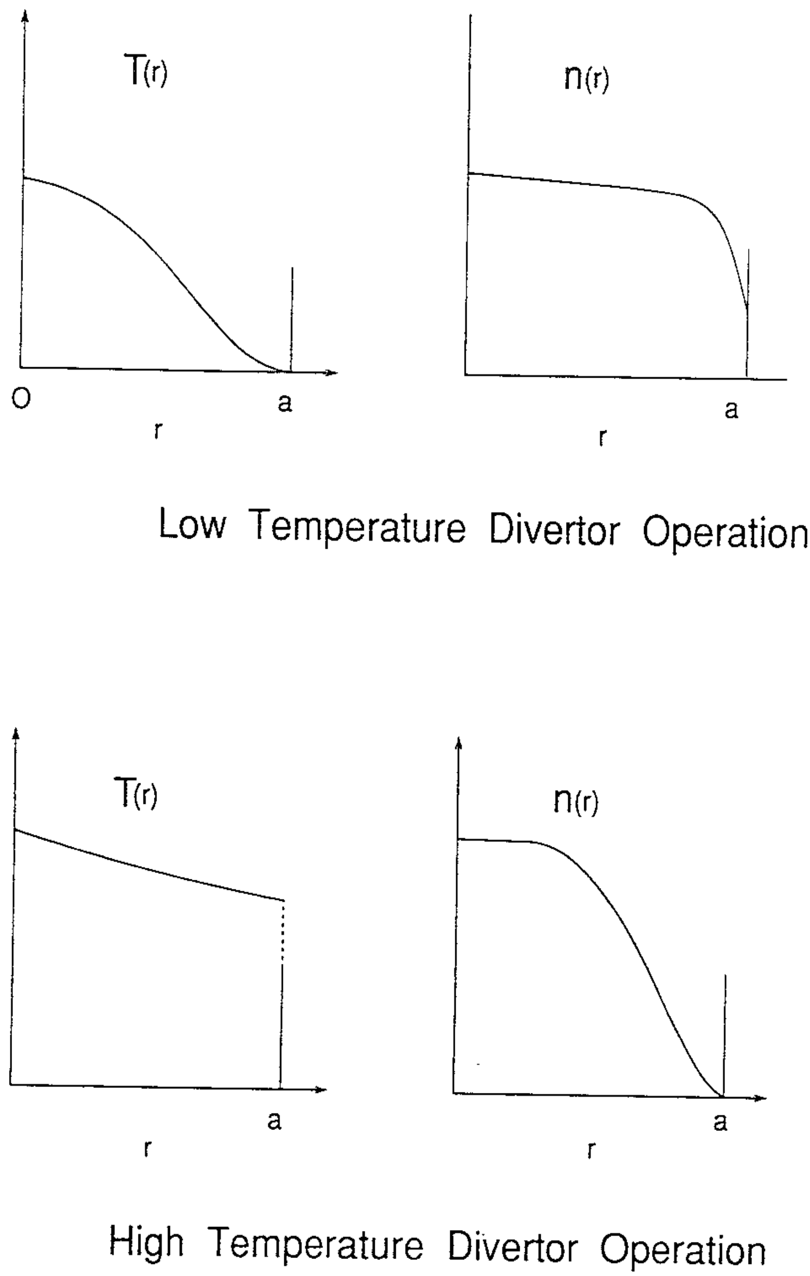


Fig. 6 Two distinctly different operations of the divertor.

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