Chapter I. Edge plasma issues in magnetic fusion devices

Today, there are two major classes of magnetic confinement devices: stellarators and tokamaks. The main difference between them is the way to form the helical structure of magnetic field lines, which is needed to compensate unwanted drift of charged particles in a curved and inhomogeneous magnetic field and to form nested magnetic flux surfaces necessary for plasma confinement (see the magnetic field structure in a tokamak in Fig. I.1). In a stellarator, which has no toroidally symmetric magnetic field, this is done by a complex shaping of the magnetic coils, whereas in a toroidally symmetric tokamak by toroidal current flowing through the plasma.

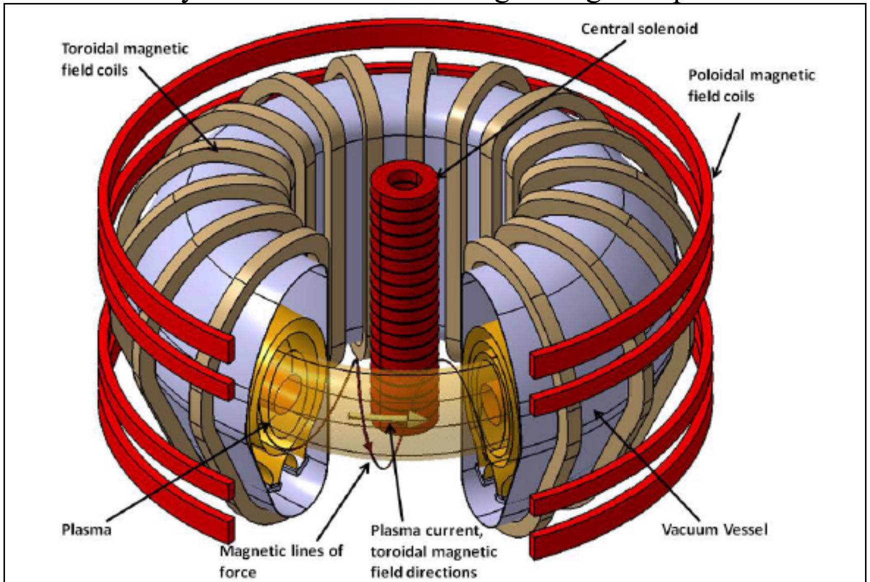

Fig. I.1. Schematic view of the structure of helical magnetic field lines (produced by both magnetic coils and plasma current) which form nested magnetic flux surfaces in a tokamak. Reproduced with permission from [1], © IAEA 2012.

As already mentioned in the Preface, the negative impact of the interaction of the hot plasma with the plasma-facing components (PFC) of the vacuum chamber on reactor performance was envisioned at the very beginning of the fusion era. The main identified issues were i) contamination of the core plasma with the eroded PFC material beyond the acceptable level (which actually is very low for high-Z impurities) where the plasma radiation loss due to impurity exceeds the fusion power released in alpha-particles so that no self-sustained fusion burn becomes possible, and ii) strong erosion of the PFC material, which can severely limit the lifetime of the fusion reactor and make it unfeasible.

L. Spitzer and Tamm and Sakharov suggested two conceptually different solutions to this problem in the 1950th. The Spitzer's idea (e.g. see [2] and the references therein) was to isolate as much as possible the region of intense plasma-wall interaction from the core plasma. For this he suggested to use special magnetic coils to divert the magnetic field lines at the edge of the magnetic fusion device into some partially closed volume – the divertor (see Fig. I.2). As a result, the magnetic field lines in the core and in the edge become separated by the so-called separatrix, whereas the impurity flux from the divertor into the core plasma is suppressed due to both a rather narrow divertor throat and plugging with the plasma flowing into the divertor.

The idea of Tamm and Sakharov was to form a cushion of neutral gas in front of the PFCs, so that the hot plasma arriving from the core would be cooled down in the course of the interaction with the neutrals so that the temperature of the plasma interacting with the material surface would be so low that virtually no material erosion would be possible [3].

Interestingly, today's concept of handling the issue of plasma interaction with the PFCs in magnetic fusion reactors is essentially a symbiosis of these two fundamental ideas adapted to the particular features of the magnetic devices.

The original Spitzer's divertor design was focused on a stellarator magnetic configuration (e.g. see [2]), where, because of the complexity of magnetic geometry, the formation of divertor was only possible by the reversal of the strong "toroidal" magnetic field. Nonetheless, the implementation of such a divertor in B-65 stellarator [4] has demonstrated encouraging results showing a significant impurity reduction in the core plasma.

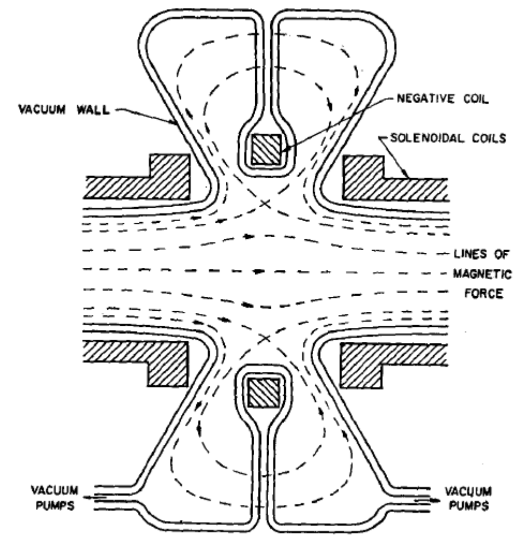

Fig. I.2. Design of the divertor suggested by Spitzer for the stellarator. Reproduced with permission from [2], © AIP Publishing 1958.

In much simpler, toroidally symmetric magnetic geometry of a tokamak, the divertor (the so-called poloidal divertor) can be relatively easy formed by proper arrangement of the electric current(-s) in additional (or even existing) magnetic coils, reversing some relatively weak poloidal magnetic field.

However, experimental studies of the impact of poloidal divertors on tokamak performance have started only at the end of 1970^{th}. Before that, the most common approach was the designation of some parts of the tokamak PFC – the "limiters" - to handle the plasma-material interaction. For example, Fig. I.3a shows the sketch of a toroidally symmetric limiter which is designated to accommodate the most severe plasma interaction with the PFCs. The so-called "last closed magnetic flux surface" (LCFS) separates the nested, closed magnetic flux surfaces in the core, which are occupied by hot fusion grade plasma, from the open ones where the magnetic field lines intersect the PFC material.

The schematic view of the poloidal cross-section of magnetic configuration in a tokamak with the simplest poloidal divertor is shown in Fig. I.3b. Such magnetic configuration can be formed just by adding a toroidally symmetric magnetic coil under the divertor targets, which carries the electric current in the same direction as the electric current in the plasma. These two currents create the magnetic separatrix that plays the role of the LCFS for the case of the toroidal limiter and separates the closed and open magnetic flux surfaces. Under the X-point, where the total poloidal magnetic field vanishes by definition, there is a so-called "private flux" region (PFR) having a very limited connection to the core plasma. Due to cross-field plasma transport, the heat from the core comes to the "scrape-off layer" (SOL) plasma where it can reach divertor targets quickly due to fast plasma transport along the magnetic field lines. The region between the X-point and the divertor targets is called the "divertor volume" or just a "divertor" and is often used for the designation of the whole ensemble of the PFR and the "outer" and "inner" divertors located, respectively at the outer and inner sides of the torus.

The footprints of the heat flux at the targets are determined by the competition of fast plasma transport along the magnetic field lines and relatively slow cross-field plasma transport. As a result, the footprints appear to be small and all estimates show that if in fusion reactor all the power Q_{SOL} coming into the SOL from the core would reach the targets, the maximum heat load on the targets would greatly exceed the tolerable level. Therefore, a large fraction of this power should be dissipated on the way to the target through the impurity and hydrogen radiation losses and this is one of the main missions of the divertors.

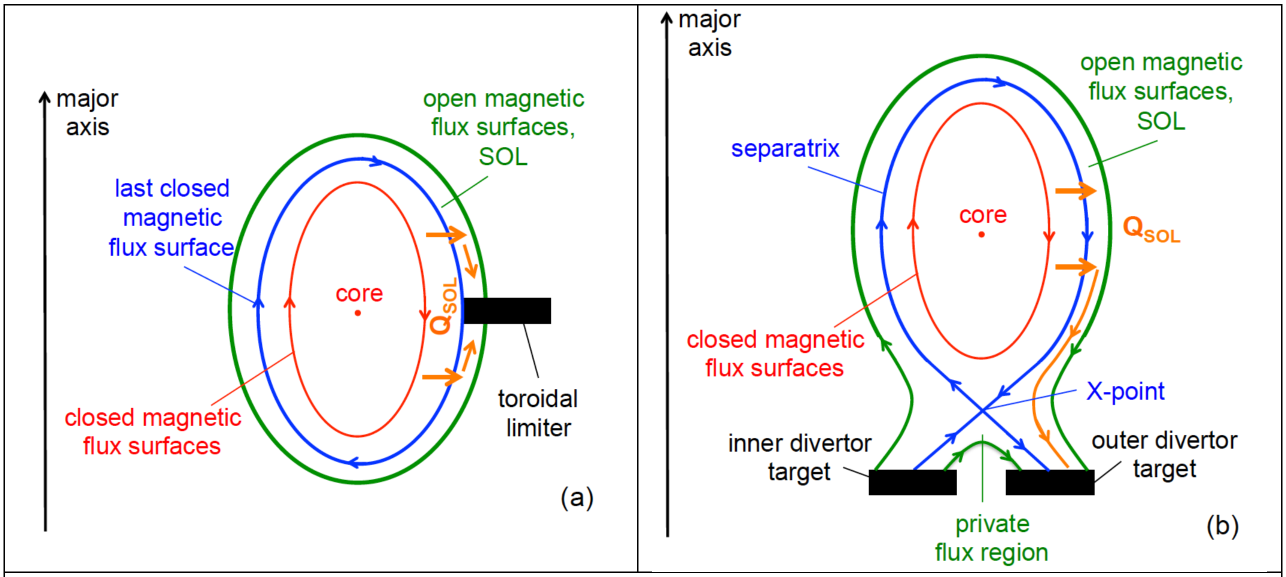

Fig. I.3. (a) Sketch of a toroidally symmetric limiter which is designated to accommodate the most severe plasma interaction with the PFCs. The LCFS that separates the nested closed magnetic flux surfaces in the core, which are occupied by hot, fusion grade plasma, from the open ones where the magnetic field lines intersect the material targets; and (b) Schematic view of the poloidal cross-section of a tokamak with a poloidal divertor formed by the plasma current and the currents in toroidal magnetic coils. In both figures, Q_{SOL} is the power coming from the core into the SOL, the red and blue arrows show the directions of the poloidal magnetic field and the orange arrows indicate the direction of the heat flux.

The first poloidal magnetic divertors were implemented in tokamaks only in the 1970s (e.g. see Fig. I.4). And, like it had been found before in the stellarators, it was demonstrated that the implementation of a divertor in a tokamak reduces the impurity content in the core plasma significantly.

Apart from that, at the beginning of the 1980s, it was discovered that divertor magnetic configuration promotes transition into new regimes of i) improved core plasma confinement, the so-called "H-mode" [7]; and ii) highly radiative divertor operation regimes with dense cold plasma and neutral gas cushion formed in the divertor region, resulting in a strong reduction of the heat loading on the PFCs (e.g. see [8], [9], [10], [11]). Since that, these new regimes became the key ingredients of the tokamak reactor designs and the main topics in the tokamak research. In some sense, these divertor operation conditions are the combination of Spitzer's divertor concept and Tamm&Sakharov's idea of the neutral gas cushion in front of the PFCs.

Such divertor regimes, called "high recycling regimes", are characterized by a very strong recirculation of neutrals and plasma in the divertor volume via neutral ionization and plasma neutralization at the divertor targets and through the volumetric recombination processes. As a result, the neutral ionization source in the divertor region in the high recycling regimes appears to be by orders of magnitude higher than the neutral puffing and pumping rates.

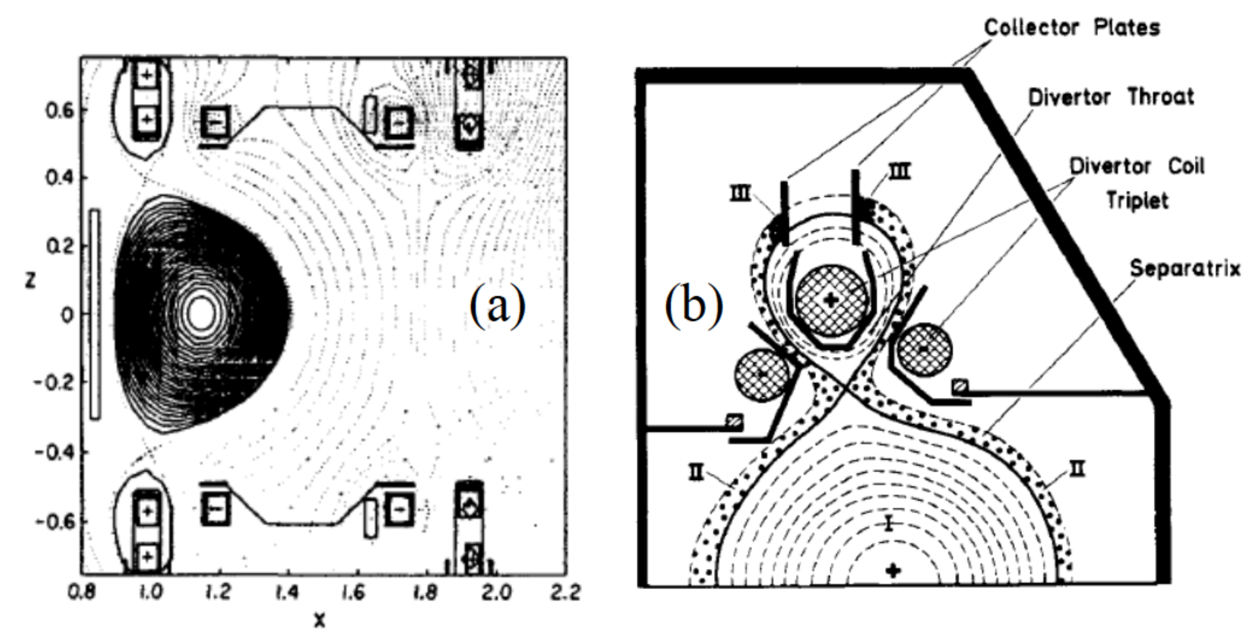

Fig. I.4. Poloidal divertor magnetic configurations in the PDX (a) (Reproduced with permission from [5], © IAEA 1981) and ASDEX (b) (Reproduced with permission from [6], © IOP Publishing 1982) tokamaks.

Further studies of divertor performance have shown that with increasing plasma density and divertor radiation loss, not only the heat load on the PFCs but also the plasma particle flux to divertor targets starts to decrease (e.g. see [12] and the references therein). In a way, it looks like the plasma detaches from the divertor targets and these regimes are called the "detached divertor" regimes. The reduction of the plasma flux to the targets allows the reduction of the power loading associated with the release of the ionization potential energy, which, for a reactor, can be very substantial and exceed the tolerable limit.

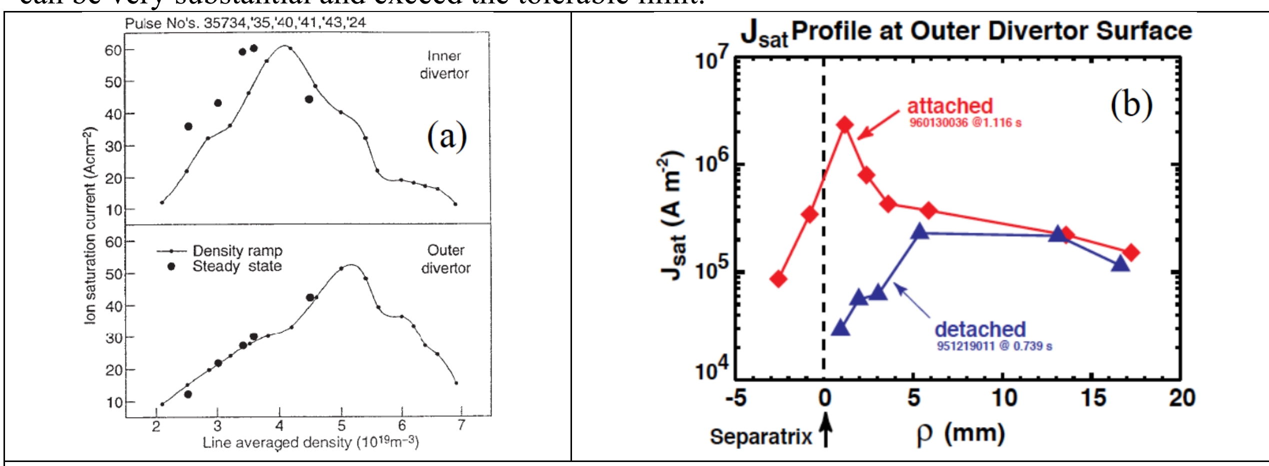

Fig. I.5. Specific plasma fluxes on (a) the inner and outer divertor targets in JET versus the line averaged plasma density (Reproduced with permission from [13], © IAEA 1998) and (b) the outer divertor target in attached and detached regimes in C-Mod tokamak (Reproduced with permission from [14], © IAEA 1999).

The examples of plasma flux reduction are shown in Fig. I.5 where the ion saturation currents, which give the specific plasma flux, measured on JET and C-Mod tokamaks are shown.

As one can see from Fig. I.5a, the plasma fluxes on both the inner and outer divertor targets increase, then saturate, and then, in the detached regime, decrease (the so-called "rollover") with increasing plasma density. However, as seen in Fig. I.5b, such a decrease can occur only on some (although initially the most loaded) part of the divertor target.

Due to the relative simplicity of the formation of the poloidal divertor configuration in tokamaks, very different magnetic configurations have been used in experiments, whereas more and, in some cases, very exotic divertor configurations have been suggested over the history of the magnetic fusion research. The main goals pursued in these developments of the divertor design are to maintain the tolerable level of the maximum target heat load together with the low plasma temperature at the targets to minimize target erosion, as well as to enhance the core plasma performance by improving plasma confinement and reducing the core plasma contamination.

For example, in Fig. I.3b, the so-called "single null" divertor configuration (having one X-point) is sketched. However, "double null" divertor configurations, having X-points above and below the core plasma, are used in some experimental studies. When the double null is exact, only one separatrix separates the core and both upper and lower divertors. In this case, the outer and inner parts of the SOL become magnetically disconnected, which has important implications for anomalous plasma transport in the inner and outer SOL regions. Such a double null configuration, along with the lower and upper single null and "near-double-null" configurations used in the C-Mod tokamak are shown in Fig. I.6a. We notice that the very first divertors were designed to allow up to 2 X-points in ASDEX and up to 4 X-points in the PDX tokamaks (see Fig. I.4).

A standard X-point magnetic geometry, which can be formed with just two effective toroidal currents, has two divertor legs and the strength of the poloidal magnetic field in the vicinity of the X-point is proportional to the distance to the X-point, r_X. However, with at least three effective toroidal currents, the X-point could produce four divertor legs and, as a result, reduce the peak heat load on divertor targets. Such a magnetic configuration can be realized with a "snowflake" divertor concept (see Fig. I.6b). In addition, in such a case, the strength of the poloidal magnetic field in the vicinity of the X-point becomes proportional to r_X^2. This increases the length of the magnetic field lines in the SOL, and, therefore, could "slow down" parallel plasma transport and, therefore, increase the footprint of the heat flux on the targets. It also results in an increase of the volume occupied by plasma in the vicinity of the X-point, which could help to increase the radiation loss from divertor.

Magnetic configurations of the TCV tokamak divertor, having a long outer divertor leg, are shown in Fig. I.6c. As one can see, manipulation of the currents in the magnetic coils can produce a very strong spread of the open magnetic flux surfaces in the divertor. More complex divertor designs are shown in Fig. I.6d and Fig. I.6e. The so-called Super-X divertor shown in Fig. I.6d has a very pronounced extension of the outer divertor leg along the major radius, whereas the even more complex X-point target divertor (shown in Fig. I.6e) in addition to the radial extension of the outer divertor leg has multiple X-points in the vicinity of the divertor targets. All these features could increase the radiation loss from the divertor and the reduction of the peak power loading of the targets. However, the complexity of such divertors can significantly limit the flexibility of the shaping of the magnetic configuration of the core plasma, which can be necessary for obtaining the best core plasma performance and maximizing the fusion yield. In addition, such divertors occupy a large volume within the toroidal magnetic field that is expensive to generate. Therefore, the usage of such divertors in future fusion reactors requires high confidence in assessments of both divertor and core plasma performance.

The high recycling and, in particular, detached divertor regimes are characterized by strongly coupled plasma-neutral interactions providing, for example, an efficient cooling channel for the plasma within ~eV temperature range where the radiation energy losses become virtually negligible. But, whereas due to fast plasma transport along the magnetic field lines, the plasma parameters can be explicitly affected by the magnetic configuration, neutral transport is not affected directly by the magnetic field (although an indirect effect, caused by the plasma parameter variation, is present). However, neutral transport can be directly impacted by the special shaping of the divertor PFCs (the so-called "closed" divertors), which can better confine neutrals in the divertor region. Therefore, in an attempt to facilitate divertor detachment, both the magnetic configuration and the geometry of the divertor material structures should be taken into account.

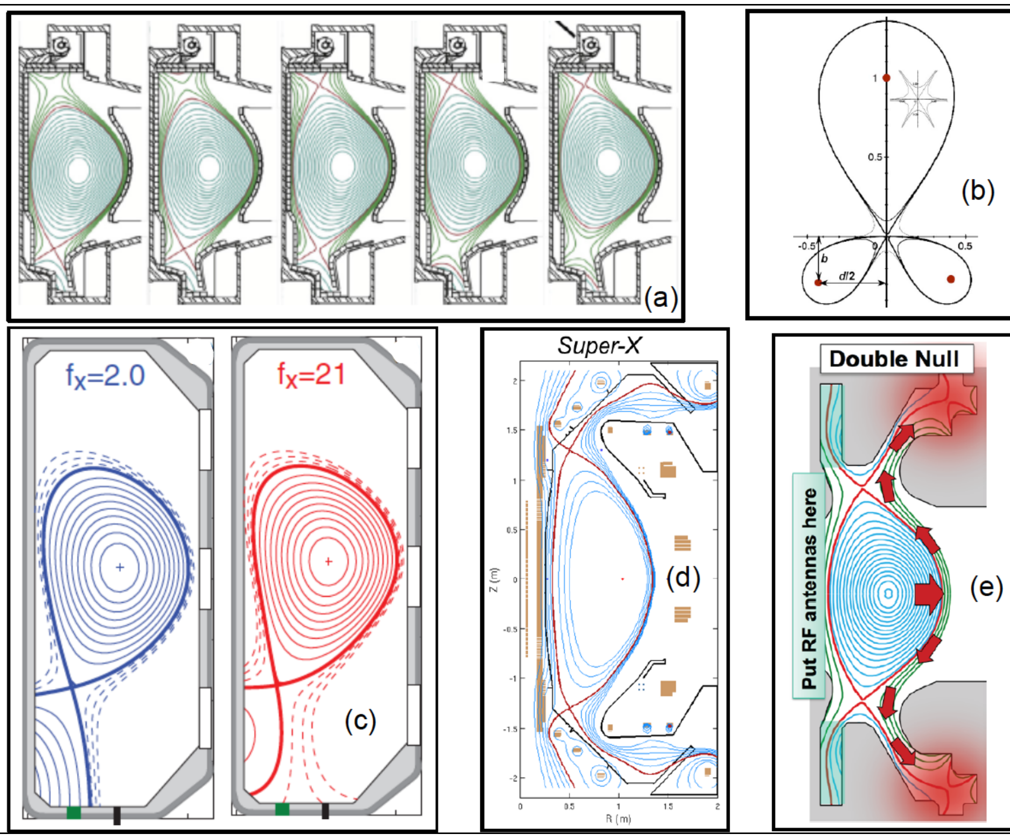

Fig. I.6. (a) Lower and upper single null, "near-double-null" and exact double null configurations used in the C-Mod tokamak (Reproduced with permissionfrom [15], © Elseivier 2017); (b) Sketch of the "snowflake" magnetic configuration (Reproduced with permission from [16], © AIP Publishing 2007); (c) Divertor configuration in the TCV tokamak with the long outer "divertor leg" and compressed (left) and expanded (right) magnetic flux surfaces in the outer divertor (Reproduced with permission from [17], © IAEA 2017); (d) So-called "Super-X" divertor configuration with a large radial extension and expanded poloidal magnetic flux in the outer divertor leg, which is reachable in the MAST-U tokamak (Reproduced with permission from [18]); (e) X-point target divertor concept suggested for the ADX tokamak project (Reproduced with permission from [19], © IAEA 2015). The thin lines show the magnetic flux surfaces.

An impact of geometrical effects on divertor detachment can be clearly seen from Fig. I.7a, where the onset of divertor detachment in the most closed, slot-like divertor geometry (Fig. I.7c) occurs at the plasma density which is significantly lower than for the most open “flat-plate” geometry (Fig. I.7d).

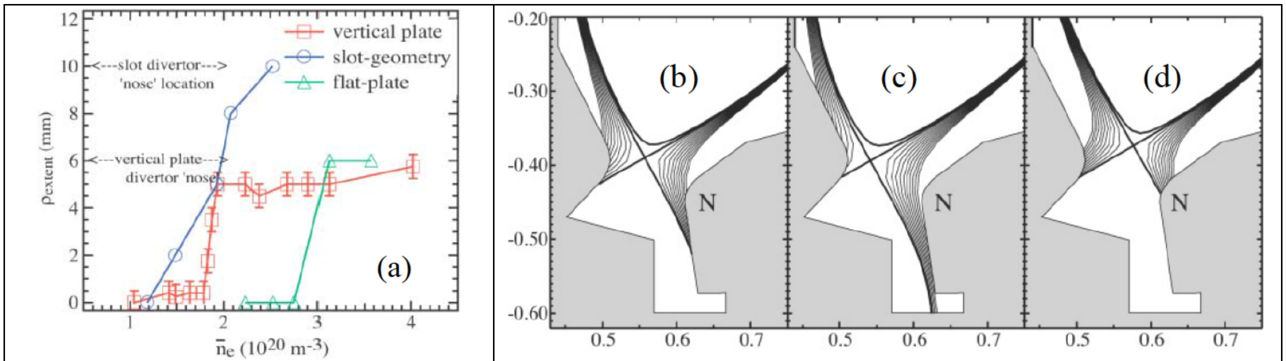

Fig. I.7. (a) The flux surface extent of divertor detachment for different divertor geometries: (b) the “vertical target”, (c) the “slot geometry”, and (d) the “flat plate”. Reproduced with permission from [20], © Taylor & Francis 2007.

The evolution of divertor geometries from the “open” to more “closed” ones, which were used in different time on JET and ASDEX tokamaks, is shown in Fig. I.8, and the impact of the closed divertor geometry on the increase of the radiation loss in the divertor volume and the reduction of the power reaching the targets is demonstrated in Fig. I.9. Both the magnetic configuration and the divertor geometry of ITER are shown in Fig. I.10. As one can see, the single-null magnetic configuration and a rather closed divertor geometry will be used there.

Summarizing this chapter, we find that the physics of the edge plasma is very complex and multifaceted. It involves i) many different species of both neutral and charged particles, including the eroded and deliberately injected atoms, molecules, and even dust particles formed due to erosion and re-deposition of the PFC material; ii) anomalous and classical (e.g. drifts) cross-field plasma transport that is complicated by X-point effects; iii) plasma transport along the magnetic field lines, which is often altered by kinetic effects (e.g. kinetic effects in plasma heat conduction along the magnetic field lines); iv) atomic physics processes playing the crucial role in the radiation losses due to impurity and hydrogenic species, plasma-neutral interactions, plasma recycling, and in establishing both the high recycling and detached divertor regimes; v) plasma interactions with the material surfaces of the PFCs, including the formation of the so-called sheath region in a close proximity to the surface, reflection and desorption of hydrogenic species, erosion and re-deposition of the PFC materials; etc.

In addition, the distribution of the plasma and neutral gas parameters in the edge plasma is very non-uniform. Whereas at the midplane of the SOL, the plasma density and temperature are $\sim 10^{13\text{-}14}$ cm^{-3} and ~100 eV, in the divertor region the plasma density can reach $\sim 10^{15}$ cm^{-3} (and even higher) and the temperature drops to ~10 eV in the attached and even to sub-eV in detached regimes. Whereas the neutral density in the midplane is well below the plasma density, in the divertor volume the neutral gas density can be comparable to the plasma one. The characteristic cross-field scale of plasma parameter variation at the midplane of the SOL is ~few mm, whereas the effective distance between the SOL midplane and divertor targets along the magnetic field (the connection length) can be ~100 m.

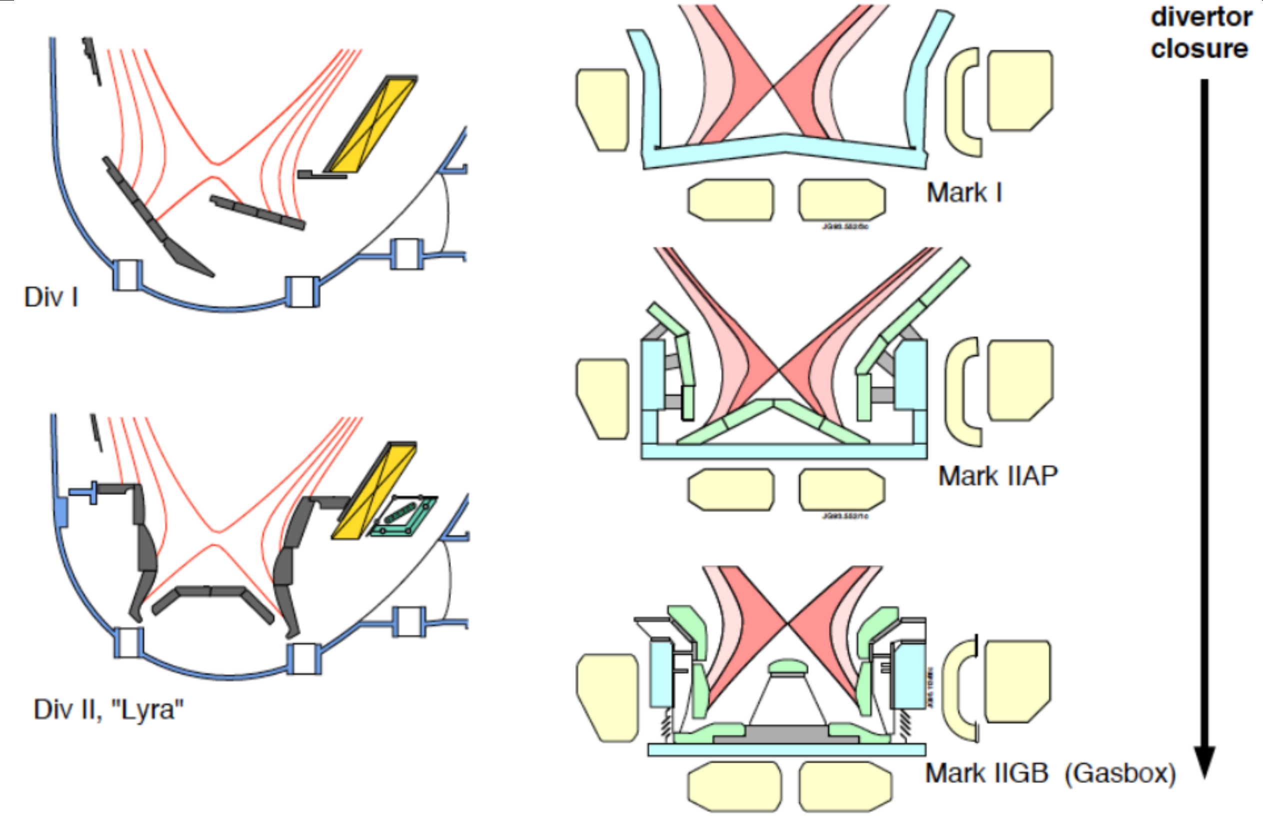

Fig. I.8. The evolution of divertor geometries from the “open” to more “closed” ones on the ASDEX Upgrade (left) and JET (right) tokamaks. Reproduced with permission from [21], © IOP Publishing 1999.

All of these make any quantitative theoretical/computational description of edge plasma phenomena very difficult. Moreover, as of today, it is not feasible to describe all processes in the edge plasma with a single “super code”. Therefore, the researchers usually separate the “fast” “micro” turbulence in the plasma and the “slow” macroscopic transport of the plasma and neutral gas and describe them with different and very complex codes. We notice that the applicability of such splitting of the processes into the turbulent and “mean-field” parts in the edge plasmas is often questionable. From the experimental side, the situation is not any simpler because of the strong non-uniformity of the plasma parameters and often a limited plasma parameter range accessible to some diagnostics (e.g. for the Langmuir probes). In addition, due to the geometrical complexity, it is often difficult to post-process the available experimental data, and incompleteness of the experimental data complicates its interpretation.

Nonetheless, by interconnecting many bits of the information coming from the experimental data, simplified theoretical models, and the results of numerical simulations, the edge plasma community has been able to build up a rather complete physical picture of many important edge plasma phenomena.

It is obvious that today it is not possible to describe in detail all aspects of the edge plasma physics because some topics are still poorly understood (e.g. modification of the PFC material under fusion plasma irradiation and the related retention of the helium and hydrogenic species). Therefore, in the following Chapters, we will discuss the main components of the edge plasma physics and phenomena in the edge plasma, which are reasonably well understood.

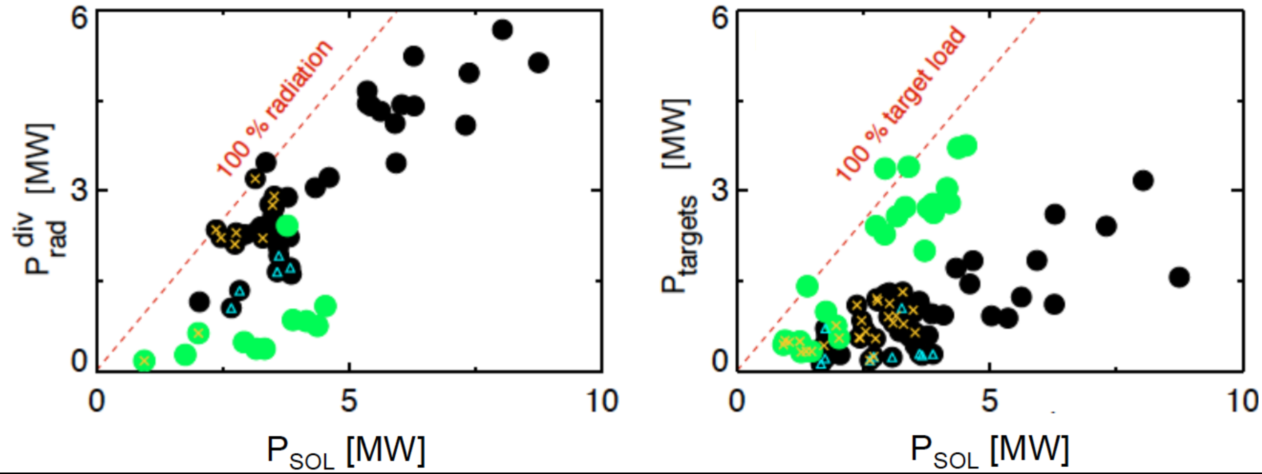

Fig. I.9. Radiation loss in divertor volume (left) and power coming to divertor targets (right) versus power coming into SOL in ASDEX tokamak for open (green circles) and closed (black circles) divertor geometries. Reproduced with permission from [21], © IOP Publishing 1999.

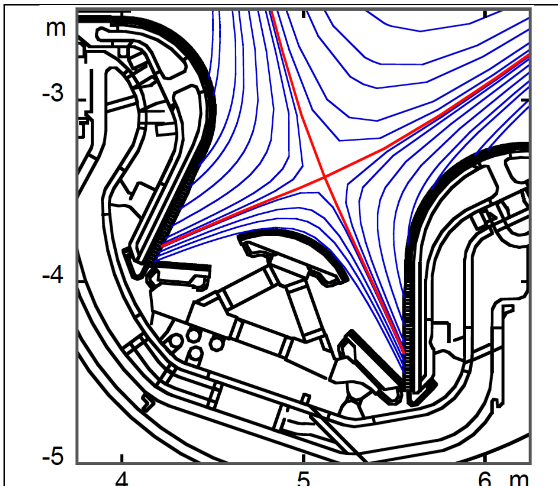

Fig. I.10. Magnetic configuration and divertor geometry in ITER.

In Chapters II and III we consider, correspondingly, the atomic physics and plasma material interaction issues relevant to the edge plasma. In Chapter IV, the basic features of the so-called sheath – a narrow region at the interface between the plasma and the material wall – are discussed. Although the sheath occupies only a tiny fraction of the whole edge plasma volume, it plays an important role in both the physics of the edge plasma and the plasma-material interaction. Chapter V is dedicated to the physics of the dust that is virtually ubiquitous in the edge plasmas. Chapter VI is dedicated to classical edge plasma transport, whereas in Chapter VII, the basic ingredients of anomalous cross-field plasma transport and the available numerical tools used for the modeling of the edge plasma turbulence are considered. In Chapter VIII, we consider the modern approaches to numerical modeling of the edge plasma transport. In Chapter IX, we discuss the physics of some macroscopic phenomena that are distinctive for the edge plasma. They include i) MARFE (which stands for the Multifaceted Asymmetric Radiation From the Edge) and poloidaly symmetric plasma detachment; ii) self-sustained edge plasma oscillations; iii) divertor plasma detachment. In Chapter X we present our assessment of the current understanding of the complex and multifaceted physics of the edge plasma and discuss the main gaps remaining there.

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