Advancing power exhaust studies from present to future tokamak devices

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Abstract:
Power exhaust is a crucial issue for future fusion devices such as ITER and DEMO. A device like DEMO despite being of a similar geometrical size of ITER will need to accommodate an about 3 to 4 times higher thermal power, aggravating the issue of power exhaust. ASDEX Upgrade with its fully tungsten covered wall, high ratio of heating power to major radius and extensive edge and SOL diagnostics is well suited for studying most aspects of power exhaust. Limiting the total power flux to the divertor target plates is only possible in the detached regime. Despite its crucial importance for safe operation of future larger devices the understanding of the processes leading to divertor detachment is incomplete. The paper summarizes the efforts undertaken in gradually advancing the understanding of power exhaust in a variety of conditions: It presents how the H-mode density limit is controlled by a fuelling limit and an enhanced loss of power at the plasma edge. The power fall off length in the divertor is determined by the volumetric dissipation in the divertor connected to the recycling of neutrals and consequently to the divertor geometry. Experiments with nitrogen as seeding impurity for L-mode and H-mode are used for validating the SOLPS code package. In such studies the activation of drift terms in the numerical model is crucial for minimizing the differences to the experimental data. The movement of the radiation in the divertor under varying conditions is exposed and maximum radiation is reached with stable radiation in the vicinity of the X-point. A phase of strong fluctuating radiation in the vicinity of the X-point on the high field side of the divertor is identified as a condition for strongest discrepancy between the numerical and experimental results. Studies on the snowflake as an alternative divertor geometry solution using the EMC3-EIRENE code are also presented.

1 Introduction

Power exhaust is recognized as a crucial issue for future fusion devices such as ITER and DEMO. The thermal power of the confined plasma needs to be exhausted such as to reach power loads below engineering limits for plasma facing components, PFCs. While for the PFCs of the divertor target in ITER a time averaged tolerable power flux density of \(\lesssim 15 \text{MW/m}^2\) is foreseen a device like DEMO would, due to the enhanced limitations in an irradiated environment, have a lower limit of \(5 \text{MW/m}^2\) to \(10 \text{MW/m}^2\). A device like DEMO might be about the same
geometrical size of ITER but will need to accommodate an about 3 to 4 times higher thermal power, aggravating the issue of power exhaust. A further limitation to the conditions in the Scrape-Off Layer, SOL, is given by the required lifetime of the plasma facing components in view of the material erosion both at the divertor target plates and at the main chamber PFCs. For ITER the combination of these constraints requires plasma operation at high densities and radiative power fractions, \( f_{\text{rad}} \), in the range of 60% to 80% (evaluated from [1]). In the case of DEMO the power that needs to be dissipated, \( f_{\text{dis}} \), is estimated to be beyond 90%. Here \( f_{\text{dis}} \) includes \( f_{\text{rad}} \), power transported to the main chamber walls by diffusion and advection as well as power exhausted by elastic and inelastic collisions with neutrals. It is hoped that this level of \( f_{\text{dis}} \) will be possible maintaining the H-mode operational scenario. Due to a similar size of the DEMO and ITER divertor it may be assumed that for DEMO the divertor might be capable of radiating only slightly more power in absolute terms, which would imply much higher \( f_{\text{rad}} \) inside the separatrix than usually encountered in the current operational regimes and that foreseen for ITER. ASDEX Upgrade with its fully tungsten covered wall, high ratio of heating power to major radius (a qualifier for the severity of the power exhaust) and extensive edge and SOL diagnostics is particularly well suited for studying most aspects of power exhaust in view of future devices. Limiting the total power flux to the divertor target plates is only possible by limiting the overall particle flux density. For low \( T_e \) the power load resulting from surface recombination of the impinging ions alone would be equal to the plasma heat flux [2].

A reduced particle flux can only be achieved in the detached regime. The degree of detachment, DoD, is then by definition > 1 [4]. The DoD will potentially be higher in a device like DEMO compared to ITER, which itself will be operating in a partially detached regime. This regime is linked to operation at high line averaged densities which are associated to low \( T_e \) at the divertor target plates, which naturally facilitates the transition into detachment. Also, for minimizing the erosion from charged impurities, such as to enhance the lifetime of the divertor PFCs, \( T_e \) needs to be below \( \sim 2\,\text{eV} \) to \( 3\,\text{eV} \) [5]. Despite its crucial importance for safe operation of future larger devices the understanding of the processes leading to divertor detachment is currently incomplete and numerical predictions therefore unreliable [6, 7]. The achievable high line averaged density, \( \bar{n}_e \), is limited by the H-mode density limit, HDL. Recent studies on ASDEX Upgrade have shown that four distinct reproducible phases can be identified in gas ramp discharges when approaching the HDL. They can be explained by a fuelling limit via an enhanced ionization in the SOL and an enhanced loss of power at the plasma edge as the H-mode degrades [8].

**FIG. 1:** \( S \) normalized by the flux expansion, \( f_x \), versus target \( T_e \) near the separatrix for most of the numerical database including results from a grid based on old divI discharge #7888. An H-mode density scan is added for comparison as well as experimental data from divIId and JET horizontal target for which maximum target \( T_e \) is used instead of \( T_e,\text{tar,sep} \), from [3].
FIG. 2: Time traces of AUG discharge #29172 showing the HFSHD front formation. The density in the HFS far SOL measured with the semi horizontal LOS SBDH1,2 (red) and at the X-point measured with the vertical LOS SBDV1,2 (blue) and the vertical interferometer VIF (green) versus the total applied heating power, top, the neutral fluxes in the inner far SOL, $\Gamma_{HFS}$, in the private flux region, $\Gamma_{PFR}$, and the inner divertor neutral compression ratio $\Gamma_{PFR}/\Gamma_{HFS}$, from [10].

agating filaments in the SOL. In fact in the case of L-mode plasmas at elevated densities larger and faster filaments propagating in the SOL have been observed. These lead to an increase of the width of the SOL density profile. As this enhanced transport coincides with the onset of divertor detachment it was deduced that the resistivity in the divertor may determine the nature of the filamentary transport in the SOL and not the local plasma conditions [9]. However, a full account of where this power is lost to is currently still pending and a fraction of the lost power might even occur in the divertor via CX reaction processes.

2 Divertor physics and power exhaust

Under the presently assumed scaling of the power fall-off length in the SOL, $\lambda_q$, and considering only the magnetic geometry, which includes the flux expansion and the poloidal inclination angle to the divertor target plates, in DEMO a power flux density of up to $80 \text{ MW/m}^2$ would be obtained at the divertor target plate [13]-[15]. Therefore a significant effort is undertaken in order to demonstrate how to maximize the power exhaust in the divertor and potentially extract scaling laws that may allow to extrapolate the findings from current to future devices [11],[16]. The inter-
play of volumetric and plasma surface interaction processes is complex and non-linear. Scaling laws based on experimental data for divertor performance are hardly existent. The power fall-off length along the divertor target plates is a linear combination of $\lambda_q$ and the spreading of power in the divertor, $S$ \textsuperscript{13,17}. The latter is determined by the volumetric dissipation of power in the divertor, which combines perpendicular transport and radiative losses. In comparison with experimental data from low to medium recycling outer target attached L-mode discharges, numerical simulations with the SOLPS code package demonstrate how the dissipation is strongly related to the recycling of neutrons. A clear correlation between the width $S$ and the divertor target temperature could be found in the simulations, see Fig. 1 \textsuperscript{3}. For otherwise fixed $B_{pol}, B_T$ this links $S$ and the recycling levels that can be achieved for open and closed divertor geometries with respect to neutrals, as these themselves impact the value of $T_e$ that can be achieved for given upstream conditions at the target plate. Assuming $T_e$ of 10 eV at the target a value of 1 mm can be expected for $S$ which would lead to a reduction by a factor of 2.6 of the peak heat flux value from 80MW/m\textsuperscript{2} to $\sim$ 30MW/m\textsuperscript{2} in DEMO. Similar L-mode conditions are furthermore used for validating the role of drift terms. Their activation in the code reproduces to a large extent the experimentally observed divertor asymmetries for low and medium $\bar{n}_e$. In the case of the ion grad-B drift directed towards the active divertor drift terms enhance the power load asymmetry between the inner and outer divertor, thereby improving the agreement for cases at low densities. Then a quantitatively satisfactory agreement ($\leq 20\%$) between the modelled and the experimentally measured values can be found in the volume of the inner divertor. Nevertheless, even for the lowest $\bar{n}_e$ no satisfactory agreement can be found along the outer target plate when comparing either the ion flux density or $n_e$, indicating that the physical model...
The numerical models struggle in reproducing the intermediate ne regime in which the plasma along the inner divertor vertical target plate completely detaches and the outer divertor transits into the partially detached regime, with a DoD > 1. While the code package used is able to principally achieve detached outer divertor conditions it underestimates the neutral pressures and ion flux densities experimentally achieved just prior to the detachment [7,20]. This questions if the reasons for which the code numerically triggers the detachment are of the same nature as in the experiment, since experimentally a higher drop (factor 2-10) in ion flux density is observed than numerically. Experimentally this phase is connected to a regime of fluctuating total radiation in the vicinity of the X-point on the high field side, HFS, and is therefore denominated the fluctuating state [21]. The fluctuating state correlates with the appearance of a region of high density that extends into the far SOL of the high field side divertor, which is labelled high field side high density, HFSHD. This regime is present in L-mode and in H-mode plasmas [10,22,23]. Furthermore, the disappearance of the radiative fluctuations correlated with the reduction of the extension of the HFSHD appears to coincide with the transition of the outer divertor into the detached state [21,23]. Enhancing the spectroscopy as well as the fast AXUV type bolometers in the divertor and X-point region has helped to characterize this regime [24,25]. With metal PFCs carbon is to a much lesser extent available as an intrinsic impurity for cooling the divertor. In ITER and DEMO extrinsically seeded impurities will be used to reach radiation levels for reducing Te in the divertor, such as to induce divertor detachment. The extension of the HFSHD region increases toward the HFS SOL with increasing heating power and decreases with N seeding, see Fig. 2. The physics mechanism of the development of this regime is unclear. Numerical simulations have not been able to reproduce this regime under unseeded L-mode or H-mode conditions [6,18]. Assuming enhanced perpendicular transport localized to the surroundings of the X-point in the HFS region of the SOL appears to provide a progress in understanding the appearances of the HFSHD region. Nevertheless this remains insufficient to fully explain the experimentally observed phenomenon and its relation to the power flux available in the HFS region [26]. It is suspected that this regime influences the fueling efficiency via the recycling neutrals and maybe connected to the increase of core plasma density when magnetic perturbation coils are activated, that generate lobes which extend into the HFSHD [21,27,28]. As seeding of light impurities is required for power exhaust an understanding of the relevance of the individual physics mechanisms involved and the ability to extrapolate these to future devices is crucial. In order to validate the code package SOLPS under nitrogen seeded conditions experiments were undertaken in L-mode and in H-mode [12,18]. The location of the impurity radiation depends on the initial plasma conditions as well as the seeding levels and the numerical model is able to explain the movement of the radiation in the divertor volume. As for unseeded conditions the activation of drift terms in the model minimizes the differences to the experimental data. For the low recycling outer target regime even under seeded conditions a satisfactory quantitative agreement is obtained. The discrepancy increases with enhanced seeding level, reduced temperature and increased recycling flux. However, the quantitative differences appear to be of a lower factor with N seeding than in the unseeded high recycling cases for L-mode plasmas [7]. As an initial attempt to consolidate a scaling law for the divertor radiation in the case of nitrogen seeded H-mode plasmas, which was presented in [11], the numerical results of the well validated L-mode plasmas are compared to this scaling law, see Fig 3. While the absolute value of the radiation
needs to be multiplied by a factor of 2 in the model, the scaling is well correlated with the neutral density in the divertor \cite{12}. Such efforts may help to provide a physics based scaling model for the divertor radiation that can be implemented into system codes used for designing the operational parameters of DEMO.

3 Highly radiating detached scenarios

Figure 3 shows that for simulated radiation levels beyond 60% the numerical results deviate above the scaling, with the scaling being based on the experimentally determined total radiation below a horizontal line above the X-point. At these levels pronounced radiation at the X-point and inside the separatrix develops. In H-mode numerically and experimentally the maximum overall radiation is reached when the discharge conditions allow for stable radiation in the vicinity of the X-point inside the separatrix. It is experimentally observed that this coincides with pronounced and complete detachment, defined as a reduction of the power and particle flux to the target over several power decay lengths with respect to attached conditions. Furthermore, the X-point radiation appears to be characterized by a region of very low $T_e \ (< 10eV)$ \cite{18, 23}. In H-mode this regime is connected to a mitigation of ELMs, a reduction of the pedestal pressure profile and an at least transient increase of the central plasma density \cite{16, 18}. Due to its widespread low divertor $T_e$ and low $\Gamma$ along the target combined with a high $f_{\text{rad}}$ beyond 80% this regime appears to be particularly attractive as a detached scenario for DEMO. The coincidence of the appearance of the strong X-point radiation and the pronounced detachment raises the question if fundamentally complete detachment in H-mode is only achievable under such conditions. It would indicate that any DoD beyond pronounced detachment is only achievable if the upstream pressure itself is reduced. Numerical simulations have been undertaken to simulate the completely detached H-mode regime. It is numerically possible, with drift terms activated, to achieve a stable radiating X-point with an overall $f_{\text{rad}}$ similar to the experimental values, see Fig. 4. Best consistency of modeling with experiment in all available diagnostics is achieved with radiation inside the confined plasma. With this radiation an even larger pressure loss at the target can be achieved in the simulations. Numerically the complete detachment is achieved if strong X-point radiation occurs, that induces a reduction of the upstream pressure combined with an assumption of enhanced perpendicular transport in the outer divertor. Experimentally a pressure loss is qualitatively observed upstream that has not been able to be quantified.

4 The snowflake geometry for risk mitigation and the effect of non-axissymmetric structures in tokamaks

It is yet undetermined if high dissipative regimes with $f_{\text{dis}}$ beyond 90% can be experimentally achieved and extrapolated to future large scale devices. Despite that considerable progress has been made in applying the numerical tools for interpreting experimental data under highly radiating and detached conditions they remain unreliable for predictions inside the required margins of confidence. A key question is if a radiation level inside the area of closed flux surfaces can be optimized for the overall performance of the device. One needs to retain sufficient power
for remaining above the threshold for the L to H mode transition. At the same time the amount of power crossing the separatrix, $P_{\text{SOL}}$, needs to be lower than the amount that can be dissipated in the SOL and divertor such as to remain below the engineering limits for the divertor target plates. Therefore an effort for evaluating alternative risk mitigation schemes is underway. Since EMC3-Eirene does not require a flux surface aligned computational grid, it is possible to implement complicated topologies in a simulation, see Fig. 5. This was the first code that simulated the TCV snowflake divertor including the structures around the secondary X-point [19]. Compared to the simulations the measured power flux is by an order of magnitude larger at the secondary strike point, being of the order of 10% of that at the primary strike point. This indicates an enhanced transport across the X-point. This enhancement is possibly driven by beta-poloidal instabilities [29]. As EMC3-EIRENE is a 3D code it is furthermore increasingly applied to model the SOL and divertor for standard single null divertor geometries of ASDEX Upgrade under the influence of magnetic perturbation coils, which break the toroidal symmetry of the magnetic field and introduce a 3D magnetic topology. Furthermore, it has also been applied to model the far SOL at the non-axissymmetric structure of the PFCs of ASDEX Upgrade, where the numerical grid has been successfully extended to the first wall incorporating limiters and recessed areas [30]. The inclusion of 3D effects and the comparison to advanced divertor geometries will allow for developing a better physics model and consequently increase the level of confidence when extrapolating to future devices.

5 Conclusions and Discussion

Despite the advances made in the numerical modelling of experimental data, power exhaust remains a critical issue for future fusion devices. Efforts are undertaken in modelling L-mode as well as H-mode plasmas up to high values of $f_{\text{dis}}$. The activation of drift terms is required to consolidate numerical modelling with experimental data at low densities as well as for varying radiation levels in the case of impurity seeding. An experimentally observed region of high density in the far SOL of the high field side, observed in an experimental regime, coinciding with a fluctuating radiative regime close to the X-point, is not recovered numerically and presents the regime with the strongest discrepancies between model and experiment. A regime of complete detachment, attractive for a reactor, appears to be linked to a strongly radiating X-point. Numerical simulations indicate that flat divertor profiles, characteristic of complete detachment, may only be achievable if already the upstream plasma pressure can be reduced. In this regime the plasma stored energy is largely retained further inside the core plasma. Configurations for a potential risk mitigation are studied numerically in order to understand the underlying physics of the redistribution of power in snowflake configurations, which might support a better understanding of transport processes in the vicinity of the X-point and the divertor volume in general.

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References

[27] E. Wolfrum et al. 40th EPS Conference on Controlled Fusion and Plasma Physics.