Journal of Nuclear Materials xxx (2015) xxx–xxx

Contents lists available at ScienceDirect

Journal of Nuclear Materials

journal homepage: www.elsevier.com/locate/jnucmat

High density operation for reactor-relevant power exhaust

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ARTICLE INFO

Article history:
Available online xxxx

ABSTRACT

With increasing size of a tokamak device and associated fusion power gain an increasing power flux density towards the divertor needs to be handled. A solution for handling this power flux is crucial for a safe and economic operation. Using purely geometric arguments in an ITER-like divertor this power flux can be reduced by approximately a factor 100. Based on a conservative extrapolation of current technology for an integrated engineering approach to remove power deposited on plasma facing components a further reduction of the power flux density via volumetric processes in the plasma by up to a factor of 50 is required. Our current ability to interpret existing power exhaust scenarios using numerical transport codes is analyzed and an operational scenario as a potential solution for ITER like divertors under high density and highly radiating reactor-relevant conditions is presented. Alternative concepts for risk mitigation as well as strategies for moving forward are outlined.

1. Introduction

A reactor such as DEMO will be operating at considerably higher total heating power and power flux density in the Scrape-Off Layer, SOL, compared to existing devices or even compared to ITER [1]. The heating power from additional heating power sources as well as from alpha particle heating needs to be exhausted. In order to optimize the lifetime of plasma facing components, PFCs, a reduction of the impinging power flux to tolerable values is mandatory [2]. The mitigation effort on the power flux required for safe operation can be estimated by comparing the expected power flux without volumetric dissipation onto the target plates and its gap to the heat load tolerable by plasma facing components, PFCs. Based on energy confinement, in order to achieve a sufficient fusion gain in a reactor, the size of the machine, i.e. the major radius, R, needs to be increased. The fusion power loss to the plasma scales $\propto R^3$. For otherwise fixed toroidal and poloidal magnetic fields the area on which the power is deposited scales $\propto R \times \lambda_q^{int}$, with λ_q^{int} being the integral power fall off length along the target [3,4]. This λ_q^{int} is a linear combination of the power fall off length upstream, λ_q, and the spreading of power in the divertor S [5,6]. The latter is determined by the volumetric dissipation of power in a single null divertor, which combines perpendicular transport and radiative losses. These quantities are defined at the upstream location of the SOL. We define 'upstream' as a location along a flux surface above the entry into the divertor, therefore above the X-point and usually taken at the outer midplane. Scalings for λ_q and S have been derived in H-mode and in L-mode plasmas [5,7–9]. These scale λ_q independently of R. They have been derived for low density and attached plasma conditions assuming negligible volumetric power losses along a flux tube. In theory λ_q could be larger due to additional effects that may not be accounted for yet at high densities [10]. With λ_q being independent of R a measure for the severity of the heat exhaust problem is the ratio of the total plasma heating power, P_{heat}, and the major radius of the device [11,12]. Assuming these scalings of λ_q and no radiative losses in the core plasma one obtains the values listed in Table 1 for the parallel power flux, $q_{\parallel}$. We now base our considerations purely on geometry and assume a single null ITER like divertor geometry and thus flux expansion. We assume equal power sharing between the inner and outer divertor volumes, no volumetric power loss, a target inclination angle between 1° and 3° and no reduction of the plasma wetted area as a consequence of tile tilting and alignment, this leads to a power flux to the target, q_{target}, given in the fourth column of Table 1. Including an available scaling for S leads to the values reported in the last column of Table 1.

Based on existing technology the PFCs of a future fusion reactor will need to be composed of a high Z material such as W in the divertor and potentially also in the main chamber. Tungsten is the most likely choice due to its low erosion yield and its low fuel retention [13,14]. The implications for tokamak operation with W as a PFC have been successfully demonstrated on ASDEX Upgrade and JET over the past years. As a consequence it is the choice for

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[1] See the Appendix of F. Romanelli et al., Proceedings of the 24th IAEA Fusion Energy Conference 2012, San Diego, US.

http://dx.doi.org/10.1016/j.jnucmat.2014.12.078
0022-3115/

the armour material for the ITER divertor [15–18]. For steady state operation in a fusion reactor not only the choice of the plasma facing material has to be considered but also the entire actively cooled component behind the material that removes the power impinging onto it. This consists of an integral approach combining the coolant (e.g. Water or Helium), the structural material of the coolant pipe (e.g. CuCrZr alloy) as well as the armour material. In a reactor a failure of this combined system needs to be prevented in the expected high neutron irradiation environment leading to a foreseen limit of the steady state tolerable power flux of 5–10 MW/m^2, which is lower than the technological limit for ITER where a 30 times lower neutron irradiation is expected [19–21].

This technological limit restricts the tolerable particle flux onto the divertor target plates. The total deposited power is a combination of heat transferred across the sheath and power released by surface recombination processes. Each ion–electron pair will deposit at least 13.6 eV and not more than 18.1 eV potential energy [22,23]. The range depends on how much of the molecular recombination energy is released to the PFCs and the level of hydrogenic saturation of the PFC. Assuming $T_e \leqslant 2.5$ eV, $T_e = T_i$, the heat flux crossing the sheath is of the order of the power deposited on the surface by surface recombination. Together with an estimated power load from radiation of 2 MW/m^2, similar to ITER [24], then a conservative assumption of a maximum tolerable power flux of 5 MW/m^2 implies a limit of the particle flux, Γ, to values below 5×10^{23} m^{-2}s^{-1}. In ITER the material limit of 10–15 MW/m^2 implies a radiative dissipation, f_{rad}, of approx. 60–75% of the total loss power of 150 MW. This estimate is derived from summing the 60–70% of the power entering the SOL, P_{SOL}, of ~100 to 120 MW that needs to be radiated and 30 MW of radiation that is expected inside the separatrix on closed field lines [25]. A DEMO device is likely of similar size as ITER and thus the power dissipation capability of the divertor will be of similar quality [26]. Under these assumptions a DEMO type device has to dissipate, f_{dis}, $\geqslant 95\%$ of the total loss power. The dissipation of power accounted for in f_{diss} includes all those processes that lead to a reduction of power reaching the divertor target plates. Therefore the process of dissipation and thus f_{diss} includes radiation, f_{rad}, Charge-Exchange processes as well as transport to the main chamber walls, e.g. by processes of filamentary and diffusive nature. Moreover this assumption of a similar and maybe slightly higher dissipative capability of the SOL and divertor in DEMO as in ITER implies that more than 70% of the loss power will need to be radiated inside the confined plasma. Therefore in DEMO the majority of the volumetric dissipation would not occur in the divertor or the SOL. The radiation inside the closed flux surfaces needs to be limited to a narrow band between the separatrix and the pedestal top region, such as to minimize the impact on core performance. If instead of no dissipation as in Table 1 an f_{rad} of 70% is considered in the core plasma of DEMO, then a peak heat flux of 80 MW/m^2 would be obtained in the third column of Table 1 instead of 300 MW/m^2. Assuming now, as done already in the last column of Table 1 for the case with no f_{diss}, T$_e$ of 10 eV in a high recycling regime at the target, a value of 1 mm can be expected for S which would lead to a reduction by 2.6 of this initial heat flux value to ≈30 MW/m^2 [27].

While the source of power that is exhausted is well localized inside the LCFS, the analysis of the mechanisms relevant for the combined particle and power exhaust is complex as the poloidal distribution of the ion fluxes impinging onto PFCs and the associated recycling fluxes are insufficiently known. This information is not easily accessible experimentally and no first principle based theory exits that can predict perpendicular transport in the SOL [28]. In order to account for an as complete as possible physics model for describing the complex non linear processes involved in power and particle exhaust numerical tools have been developed. The back-bone of these tools are fluid transport codes solving the modified Braginskii equations which are coupled to Monte Carlo neutrals transport codes (EIRENE [29,30], NEUT2D [31], DEGAS [32]). They do not contain a physics based perpendicular transport model and transport is adjusted by undertaking a fitting procedure to available experimental profiles or an ad hoc assumption based on experience for predictive modeling. Kinetic effects are only partially accounted for (e.g. global parallel heat flux limit, boundary conditions at PFCs). Sputtering of impurities only occurs at the divertor target plates in a realistic geometry and an extension of the employed grids to the main chamber wall is underway [33,34]. With the EMC3-EIRENE code a first comparison to experimental data has been made, which includes plasma wall interaction at the main chamber wall and related 3D effects [35]. The numerical code packages in use are UEDGE [36], SOLPS4.x (B2) [37], SOLPS5.x (B2.5) [38], EDGE2D [39,40], SONIC [41] with SOLPS-ITER [42] currently being released for 2D problems as well as EMC3-EIRENE [43] for 3D and 2D physics.

2. Understanding power dissipation in the divertor

The limitation on the particle flux as a consequence of the power handling capability of the divertor component requires operation in the detached regime. Detachment is defined as a reduction of the ion flux to the divertor target compared to what is expected from the two point model [44]. In this context the degree of detachment is used in an attempt to quantify the "strength" of the detachment [45]. A useful guide in analyzing the dependencies of the divertor target parameters on upstream conditions and volumetric losses in the SOL is the so called 'corrected two point model' [46]. It states that $\Gamma \propto \frac{f_{mom}^2 f_{cond}^{4/7}}{1-f_{pow}}$. These are the loss factors for momentum, f_{mom}, power, f_{power}, and conducted heat transport, f_{cond}. In order to define the design specifications and the operational point of a device such as DEMO it will be important to derive scalings of the loss factors contained in this model and their related divertor plasma parameters as e.g. done in [47]. Following the corrected two point model the prerequisite for detachment is ought to be a loss of pressure in the SOL. Pressure can be lost along a field line by perpendicular transport processes such as convective or diffusive transport whose dependency on T_e is not known. Charge exchange reaction losses and volume recombination can act at low T_e (most effective for $T_e < 5$ eV). Fig. 1 shows the efficiency for the loss of pressure for low T_e. Maximizing the loss of pressure is only possible if a sufficiently large volume with

Table 1
Values for P/R, parallel power flux, $q_\parallel$, at an upstream location in the SOL and power flux, $q_\perp$, onto divertor target. A power fall-off length λ_q of 1 mm is assumed. The fourth column only considers an ITER-like divertor geometry, no volumetric dissipation, no dissipation in the divertor and includes only geometric considerations such as the poloidal impact angle of a field line (1–3°) as well as a flux expansion, f_x, of 5. Assuming T_e of 10 eV and not considering potential dependencies on the poloidal field, a value for S/f_x of 1 mm can be assumed [27], leading to a λ_{int} of 2.6 mm.

Device	P_{heat}/R (MW/m)	Upstream $q_\parallel$ to each divertor (GW/m^2)	Unmitigated $q_\perp$ (MW/m^2)	$\approx q_\perp$ assuming $\lambda_{int} = 2.6$ mm (MW/m^2)
JET	7–12	2	20	8
ASDEX Upgrade	14	3.5	35	13
ITER	20	5	50	20
DEMO	80–100	$\geqslant$30	300	115