Edge Plasma Physics Issues for the Fusion Advanced Studies Torus (FAST) in Reactor Relevant Conditions

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Abstract. The issue of "First wall materials & compatibility with ITER /DEMO relevant plasmas" is among the R&D missions for possible new European plasma fusion devices that the FAST project will address. FAST can operate with ITER relevant values of P/R (up to 22 MW/m, against the ITER 24 MW/m, inclusive of the α particles power), thanks to its compactness; thus it can investigate the physics of large heat loads on divertor plates. The FAST divertor will be made of bulk W tiles, for basic operations, but also fully toroidal divertor targets made of liquid lithium (L-Li) are foreseen. To have reliable predictions of the thermal loads on the divertor plates and of the core plasma purity a number of numerical self-consistent simulations have been made for the H-mode and steady-state scenario by using the code COREDIV. This code, already validated in the past on experimental data (namely JET, FTU, Textor), is able to describe self-consistently the core and edge plasma in a tokamak device by imposing the continuity of energy and particle fluxes and of particle densities and temperatures at the separatrix. In the present work the results of such calculations will be illustrated, including heat loads on the divertor. The overall picture shows that, marginally in the intermediate and, necessarily in the high density H-mode scenarios ($\langle n_e \rangle$=2 and $5 \cdot 10^{20}$ m^{-3}respectively), impurity seeding should be foreseen with W as target material: however, only a small amount of Ar (0.03% atomic concentration), not affecting the core purity, is sufficient to maintain the divertor peak loads below 18 MW/m^2, that represents the safety limit for the W monoblock technology, presently accepted for the ITER divertor tiles. Li always needs additional impurities for decreasing divertor heat loads, the Z_{eff} value being ≤ than 1.8. At low plasma densities (but ≥ $1.3 \cdot 10^{20}$ m^{-3}), typical of steady state regimes, W by alone is effective in dissipating the input power by radiative losses, without excessive core contamination. Impurity seeding would lead to excessive W sputtering by Ar and too high Z_{eff}. The impact of the ELMs on the divertor in the case of a good H-mode with low pedestal dimensionless collisionality will be discussed too.

1. Introduction

In a tokamak reactor the compatibility of the burning plasma with the plasma facing components (PFCs) is of outmost importance. Transient loads, as Edge Localised Modes (ELMs), or contact with too harsh plasma could cause irreversible damage and/or deep erosion well before the established device lifetime. The released impurities must be minimized and retained efficiently within the scrape-off layer (SOL) in order not to pollute the main plasma. In addition, PFCs should bear very large neutron fluxes, without significant degradation over the reactor lifetime, and retain tritium so poorly to not exceed the allowed inventory.

The ITER proposed solutions, wholly Be chamber walls and W divertor tiles (plus Carbon Fibre Composites (CFC) targets in the H_2 initial phase), are compatible with the demands outlined above, provided the SOL and edge plasmas meet quite precise constraints: ELMs energy close to 1 MJ, and SOL plasma quasi-detached from the divertor plates, i.e. with electron density and temperature $n_{e,plate} \sim 1 \cdot 10^{21}$ m^{-3}, $T_{e,plate}$<5 eV. The full and reliable compatibility of both these requests with highly performing plasma still raises some concerns. The DEMO main plasma will suffer even more stringent constraints since it should be more performing, while the PFC materials presently available will pose the same restrictions.

The Fusion Advanced Studies Torus (FAST) project [1], besides contributing to understand

the α particle behaviours in burning plasmas by using fast ions and exploiting advanced tokamak (AT) regimes with long pulse duration, will test technical solutions for the first wall/divertor directly relevant for ITER and DEMO. FAST will have an additional power up to 40 MW and its figure of merit P/R (P=input power, R=major radius), which establishes the reactor relevance of an edge/SOL plasma, is ~22 MW/m (R_{FAST}=1.82 m) against the ITER 24 MW/m, inclusive of the α particles power. FAST will be a full W machine. The divertor will be made of bulk W tiles, for basic operations, but also fully toroidal divertor targets made of liquid lithium (L-Li) are foreseen. Viability tests of such a solution for DEMO divertor will be carried out as final step of an extended programme started on Frascati Tokamak Upgrade (FTU) tokamak by using a liquid lithium (L-Li) limiter [2]. If these tests will be positive, they would open the way to solve the problems of permanent damages, as cracks and erosion of divertor targets, due to the possibility of external refilling of L-Li.
The present work intends to illustrate the results of the modelling activity of the FAST SOL/edge plasma, including heat loads on the divertor. A forecast of the impact of the ELMs on divertor for a good H-mode scenario with low pedestal collisionality is also reported.

2. Computing Tools

In order to predict the edge plasma parameters in the FAST device under different operating scenarios two different approaches have been used. First the general plasma performance in the FAST tokamak has been analyzed in the frame of a relatively simple self-consistent description of the core and edge plasma. Secondly, the simulations of the plasma parameters in the scrape-off layer and divertor region have been done with the help of a more sophisticated 2D multi-fluid model of the boundary plasma implemented in the COREDIV code [3, 4].
In the simple model the edge plasma density is prescribed at the stagnation point (upstream plasma density n_{sep}) and used as input parameter. The upstream temperature and the plate density and temperature are evaluated using the usual 2-point model [5].
The COREDIV code treats the coupled SOL-bulk system by imposing the continuity of energy and particle fluxes and of particle densities and temperatures at the separatrix. The code solves self-consistently radial 1D energy and particle transport of plasma and impurities in the core region and 2D multi-fluid transport in the SOL. This requires that the core and edge part of COREDIV are iterated alternatively until steady state is achieved. Neoclassical transport is considered in the bulk, with a contribution from anomalous transport which is scaled to reproduce the experimental energy confinement scaling law ($H98(y;\ 2)$). It is assumed that all ions have the same temperature. The 2D multi-fluid model in the SOL and divertor region is based on Braginskij-like equations for the background plasma and rate equations for the ionization state of each impurity species. The impurities at the divertor targets, made either of W or Li, are produced only via sputtering + self-sputtering. Impurity seeding by either Ar or Ne is considered as an option to mitigate the thermal loads onto the divertor tiles by increasing the total radiation losses. The source (gas puff) of the injected impurities (Ar, Ne) is assumed to be located in the divertor. The intrinsic impurity release caused by impact of the seeded impurities on the plate is accounted for. The COREDIV code has been recently applied to simulate nitrogen and neon seeded JET discharges and good agreement with the experimental data has been found [6]. The divertor is assumed to be in the attached mode and the hydrogen recycling coefficient is an external parameter. The energy losses due to interactions with hydrogenic atoms (line radiation, ionization and charge exchange) are accounted for in the model.
A simple slab geometry (poloidal and radial directions) with classical parallel transport and anomalous radial transport is used for the SOL and the impurity fluxes and radiation losses

caused by intrinsic and seeded impurity ions are calculated fully self consistently. The main limitations of our modelling refer to the lack of pedestals and to the absence of the impurity anomalous pinch. The presence of a temperature pedestal may lead to a broadening of the radiation zone inside the separatrix. The anomalous pinch velocity can affect significantly the high Z impurity density profile.

3. Divertor Heat Loads

The divertor heat load is calculated by assuming attached plasma and a well closed geometric configuration. The latter assumption results in the benefit of flux expansion Φ being reduced. The heat diffusion in the private region has been neglected. The ratio between the power flowing to the outer and the inner divertor is assumed = 2.

One of the key plasma parameters in defining the heat load is the power flux e-folding length at the outer midplane λ_p^{omp}, the heat flux profile at the target being usually mapped from the outer midplane by using $\lambda_p^{target} = \lambda_p^{omp}\,\Phi$, with Φ evaluated by equilibrium reconstruction to be equal to 5. Many scaling laws for the divertor power deposition width can be found in literature, often differing largely from one another, especially in the dependence on the power to the divertor (both positive and negative dependences are reported) [7]. By assuming the separatrix ion collisionality $\nu_{isep}^* \equiv L_{\|}/\lambda_{ii}$, where $L_{\|}$ is the connection length and λ_{ii} is the ion-ion collisional mean free path, as the governing parameter, we applied to FAST H-mode scenarios the following scaling, obtained for JET H-mode discharges by regression analysis on experimental power deposition profiles measured by the swept strike-point thermocouple technique (formula 1 of ref [8]):

$$\lambda^{TC}_{q\,\text{H-mode}} \propto A(Z)^{1.1}\, B_{\varphi}^{-0.9}\, q_{95}^{\,0.4}\, P_t^{-0.5}\, n_{e,u}^{\,0.15} \qquad (1)$$

with A and Z the ion mass and charge, B_{φ} the toroidal field, q_{95} the safety factor, P_t the outer target power and $n_{e,u}$ the upstream density on the separatrix. This scaling is best reproduced by the collisionality modified ion orbit loss theory, which links the radial power flux to the classical ion orbit loss and takes into account collisions in the SOL. Following ref [8] we introduced in (1) the dependence on machine dimension and extrapolated the FAST λ_p^{target} from a low collisionality JET discharge. For both reference and extreme H-mode scenario [10], exhibiting ν_{isep}^*~1, we got a very short λ_p^{target}, of the order of $1\cdot10^{-3}$ m. On the other hand the application at the FAST H-mode scenario of a multi-machine scaling [11] provides a value λ_p^{target} ~ $15\cdot10^{-3}$ m or ~ $6.5\cdot10^{-3}$ m depending on the scaling being calculated with the measured power flux to the outer divertor or with the total input power. These much longer λ_p^{target} values mainly result from their positive, instead of negative (like in (1)), dependence on power. In any case, the closed geometric configuration envisaged for the FAST divertor [10] is expected to result in the broadening of the heat flux profile at the target with respect to the one at the entrance of the divertor, because of the plasma-neutral interaction, larger than in a rather open divertor configuration, like the JET one. Following the above considerations and taking into account that the dependence on separatrix density used in (1) was inferred from experimental values well lower than the ones foreseen in FAST, a conservative λ_p^{omp} = 5 mm was assumed for FAST H-mode scenarios. The average heat flux on the divertor has been calculated as q_{target} [$MW\cdot m^{-2}$] = $f_{out}P_{div}\cos\theta_p/(2\pi R_{out}\lambda_p^{target})$, where f_{out} is the fraction of P_{div} flowing to the outer target (= 2/3), θ_p is the tilt angle of the target in the poloidal cross section, assumed = 70° and R_{out} = 1.6 m is the major radius of outer target. The resulting heat loads refer to non-shaped, toroidally flat, target tiles. Including tile shaping in the toroidal direction, for avoiding possible edge overheating, should reduce the effective divertor surface and therefore increase the heat load. The reduction of the effective surface depends mainly on the magnetic field line inclination angle $\theta_\perp$ and the smoothing angle, the latter in turn depending on the toroidal length of the tiles.

The divertor heat loads estimated with this modelling are to be compared with the safety limit of 18 $MW \cdot m^{-2}$ tested for monoblock W tiles constructed according a recently developed technique [12], presently accepted for the ITER divertor tiles.

4. Results and Discussion

The FAST edge plasma has been modelled for the six reference scenarios, as defined in [1,10]. In this paper the results for the three most representative scenarios are reported. They are the reference H-mode, aimed to optimize performance, the extreme H-mode, aimed to test scenarios with Q=2.5 and the full Non Inductive Current Drive (NICD) regime. The values of plasma current, toroidal magnetic field, volume averaged density, peak electron temperature and additional power are reported in the first four rows of table I. Several numerical self-consistent simulations have been made for these scenarios. First, the calculations have been carried out for a full W machine, then the option of injecting Ar and Ne to mitigate the thermal loads has been considered and finally the case, relevant for DEMO, with liquid lithium as divertor target has been analysed.

In order to sketch the general behaviour, a scan of $n_{e,sep}$ for the three simulated scenarios, without impurity seeding, has been made with the simple self-consistent model, 0D in the core and two-points in the SOL. Results for radiative losses, given as the ratio $f_{rad}=P_{rad}/P_{tot}$, power to the divertor P_{div} (MW), Z_{eff} (average plasma ion charge) in the core and plasma temperature T_{eplate} at the plate are shown in fig. 1. The most probable value of $n_{e,sep}$ for each scenario can be assumed to be about 1/3 of the volume averaged plasma density and is evidenced in the plots by an arrow. At the typical high $n_{e,sep}$ ~$1.7 \cdot 10^{20}$ m^{-3} of the H-mode extreme scenario, f_{rad} is low, close to 20%. Even though W radiates efficiently, its production is negligible at the associated low SOL temperatures at the plate ($T_{e,plate}$~30 eV), and moreover the SOL retention is quite effective. As a result almost all the input power goes to the SOL. In this case the coupling between core and edge plasma is almost negligible and the simple model is accurate enough. On the contrary, at the low $n_{e,sep}$, ~$0.3 \cdot 10^{20}$ m^{-3}, typical of the full NICD scenario, the rate of W sputtering by the hot SOL plasma ($T_{e,plate}$~100 eV), associated to low density, is high enough to affect the whole power balance. The tungsten radiation increases remarkably f_{rad} (>40%) and the power flowing to the divertor plate is accordingly reduced. This strong edge-core coupling, however, suggests that in this case the simple model may be quite inaccurate. At intermediate $n_{e,sep}$ ~$0.7 \cdot 10^{20}$ m^{-3}, typical of the H-mode reference scenario, a significant fraction of the heating power is still delivered to the divertor plates, the plasma temperature at the plate ($T_{e,plate}$~70 eV) being not high enough for a tungsten impurity production able to increase f_{rad} adequately.

Successively the COREDIV code was applied to the three chosen plasma scenarios for not seeded plasma with W divertor. In Table I the quantities $n_{e,sep}$, f_{rad}, P_{div}, Z_{eff} and T_{plate} (the last being an average between plasma electron and ion temperature at the plate), as calculated by COREDIV code, are reported (in the same table cases with impurity seeding are shown too and they will be discussed later). In the last row the outer divertor heat load, as calculated according to the assumptions made in section 3, is reported too. The COREDIV results for the H-mode reference and extreme scenarios confirm the trend already observed with the simple model. In the H-mode extreme scenario the core Z_{eff} is close to 1 but the divertor heat load largely exceeds the safety limit. Also for H-mode reference scenario Z_{eff} remains very low and moreover the divertor heat load overcomes safety limit only marginally, suggesting that operation with full W machine and pure deuterium plasma could be possible even at the maximum foreseen input power. For full NICD scenario with $\langle n_e \rangle$ ~ $1 \cdot 10^{20}$ m^{-3} instead, the

W impurity contamination is rather large and the Z_{eff} is calculated to be 2.4, the divertor heat load not being a problem because of the large radiated fraction. In this case, increasing the density to $1.3 \cdot 10^{20}$ m^{-3} is sufficient to attain the acceptable Z_{eff} and q_{target} value of 1.7 and 11 $MW \cdot m^{-2}$, respectively (see Table I). Besides, the overall current drive efficiency is little affected, because the decrease due to the higher density and the increase due the lower Z_{eff} almost balance.

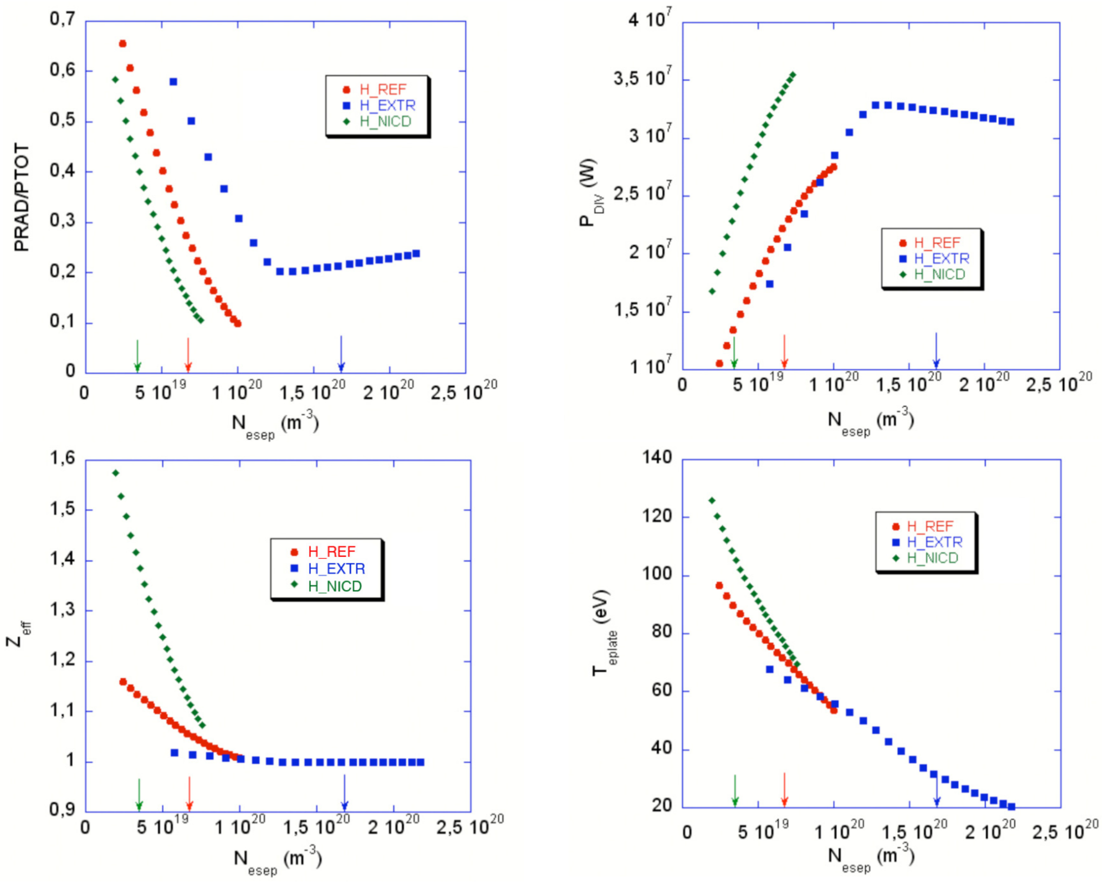

FIG. 1. Radiated power fraction $f_{rad}=P_{rad}/P_{tot}$, power to the divertor P_{div} (MW), Z_{eff} in the core and plasma temperature T_{plate} at the plate for H-mode reference(circles),extreme(squares) and full NICD (diamonds) scenario, without impurity seeding, calculated by simple model as a function of separatrix density $n_{e,sep}$

Because of the severe heat loads the divertor plate could suffer, especially in the H-mode extreme scenario at the largest foreseen input power, the injection by gas puff of additional impurity (Ar) was considered. In this case simulations were carried out with the COREDIV code only, the simple model being not reliable even for intermediate densities, because the tungsten sputtering by noble gases is not accounted for. In the case of high density H-mode extreme scenario it is relatively easy with additional impurity to radiate a large fraction of the input power, while keeping simultaneously low core Z_{eff}. With only a resultant Ar concentration of 0.02% in the core, f_{rad} as high as ~45 % can be obtained at $n_{e,sep} = 1.7 \cdot 10^{20}$ m^{-3}, the core Z_{eff} being 1.1. In the case of H-mode reference scenario, with intermediate $n_{e,sep}$ value, the situation is similar: impurity seeding still helps in pushing the heat load on divertor below the safety limit of 18 $MW \cdot m^{-2}$, by increasing the radiated power fraction from about 20% to about 60%. Z_{eff} values larger than for extreme scenario are attained, but they are always within allowed limits. On the contrary, for full NICD to add impurities for reducing the W influx by cooling the SOL plasma and decreasing sputtering by deuterium, has the opposite effect to raise it because the W sputtering yield by Ar (with a heavier mass) has a threshold temperature lower than by deuterium. The fraction of radiated power is 90%. The role of the W is always dominant respect to the extrinsic impurities such as Ar (or Ne), with much lower radiative loss rate. The divertor heat load is strongly reduced but the core Z_{eff} grows up to unacceptable values. Unlike non-seeded case, to operate at density a little higher ($1.3{\cdot}10^{20}$ m^{-3}) is not sufficient to maintain the Z_{eff} value~~s~~ within acceptable limits. Finally, simulations with COREDIV code have been made for liquid Li as divertor target material with added noble gas. The very low radiative capability of Li implies to seed an impurity in all scenarios to keep the heat load at an acceptable level. The seeded impurity properties always govern the plasma radiation. The results for the H-mode reference scenario, with Ne seeding at 0.7 % of atomic concentration, are reported in the rightmost column of Table I. In this case Ne is used instead of Ar, since the latter results in a higher $Z_{eff,}$ under the same injected quantity. Li evaporation is neglected in the calculations since it is assumed that the Li modules are actively cooled and maintained at a fixed temperature. At the intermediate density of the H-mode reference scenario the radiated power fraction is lower than with the tungsten divertor with added Ar and is in the range of 30%, corresponding to a maximum heat load of 15.5 $MW{\cdot}m^{-2}$. Even though this is well below the limit recently achieved [13] in several test samples (20 $MW{\cdot}m^{-2}$), the test of the liquid divertor response to such a high load is one of the objectives of the experiment. In this case the Z_{eff} value is about 1.8.

TABLE I : COREDIV RESULTS

PFC material	**W**				**W +Ar**			**L-Li+Ne**
Scenario	**H-mode reference**	**H-mode extreme**	**Full NICD**		**H-mode reference**	**H-mode extreme**	**Full NICD**	**H-mode reference**
I_p (MA)	6.5	8	2		6.5	8	2	6.5
B_T (T)	7.5	8.5	3.5		7.5	8.5	3.5	7.5
$\langle n_e \rangle$ ($10^{20}m^{-3}$)	2	5	1.0	1.3	2	5	1.0	2
T_0 (keV)	13.0	9.0	7.5		13.0	9.0	7.5	13.0
P_{ADD} (MW)	30	40	40		30	40	40	30
COREDIV output								
Ar (Ne) (%)	–	–	–		0.03	0.02	0.6	0.68
$n_{e,sep}$($10^{20}m^{-3}$)	0.73	1.67	0.29	0.40	0.76	1.75	0.32	0.92
Z_{eff}	1.1	1.0	2.4	1.7	1.4	1.1	3.6	1.8
f_{rad} (%)	19	21	72	64	57	46	92	31
T_{plate} (eV)	57	32	86	76	17	6	10	34
P_{DIV} (MW)	22.7	32.5	9.2	12.2	11.7	17.8	2.2	17.1
q_{target} (MWm^{-2})	20.6	29.5	8.3	11.1	10.6	16.1	2.0	15.5

5. ELMs

ELMs are one of the major “concerns” intrinsically connected with the “standard” H Mode scenario. Since the FAST reference scenarios rely on a good quality (H_{98}=1) H mode it is natural to imagine that also on FAST there will be a noticeable ELM activity. Consequently, it is quite important to figure out what types of ELMs have to be expected, verify their compatibility with machine PFCs and optimize plasma operations to allow as much flexibility as possible to study the ELMs behaviour in reactor relevant conditions.

The estimate of the heat load on divertor by ELMs is based on the hypothesis, valid for a good H-mode with low pedestal dimensionless collisionality ($\nu^*_{PED,FAST} \sim 0.1$), i.e. for a big ELM, that the ELM energy $W_{ELM} \sim 0.15\ W_{PED}$, with the energy confined in the plasma pedestal $W_{PED} \sim 0.4\ W_{TOT}$ [7]. Using this assumption for the FAST H-mode reference scenario at I_p = 6.5 MA, but at rather low density ($n_e/n_{eGW} \approx 0.3$, with n_{eGW} the Greenwald density [14]), we obtain $W_{ELM} \approx 1.5$ MJ and $f_{ELM} \approx 10$ Hz. Then it is to be considered that for $W_{ELM}/W_{PED} > 0.1$ only about half of W_{ELM} reaches the divertor [15] and the ELM decay time is larger than the rise time so that the fraction of this energy mostly contributing to material damage, i.e. the one deposited in short timescales [16], is about 40% for low collisionality [17]

By assuming the same spatial deposition profile as inter-ELM and a factor 2 asymmetry in the in-out ELMs energy deposition, the energy density on the inner divertor is expected to be about 0.4 $MJ \cdot m^{-2}$, to be compared with the recommended threshold for damage (0.5 $MJ \cdot m^{-2}$), adopted by ITER for avoiding too strong W erosion [18].

6. Conclusions

A modelling activity of the FAST SOL/edge has been carried out. Three scenarios have been analysed: H-mode reference, H-mode extreme and full Non Inductive Current Drive scenario. First, a simple self-consistent model, 0D in the core and two points in the SOL, was used for getting the trend of the edge parameters dependence on the separatrix density, in a full W machine without impurity seeding. Then the code COREDIV, which couples self-consistently the radial 1D transport of the bulk plasma with the 2D multifluid transport model for the SOL, was applied to all the cases with strong core-edge coupling and when Ne or Ar seeding was included in the simulation to mitigate divertor heat loads. The main results from this modelling are: in the H-mode reference scenario (I_p = 6.5 MA, B_T = 7.5 T, $\langle n_e \rangle = 2.0 \cdot 10^{20}\ m^{-3}$, P_{ADD} = 30 MW) impurity seeding could reveal not essential, with a beneficial effect on the core Z_{eff} (~ 1), the outer divertor heat load exceeding only marginally the design value of 18 $MW \cdot m^{-2}$. This question could be resolved only by the direct experiment that then would provide very precious information for the reactor plasma facing materials and for the SOL physics. However, impurity seeding could again be necessary if tales shaping were introduced and the effective divertor area were substantially reduced. This is not the case for the full NICD scenario (I_p = 2.0 MA, B_T = 3.5 T, $\langle n_e \rangle = 1.0 \cdot 10^{20}\ m^{-3}$, P_{ADD} = 40 MW): impurity seeding, in connection with the higher plate temperature, results in a strong production of tungsten atoms by sputtering of divertor tiles by Ne or Ar ions and in turn in a unacceptable increase of the core Z_{eff}. In this scenario, COREDIV simulations show that by slightly increasing the density, without impurity seeding, core Z_{eff} can be reduced with a negligible increase of q_{target}. On the contrary, in the extreme H-mode without impurity seeding (I_p = 8.0 MA, B_T = 8.5 T, $\langle n_e \rangle = 5.0 \cdot 10^{20}$, P_{ADD} = 40 MW), the large density and the consequent low plate temperature reduce the tungsten production and therefore the radiated power, leading to huge divertor load. In this case impurity seeding has the most beneficial effect: the radiation fraction is increased and the power flowing to the divertor is decreased, the core Z_{eff} value staying always below 1.2. The energy load of a big ELM (W_{ELM} ~1.5 MJ) on the divertor was evaluated in the case of low pedestal collisionality. A value of about 0.4 $MJ \cdot m^{-2}$ was inferred from the available literature, below the presently recommended limit for avoiding W damage by ELMS in ITER.

7. References

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