

ITER divertor operation in low power discharges

A.S. Kukushkin¹, H.D. Pacher², G.W. Pacher³,

V. Kotov⁴, R.A. Pitts¹, D. Reiter⁴

¹ITER Organization, Route de Vinon sur Verdon, 13115 St Paul Lez Durance,
France; ²INRS-EMT, Varennes, Québec, Canada;

³Hydro-Québec, Varennes, Québec, Canada; ⁴FZ Jülich, Jülich, Germany

To defer activation to a subsequent operation phase, ITER will initially use H and He plasmas, intending with the latter to benefit from the lower L-H transition power threshold in He to achieve H-mode, even if heating systems have not reached full capacity. The aim of this early non-nuclear phase is to commission and test the various systems and to acquire relevant operational experience in a non-activated environment. An important aspect will be to study both the physics and engineering performance of the divertor, including the achievement of partial detachment, significant steady state power loading, material erosion and gaining experience of operation on carbon and tungsten divertor targets. This paper addresses this divertor performance, discussing the operational space in pure He plasmas and reporting on initial simulations with raised strike points contacting W surfaces.

Operational window for He-H plasma

We first determine an operating window for a He-H plasma produced by a He gas puff Γ_{He_puff} supplemented by core fuelling Γ_{H_core} (H pellets since there are no He pellets) to increase the density to prevent neutral beam shine-through. We follow the philosophy of [1] using the parameterized results of SOLPS scrape-off layer (SOL) and divertor plasma simulations as the boundary conditions for a 1D core-and-pedestal model. The latter includes the core transport of energy (MMM model) and particles ($D_{\perp} = 0.1(\chi_{Le} + \chi_{Li})$ throughout the core). In H-mode, appropriate for higher helium concentrations, a pedestal is formed in the model by reduction of the transport coefficients near the edge (at locations where a criterion for suppression of the plasma instabilities is satisfied), and the pressure gradient is then limited to the ballooning limit (see Ref. [1] for the details). In L-mode, appropriate for lower helium concentrations, it is assumed that no stabilization occurs, so that no pedestal is formed and the above transport coefficients apply up to the separatrix. The particle sources are the neutral fluxes of He and H across the separatrix as obtained from the SOLPS results with an addition due to the H pellet fuelling, represented simply by an H particle source with fall-off length of 0.4 m from the separatrix inward. Since current experiments indicate that an anomalous inward pinch may exist, but no reasonable model for this pinch is available, we postulate a pinch given by $k_{pinch}(2r/a^2)D$ and vary k_{pinch} . All calculations are carried out at half the nominal toroidal field for ITER in order to minimize the threshold for the L-H transition in He-H plasmas, and at half the nominal plasma current to give the same q .

Scans are performed at different $\Gamma_{H_core} = 1 - 20 \text{ Pa}\cdot\text{m}^3/\text{s}$ and $k_{pinch} = 0 - 2$ in increments of 0.25, increased stepwise after steady state is reached. We require that the He-H operation be conducted at significant divertor power loading, with peak power flux density, $q_{pk} = 5 \text{ MW}/\text{m}^2$ (a case of lower power loading is considered in [2]) and accordingly adjust the He puffing rate (according to Eq. 1 and 2 of [3]) to maintain this value for all the conditions considered here. Studies on ASDEX Upgrade indicate that an H admixture does not affect the L-H threshold until the helium ion fraction $\zeta_{He} = n_{He} / (n_{He} + n_H)$ drops below 70% at some

point near the edge but then the threshold rises to its level in H plasmas ([4], see also discussion in [3]). We take this point to be $r/a=0.9$, at the top of the pedestal for our H-mode cases, and require that $\zeta_{\text{He}}(r/a=0.9)$ exceed 0.7 for good H mode operation. We further require that the average electron density $\langle n_e \rangle$ exceed $0.3 \cdot 10^{20} \text{ m}^{-3}$ to avoid beam shine-through [5]. Calculations were performed both at reduced power (60 MW, Fig. 1) and full power (73 MW, Fig. 2).

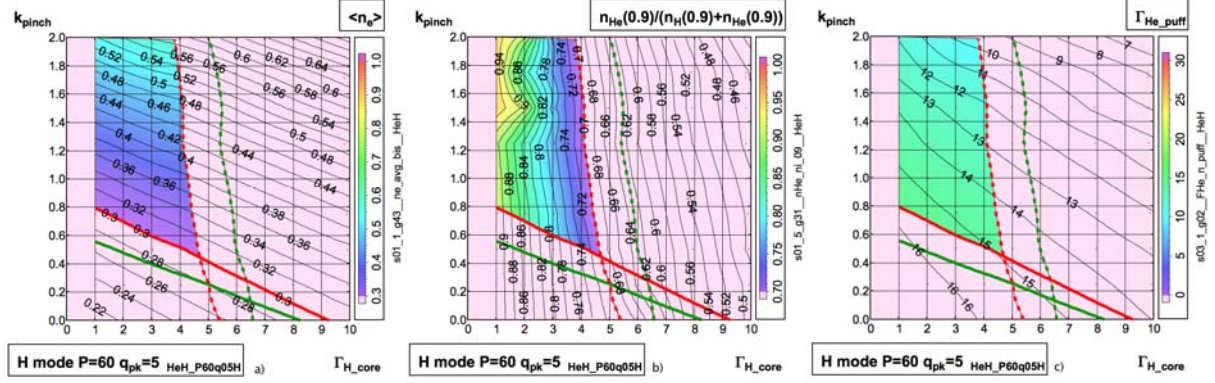


Fig. 1 Contours of a) $\langle n_e \rangle$ [10^{20} m^{-3}], b) $\zeta_{\text{He}}(r/a=0.9)$ and c) $\Gamma_{\text{He,puff}}$ [$\text{Pa}\cdot\text{m}^3/\text{s}$], in the plane $k_{\text{pinch}} - \Gamma_{\text{H,core}}$ [$\text{Pa}\cdot\text{m}^3/\text{s}$] for $P_{\text{aux}}=60 \text{ MW}$ and $q_{pk}=5 \text{ MW/m}^2$. Regions outside the limiting curves (see text) for 60 MW, or not calculated ($\Gamma_{\text{H,core}} < 1 \text{ Pa}\cdot\text{m}^3/\text{s}$) are blanked out in pink. Limits $\langle n_e \rangle = 0.3 \cdot 10^{20} \text{ m}^{-3}$ (solid) and $\zeta_{\text{He}}(r/a=0.9)=0.7$ (dashed) are superposed for $P=60 \text{ MW}$ (red) and $P=73 \text{ MW}$ (green).

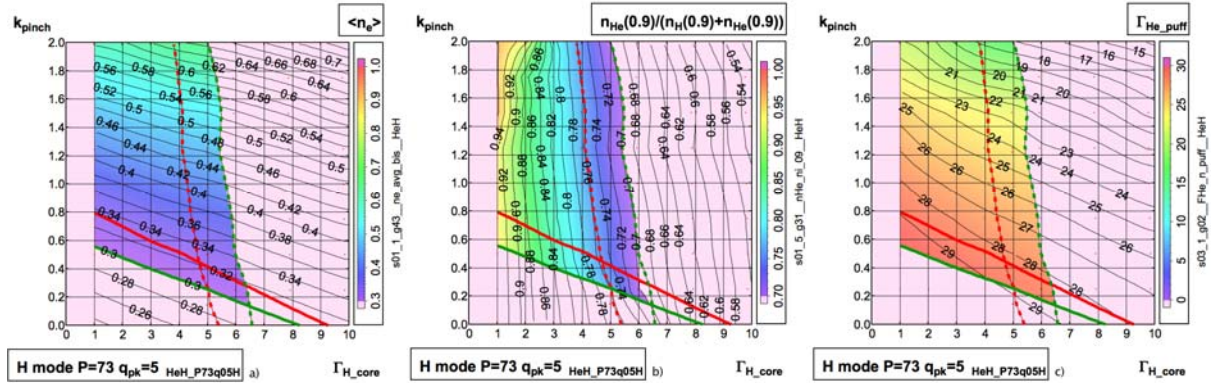


Fig. 2 As in Fig. 1 but for 73 MW rather than 60 MW.

Fig. 1 shows that $k_{\text{pinch}} \geq 0.5$ is required to satisfy $\langle n_e \rangle > 0.3 \cdot 10^{20} \text{ m}^{-3}$ and $\zeta_{\text{He}}(r/a=0.9) > 0.7$ at 60 MW and $q_{pk}=5 \text{ MW/m}^2$. At full power, 73 MW (Fig. 2), at the same point in the operating diagram a higher $\Gamma_{\text{He,puff}}$ is required by the scaling of [3] to keep q_{pk} the same so that $\langle n_e \rangle$ is higher there despite the somewhat lower confinement time. The same density as at 60 MW can thus be obtained for a lower pinch, $k_{\text{pinch}} \geq 0.2$. In the absence of pinch, q_{pk} would have to be reduced, but we find its minimum value to be limited because detachment ($\mu=1$, see [3, 9]) is encountered, even at full power, before these limits are satisfied. Because the core radiation is weak, the SOL input power, $P_{\text{SOL}} \sim P_{\text{heat}}$, and the usual scalings [6] predict an almost constant L-H threshold for fixed $\langle n_e \rangle$, giving an L-H margin of about 1.9 (2.2) for He and about 1.3 (1.6) for H for 60 (73) MW heating power and $\langle n_e \rangle = 0.3 \cdot 10^{20} \text{ m}^{-3}$. Since this margin is small at 60 MW for H-like operation, the plasma may revert to L mode for this power for $\zeta_{\text{He}}(r/a=0.9) > 0.7$, whereas at 73 MW, the margin is more comfortable. To analyze the result of such a back transition, we have also performed L mode calculations with the same considerations as above. If we were to assume in addition that there is no hysteresis between the L-H and the H-L transitions and that the (anomalous) k_{pinch} is the same in both modes, the L mode case has lower $\langle n_e \rangle$ at the same k_{pinch} and therefore an operating scenario

that maintains $q_{pk} = 5 \text{ MW/m}^2$ and $\langle n_e \rangle \geq 0.3 \cdot 10^{20} \text{ m}^{-3}$ would demand an increase of Γ_{H_core} (Γ_{He_puff} is similar for both modes) but still results in higher $\zeta_{He} > 0.7$ than in H mode, because the pedestal in H mode is mostly H, determined by Γ_{H_core} , and there is no pedestal in L mode. In this case the plasma would dither (oscillate) between H mode and L mode on the timescale of the creation and destruction of the pedestal. Since hysteresis for mixed He-H conditions has not been experimentally studied, and the behaviour of the pinch factor is unknown, a time-dependent calculation for the transition cannot yet be performed. In addition, the critical point in the L-mode profile for a return to H mode is unlikely to be $\zeta_{He}(r/a=0.9)$. Since there is no pedestal and the profiles are much flatter, the average $\langle \zeta_{He} \rangle$ may well be more representative of the transition in L mode. $\langle \zeta_{He} \rangle$ however is < 0.7 also in L mode and thus dithering would not occur. In view of these uncertainties for L mode operation and because the scaling of the H-mode threshold power for mixed He-H operation is also uncertain, it is prudent to maintain the requirement on $\zeta_{He}(r/a=0.9)$.

We conclude that, to have reasonable certainty to obtain a good H mode without excessive beam shine-through at significant peak power loading $q_{pk} = 5 \text{ MW/m}^2$, a moderate k_{pinch} of at least 0.5 for 60 MW or 0.2 for 73 MW is required, whereas at $k_{pinch}=0$ even a reduction of q_{pk} would not be sufficient to ensure quasi-stationary operation with a good He H-mode unless the power could be increased further.

Operation with strike points on tungsten.

Although ITER will begin non-active phase experiments with CFC on the highest heat flux handling areas of the divertor targets, the design includes a transition to the W baffle region – low enough for operation on W if the strike points are raised compared with the reference equilibrium [7]. In this way, early experience can be acquired for subsequent ITER operation with the full W divertor planned for the active phases. Initial SOLPS modelling of the divertor performance in this configuration was carried out to see the effect of the change of the geometry before a change in the material. Runs have therefore been performed for an all-carbon wall and using D as the working gas. Fig. 3 shows the variation considered: in the F60 configuration the wall and divertor shape remain the same as in the reference F57 geometry [8] but a new equilibrium has been generated using the CORSICA code with strike points just above but close enough to the CFC-W transition to ensure separatrix contact still on the vertical part of the target plates. To facilitate comparison with the standard D-T operation conditions, the first series of SOLPS runs assume the same DT nuclear operation as the reference ($P_{SOL} \sim 100 \text{ MW}$). However, since higher strike points are only possible at lower plasma current (hence lower density), the fusion power and the He production rate were reduced by a factor 4, resulting in $P_{SOL} = 80 \text{ MW}$ ($Q \cong 2$).

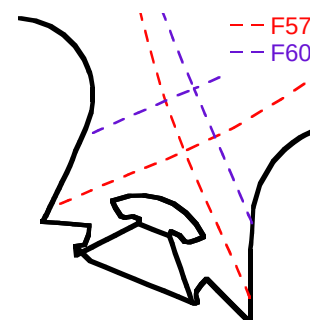


Fig. 3. Configuration variation: reference (F57) and different equilibrium (F60)

The results are illustrated in Fig. 4. The roll-over of total target particle flux (I_{sat}) in the modified geometry occurs at lower neutral pressure, p_n in the private flux region (PFR). This appears natural given the increased area of the plasma contact with neutral gas in the PFR and hence higher neutral influx for the same p_n . The peak power for the F60 geometry is close to that in the standard F57 with $P_{SOL} = 80 \text{ MW}$, taking into account the pressure re-normalization to the detachment point indicated by the I_{sat} roll-over. (The incidence angles of the magnetic field on the target are 2.6° (2.5°) on the outer target for F57 (F60).) The in-out asymmetry of the total power delivered to the targets reduces as the strike points are raised (easier neutral exchange between the divertor legs). The effective pumping speed S_{eff} , defined

as the ratio of pumping throughput to p_n , remains similar to that in the standard case, at least for $p_n \geq 2$ Pa. This is understandable since the neutrals in PFR are collisional and the gas conductance of the “void” volume above the dome is large compared to that of the channel under the dome (for both F57 and F60). At lower p_n , S_{eff} falls sharply, preventing us from finding a solution with balanced He production and pumping.

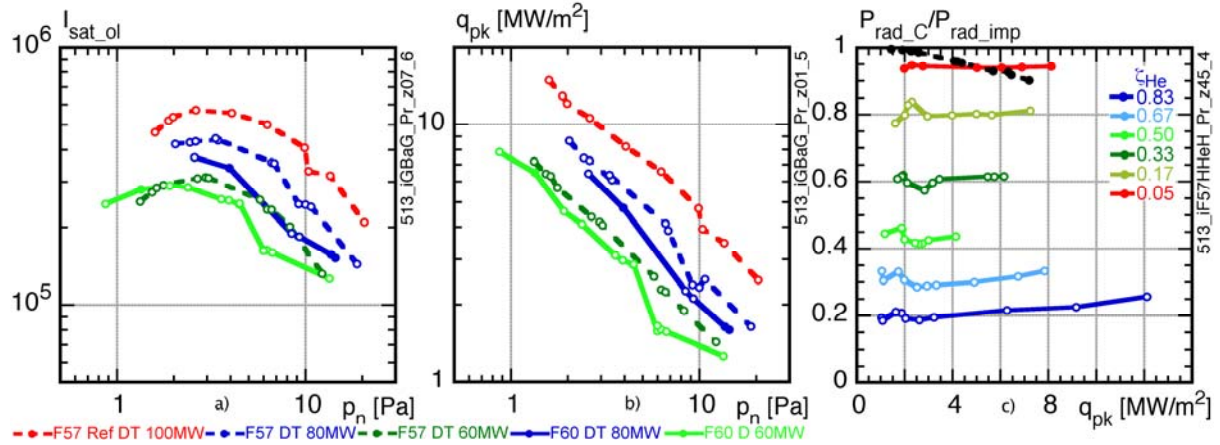


Fig. 4. a) I_{sat} and b) q_{pk} on the outer target vs. p_n for the configurations of Fig. 3. Both D-T and D (no He) cases are shown for F60. c) Carbon-produced fraction of the total power radiated from impurities in He+H plasmas, $P_{SOL} = 60$ MW, vs. q_{pk} for various $\zeta_{He}(edge)$ (ζ_{He} integrated over the edge plasma). D plasma is also shown with dashed black line.

For less activated operation with D plasma without T, still a full C machine and no He, P_{SOL} is determined only by the available auxiliary heating, so we take $P_{SOL} = 60$ MW. It is seen (Fig. 4) that q_{pk} can reach 5 MW/m² for which the W targets are rated, which would ensure adequate technology testing. Although for this geometry we have so far done only calculations with C and not W with impurity seeding, we note (Fig. 4c) that with He+H plasmas in the standard F57 configuration most of the impurity radiation comes from He rather than from C for $\zeta_{He}(edge) > 0.5$, the conditions of interest. Since calculations [3] have found that q_{pk} in He plasma can be close to that in D and it was found in [9] that Ne-seeded discharges show performance similar to that of discharges with sputtered C as the main radiator, we would expect that replacement of the minor contributor to the radiation for these He plasmas, intrinsic C, by seeded Ne will not lead to major changes in the parameters obtained. Our results therefore indicate that He plasma with the separatrix strike-points on the W parts should be able to serve as a representative test bed for ITER plasma operation with full-W targets and with additional impurity seeding in terms of power loading, but not W erosion by the seeded impurity ions. Simulations of W divertor operation with some impurity seeding in H-mode using He plasma with elevated strike points are in progress.

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization

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