Diverted negative triangularity plasmas on DIII-D: The benefit of high confinement without the liability of an edge pedestal

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Abstract. Diverted discharges at negative triangularity on the DIII-D tokamak sustain normalized confinement and pressure levels typical of standard H-mode scenarios ($H_{98y2} \approx 1, \beta_N \approx 3$) without developing an edge pressure pedestal, despite the auxiliary power far exceeding the $L \rightarrow H$ power threshold expected from conventional scaling laws. The power degradation of confinement is substantially weaker than the ITER-89P scaling, resulting in a confinement factor that improves with increasing auxiliary power. The absence of the edge pedestal is beneficial in several aspects, such as eliminating the need for active mitigation or suppression of edge localized modes, low impurity retention and a reconstructed scrape-off layer heat flux width at the mid-plane that exceeds the ITPA multi-machine scaling law by up to 50%. Together with technological advantages granted by placing the divertor at larger radii, plasmas at Negative Triangularity without an edge pedestal feature both core confinement and power handling characteristics that are potentially suitable for operation in future fusion reactors.

Keywords: Reactor, TEM, ELM, Negative Triangularity, ballooning

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1. Introduction

The leading candidate for operation in future magnetically controlled reactors is the H-mode regime \[1\], an excited state (i.e. above ground state) in which the level of energy and particle confinement necessary to self-sustain thermonuclear reactions is obtained by employing an edge transport barrier (ETB). Such barrier is a narrow layer near the plasma edge where the radial pressure profile increases rapidly thanks to turbulence suppression by strongly sheared plasma flow \[2\]. Although attractive thanks to its good confinement properties, the H-mode regime extrapolated to reactors suffers from congenital challenges, the main of which is the conflicting requirements dictated by the core and the wall in the presence of an ETB. More specifically, in order to operate in the H-mode regime, the power flow across the last closed flux surface (LCFS) must exceed and constantly remain above the \( L \rightarrow H \) power threshold; however, the power flow exiting a reactor plasma must not exceed the damage limit of plasma facing components (PFCs). As a consequence, since the power flow cannot decrease below the \( L \rightarrow H \) power threshold for the H-mode regime to persist, intense research is being conducted worldwide to obtain stable H-mode regimes in which most of the power can be quickly radiated in the region between the plasma and the wall. Such configuration results in effective shielding of PFCs from the main plasma and is called detachment \[3\].

Due to economical reasons, for a given size of the toroidal field coils it desirable for such regions to be as small as reasonably achievable because the cold plasma therein does not undergo fusion reactions. This results in the need to trigger detachment in small volume, which is usually achieved by seeding highly radiating impurities \[4\]. However, detached regimes are not easily controlled as they can undergo spontaneous transitions between attached and partially detached conditions, a phenomenon known as detachment cliff \[5\]. It has been recently predicted that the detachment cliff becomes more and more severe as the ETB steepens, due to poloidally driven plasma flows that are typically strongly driven in the H-mode regime \[6\]. Pedestals also give rise to other outstanding issues hampering the H-mode regime. The extremely low transport coefficients characterizing ETBs are such that, at fixed values of energy and particle fluxes originated in the core, pedestals develop radial pressure gradients large enough to trigger bursting instabilities known as edge localized modes (ELMs). These instabilities, if uncontrolled, cannot be tolerated in future reactors because energy fluxes they convect to the wall are projected to significantly decrease the lifetime of PFCs in absence of a large expansion of the nominal heat flux footprint on the wall \[7\]. Therefore, if reactors are to operate in the standard H-mode regimes, ELMs will have to be suppressed or mitigated by active techniques such as resonant magnetic perturbations (RMPs) or ELM pacing via pellets or periodic vertical movement of the plasma column \[8\]. It goes without saying that, due to safety protocols, the use of passive techniques would be much preferred in future reactors. Additionally, the high particle confinement that characterizes the H-mode regime causes significant core impurity retention, thereby lowering fusion performance due to excessive dilution of the main ion species or radiation from high-Z first wall
Most of such challenges might be solved, or at least alleviated, by modifying the cross-sectional shape of the plasma in such a way as to reverse its triangularity from positive to negative values, provided that sufficient confinement can be maintained. Experiments on the DIII-D and TCV tokamaks demonstrated that confinement and pressure levels typical of the H-mode regime can be obtained without ETBs. More specifically, in inner-wall limited (IWl) configurations with relaxed edge pressure profiles, the overall confinement significantly exceeds that of the conventional L-mode regime due to a reduced level of turbulent losses that correlates with a lower intensity of fluctuations. Such configuration was first explored on the TCV tokamak, where H-mode grade confinement was obtained in purely electron heated discharges at low plasma pressure \[9\]; these results were interpreted from a non-linear gyro kinetic standpoint as a stabilization of trapped electron modes (TEM) exerted by negative triangularity \[10\]. The TCV results were extended on DIII-D to a more reactor relevant regimes where the ion temperature is close to that of electrons and where stable operation at moderately high normalized pressure values was for the first time demonstrated \[11, 12\]. After the installation of a neutral beam heating system, high confinement regimes with electron to ion temperature ratio close to unity was also obtained on TCV \[13\].

The absence of a pedestal naturally removes ELMs as well as the need to maintain the energy flux crossing the LCFS to levels above the L $\rightarrow$ H power threshold, thereby allowing the development of a radiative mantle regime in which a significant portion of the energy exhaust is radiated inside the confined plasma region and confinement is not affected by the detachment front approaching the pedestal. Compared to standard H-mode scenarios, the scrape-off layer in such regime will be characterized by lower power, due to the radiative mantle, and higher density, due to the absence of the pedestal, which are expected to facilitate divertor detachment. Additionally, lower pressure gradients near the edge of the confined plasma region should ease the control of the detachment cliff, if present. Finally, the absence of an ETB is expected to yield worse than H-mode particle transport, thereby lowering the impurity content of the plasma.

This paper extends the inner-wall limited results above to a diverted configuration, thereby exploring the properties of this plasma shape regarding edge/wall parameters such as L $\rightarrow$ H transition and heat flux convected to the divertor plate. The paper is organized as follows: Sec. 2 describes how the experiment was designed and executed; Sec. 3 overviews the main results; conclusions in Sec. 4 end the paper.

2. Overview of the experiment

As briefly mentioned in Sec. 1, experiments on the TCV ad DIII-D tokamaks demonstrated that IWl plasmas with a negative triangularity shape sustain H-mode levels of confinement despite being operated with relaxed edge pressure profiles typical of L-mode regimes. More specifically, normalized confinement enhancement factors (\(H_{98y2}\)) close to unity were obtained on both machines \[11, 13\], with DIII-D discharges being able
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To also reach relatively high normalized pressure corresponding to $\beta_N \simeq 3$ [12]. In order to explore whether negative triangularity plasmas could be a viable solution in reactors, the IWL scenario has been recently extended to diverted configurations by creating a novel lower single null equilibrium visible in figure 1. The up-down symmetric shape employed in the IWL configuration did not allow diverted operation, at any power level, because portions of the outer-wall on DIII-D are not armored to withstand the power flux at the strike locations of a diverted symmetrical negative triangularity plasma. In order to maintain a negative top-bottom averaged triangularity, a double null control was developed that features a dominant lower X-point placed at the outermost radial location that decreased the lower triangularity to near zero while keeping the outer strike-point off of outer wall locations that are not appropriately armored. The non-dominant upper X-point is placed at a location similar to that in the symmetric limited configuration and is controlled to define the 2.0 cm scrape-off layer (SOL) surface. The separatrix above the mid-plane then displays strong shaping with upper triangularity $\delta_u = -0.4$. This plasma shape features upper-lower averaged triangularity that is negative and is compatible with operations at continuous high auxiliary power on DIII-D. The separatrix on the midplane at the low field side was set to a position compatible with high spatial resolution measurements from the charge exchange recombination (CER) diagnostic [14], while the separatrix gap on the high-field-side was set to a value equal to a few e-folding lengths of the heat flux in the SOL. The successful creation of this unusual shape was streamlined by the flexibility of the DIII-D control system, which employed recently installed bipolar power supplies that allowed creation and control of the X-point after forming an up-down symmetric breakdown null. The unusual control trajectory from null to negative triangularity asymmetric double null was developed within the TokSys development environment [15] that utilized a full predictive simulation of the discharge from breakdown to flattop. Plasma discharges were executed at plasma current $I_P = 0.9$ MA, value dictated by current limits in the F-coils, line averaged density $\langle n_e \rangle = 3.5-5 \times 10^{19} \text{ m}^{-3}$, elongation $\kappa = 1.6$; the toroidal field was oriented in such a way as to direct the ion $\nabla B$ drift towards the X-point, and its value was set to 2T to ease diagnostic coverage, resulting in $q_{95} \simeq 4.2$ and normalized current $I_p/(a B_T) = 0.75$. The main diagnostics monitoring the experiment are listed below. The electron temperature was measured by Thomson scattering [16] (TS) and electron cyclotron emission [17] (ECE); density was gauged by TS and CO$_2$ interferometer [18]; CER was used to obtain the main and impurity ion density and temperatures, along with the toroidal and poloidal velocity components; impurity transport was probed with the recently installed laser blow-off system; divertor conditions were monitored by fixed Langmuir probes [19], infra-red cameras [20] as well as the divertor thomson scattering diagnostic [21] (DTS); the proximity of ideal beta limits was monitored by active MHD spectroscopy in high power phases; density fluctuations were mainly detected by beam emission spectroscopy [22] (BES). The radial coordinate used in the paper is defined as the normalized poloidal flux $\psi$. Plasmas were heated with NBI power steps designed to probe the L $\rightarrow$ H power threshold as well as to obtain time windows long enough to
**Figure 1.** Lower Single Null equilibrium at Negative Triangularity along with the value of the measured currents in the F-coils. Positive currents means that the coil is pulling the plasma towards itself. Contour lines in the SOL are traced every e-folding length of the heat flux.

compute time averages suitable to estimate both the degradation of confinement with auxiliary power and MHD stability. In order to maintain electron and ion temperatures as close as possible, half of the discharges were also heated with the 110 GHz electron cyclotron (EC) heating system configured to deposit 1.5 MW of power, which was the maximum available to this experiment, near the $\psi = 0.2$ surface. Besides heating, a short EC heating phase was also used to make breakdown more reliable. The presence of a divertor caused two major changes with respect to the previous IWL experiments. First, plasmas maintained an approximately constant line averaged density up to five beam sources, featuring a slight density increase when more sources are used, despite the strike points not being optimally located for efficient cryo-pumping; second, the effective ion charge radial profile, $Z_{\text{eff}}$, was in the range $1.4 - 1.7$. These are significant improvements over previous inner-wall limited experiments for which the line averaged density increased with any level of beam fueling and $Z_{\text{eff}}$ was as large as 3. Discharges, despite being generally free of tearing modes, featured fishbone activity.

### 3. Main results

The main results obtained in this experiment are grouped in three sections: L → H transition and edge MHD stability will be exposed in Sec. 3.1, core confinement in Sec. 3.2 and the properties of the scrape-off layer will be the subject of Sec. 3.3.
3.1. H-mode transition and MHD pedestal stability

In this experiment only one discharge (180520) was observed to transition into H-mode when, while subject to constant heating and fuelling, an issue in the control of the F7A coil (Fig. 1) relaxed the shape of the upper part of the separatrix with the lower part of the poloidal cross section being unaffected.

The dynamics of the L → H transition is shown in some detail in Fig. 2. Beam auxiliary heating is increased from 1.2 MW to 3.7 MW at 2.0 s with the normalized pressure reaching a new stationary state in about 100 ms; as the upper triangularity relaxes to $\delta_u = -0.3$, the $D_\alpha$ signal shows a characteristic dithering behavior commonly interpreted as proximity to the L → H transition. At 2.2 s, when the triangularity further relaxes to $\delta_u = -0.18$, the plasma transitions into H-mode. The resulting ELMs are seen to increase in frequency with increasing auxiliary power following a functional dependence that, in the power window available to the experiment, is approximately linear between 250±30 Hz at $P_{\text{net}} < 3$ MW to 350±40 Hz at $P_{\text{net}} > 5$ MW, where errorbars correspond to one-σ population uncertainties. The measured energy loss caused by each ELM is between 1% and 3.5% of the stored energy which, due to the high frequency, translates into 68% ± 12% of the average power transferred to the SOL. The ELM frequency span is too limited to discern any statistically significant dependence of the energy loss on the ELM frequency itself. Previous experiments on the TCV tokamak using a similar shape observed H-mode transitions with $\delta_u = -0.2$, although
more values of the top triangularity were not attempted, and resulted in ELMs with comparable characteristics [23]. It is interesting to note that the pressure pedestal top of the only H-mode case obtained reaches values in the region 2−3 kPa, significantly lower than that typically obtained in similar discharges at positive triangularity. This result was expected based on the Peeling-Balloning stability of plasmas at negative triangularity [24] and is consistent with extensive modelling of TCV experiments reported in [25]. Detailed radial profile analysis was typically performed in 200 ms time windows along the power staircase used to heat the discharge, adjusting the position of the electron density and temperature profiles with respect to the separatrix. The electron temperature at the separatrix, \( T_{e-\text{sep}} \), was set using the two-point model [26], which constrains the kinetically corrected electron Spitzer parallel heat flux computed using Thomson electron and density radial profiles to that derived using the power entering the SOL. The procedure is iterated until values for \( T_{e-\text{sep}} \) and for the power fall-off length in the SOL, \( \lambda_q \), are self-consistently obtained. When comparing radial profiles between the H-mode case and an L-mode case at similar input power and line averaged density, displayed in Fig. 3, it is apparent how the two cases have similar ion and electron temperature profiles, while the electron density in H-mode clearly exceeds its counterpart in L-mode.

The stability of the H-mode pedestal was investigated following a standard workflow described as follows. For a given kinetic equilibrium reconstruction, the pressure and the current density in the pedestal region were varied independently and the corresponding equilibrium recomputed using the EFIT code as a fixed boundary solution. The resulting varied equilibria were subsequently read by the ELITE [27] code to generate the stability diagram over ten low-n modes in the range \( n = 5 − 50 \); the maximum n-number was increased from the commonly used value 30 so that the most unstable mode in any modified equilibrium near the experimental point is within the computed range. The reconstructed profiles sit in the proximity of the stability boundary, as seen in Fig.3 which displays the stability diagram for discharge #180520 generated with profiles measured between 80% and 98% of the ELM cycle in the time window 3200−3400 ms. The same analysis with a modified value for \( T_{e-\text{sep}} \), which accounts for error-bars.
propagating through the two-point model workflow, does not significantly alter the results. The stability boundary is computed using an effective diamagnetic stabilization derived from a bi-linear fit to BOUT++ simulations [28]. This model, implemented as default in EPED v1.6, captures a roll-over of the effective diamagnetic stabilization at high mode numbers and performs significantly better than setting the threshold to half the ion diamagnetic frequency; indeed, if the latter model is used, all the experimental points tend to lie in the stable region. It has also to be mentioned that when profiles measured in a longer ELM cycle are retained, experimental points in all time windows considerably depart from the stability boundary to sit in the stable region. This is consistent with the fact that, during the ELM cycle, the pressure profiles in the pedestal region are seen to monotonically increase between any two consecutive ELM crashes. The stability diagram displayed in Fig. 4 as compared to similar diagrams computed for standard H-modes at positive triangularity, shows that the stable region at relatively low-n numbers is considerably restricted at negative triangularity. This result is in agreement with previous work in the literature; in particular, detailed comparisons of stability diagrams for positive vs negative triangularity plasmas on TCV can be found in [25].

The fact that the transition displayed in Fig. 2 happens at varying triangularity and fixed auxiliary power and fuelling, after twice the fast ions build-up time, hints to a triangularity dependence of the H-mode power threshold. In this respect, it is important to underline that all other discharges remained robustly in L-mode, despite attempts to trigger the H-mode transition were carried out by altering the heating mix, the line averaged density and the vertical position of the X-point. More specifically, based on the multimachine scaling for the L → H power threshold [29], the expected threshold of net coupled power is approximately equal to 2 MW for these NegD discharges; nevertheless, plasmas that maintained the nominal shape did not transition to H-mode despite the power flow crossing the separatrix exceeded the expected L → H power threshold by a factor of 4 – 5. Further indication of a triangularity dependence of the L → H power
threshold can be obtained in the previous IWL experiments [11], where confinement properties of negative triangularity plasmas were also evaluated by direct comparisons to those in sister discharges at $\delta = +0.4$ with matched actuators ($I_P$, $B_T$, $\langle n_e \rangle$, $P_{aux}$). Discharges at positive triangularity underwent an H-mode transition at 6 MW power level, i.e. about half the value their negative triangularity counterparts sustained while remaining in L-mode [12]. This result could make L-mode edge a viable candidate for operation in future reactors, provided that the $L \rightarrow H$ power threshold will remain large enough to prevent plasmas from developing an edge pedestal. While, in the IWL case, one could argue that varying triangularity affects the plasma-limiter interaction thereby altering the $L \rightarrow H$ power threshold, it is evident that the triangularity dependence observed in the experiment at fixed X-point positioning and wall recycling has to do with some particular effect near the separatrix. In this work two options were explored: Reynolds’ stresses and ballooning stability. The former were probed in view of a collection of recent results which suggest that the $L \rightarrow H$ transition is connected to the interaction of turbulence with shear flow near the plasma edge [30]. Reynolds’ stresses were estimated by the BES diagnostic, whose detectors were organized in a $8 \times 8$ matrix located in the region $\psi > 0.8$ and crossing the separatrix, i.e. in a position allowing one to measure any ion gyro-radius scale fluctuation able to impact the creation/growth of an edge pedestal. In Fig. 5 we compare sister discharges #180519 and #180520; the former maintained the design shape and stayed in L-mode while the latter, due to the relaxed shape, transitioned in H-mode at 2.205 s. Neither the poloidal velocities, the inferred Reynolds stress (RS), nor their spatial gradients show any notable difference that would suggest a measurable impact of triangularity on this mechanism. More specifically, although the measured levels of the RS and of its radial gradient are similar to those measured in other $L \rightarrow H$ transitions [31], the transition is somehow inhibited at strong negative triangularity. The second option explored the impact of triangularity on the ballooning stability by using the HELENA [32] and BALOO [33] codes. It is known since the 1980s that the stability of ballooning modes worsens at negative triangularity due to the increased length field lines spend in the bad curvature region. In the $n = \infty$ ballooning formalism, each flux surface can be treated independently and its stability is determined by the normalized pressure gradient, $\alpha$, crossing a given threshold. Ballooning stability diagrams can be constructed for any given experimental case and flux surface by solving the stability equation for the values of $\alpha$ that destabilize the modes [34]. In order to evaluate whether ballooning modes determine accessibility to H-mode profiles, the stability diagram was computed for four cases. The first two cases correspond to the actual experimental cases in the H-mode plasma, i.e. $\delta_u = -0.18$, and in one of the L-mode discharges at similar power level, i.e. $\delta_u = -0.36$. The remaining two cases correspond to an idealized experiment in which each pressure profile was realized in the plasma shape for which the other profile was measured. Such stability diagrams, displayed in Fig. 6, show that L-mode profiles are ballooning limited at any radial location at both values of triangularity, although the threshold is higher for the relaxed shape; however, as the pressure profile steepens, the
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Figure 5. Time evolution of discharge #180519 (Blue, L-mode throughout) and #180520 (Red, H-mode transition at 2.205 s). From top to bottom: poloidal velocity and its radial gradient, $D_\alpha$ signal, Reynolds stress and its radial gradient.

Figure 6. Stability diagrams, expressed as a function of the normalized pressure gradient ($\alpha$) versus the radial coordinate ($\psi$), of $n = \infty$ ballooning modes for pressure profiles measured in L-mode (left), and a profile corresponding to 40% – 60% weighted average between L- and H-mode (right). The stability is calculated for a plasma shape featuring both $\delta_u = -0.36$ (red) and $\delta_u = -0.18$ (blue). (Adapted from [35])

Bootstrap current opens access to second stability at $\delta_u = -0.18$, with the $\delta_u = -0.36$ case remaining ballooning limited. This might explain the lack of an L → H transition despite the fact that the measured levels of the RS and of its radial gradient are similar to those measured for transitions in plasmas with positive triangularity. A complete study of the ballooning stability in these cases can be found in [35], in which it is also shown that, in the case $\delta_u = -0.36$, the pedestal top temperature threshold for access to second stability increases to 1 keV, thereby preventing the edge pedestal from developing in strongly shaped plasmas at negative triangularity. In two discharges, a 40 cm radial sweep of the X-point was executed, gradually moving the lower triangularity from near zero to $\delta_l \simeq 0.5$. Plasmas did not transition into H-mode and experienced a
10% decrease in normalized pressure ($\beta_N$) and normalized enhanced confinement factor ($H_{98,y2}$). The lack of the H-mode transition is consistent with modelling work [21] where it is found that ballooning modes are still driven unstable by the upper negative triangularity which, in the shots where the radial X-point sweep was executed, remained at the design value $\delta_u = -0.4$. The small decrease in normalized confinement and pressure is qualitatively understandable in terms of an increase in top-bottom averaged triangularity from $\delta_{av} = -0.2$ to $\delta_{av} \approx 0$, as discussed in Sec. 3.2.

To conclude, during usual limit cycle oscillation (LCO) phases, the pressure gradient is usually seen to increase slowly approaching the L $\rightarrow$ H transition. However, at negative triangularity the $n = \infty$ ballooning modes might prevent the pressure gradient from triggering the transition, even at moderately high levels of RS, thereby maintaining the plasma in the LCO phase. Since we only have one discharge that transitioned into H-mode, future work will be devoted to explore the L $\rightarrow$ H transition more systematically, by comparing turbulence-Reynolds Stress decorrelation dynamics with increased statistical confidence.

3.2. Core confinement

Negative triangularity discharges sustained H-mode grade confinement and pressure levels despite maintaining relaxed edge pressure radial profiles typical of L-mode scenarios. More specifically, plasmas routinely featured relatively high normalized pressure levels in the range $2.8 < \beta_N < 3$, or $\beta_N l_i \approx 3$, and thermal confinement enhancement factor $H_{98,y2}$ up to 1, while being intrinsically free of ELMs. An overview of the time evolution of discharge #180526 is displayed in Fig. 7. Considering now the total stored energy, the power degradation of confinement is weaker than that characterizing the L-mode ITER-89P scaling law, as displayed in Fig. 8, resulting in a confinement enhancement factor that improves with increasing auxiliary power. This result substantially agrees with previous IWL experiments, although in that case the power degradation was measured to be near zero. There exist a few possible explanations for this discrepancy. First, the upper-lower averaged triangularity is $\delta_{av} = -0.2$, as opposed to $\delta_{av} = -0.4$ in the IWL discharges; as such, the stabilizing effect of negative triangularity is expected to be weaker and might result in a more pronounced power degradation of confinement. Indeed, the beneficial impact of negative triangularity on turbulence can be heuristically understood by looking at Fig. 9. If one discards particles close to the barely and the deeply trapped limits, the trajectory of a general trapped particle can extrude into the good curvature region at NegD, while, for the same conditions, it is entirely confined in the bad curvature region at positive triangularity. As the plasma (negative) triangularity in increased, trapped particles spend more and more time in the good curvature region, which has a less destabilizing effect on turbulence. This reasoning can be made more quantitative by considering the idealized case of an up-down symmetric flux surface, with infinite safety factor so as to neglect the poloidal magnetic field, and whose spatial coordinates ($R,Z$) are described by the commonly
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Figure 7. Time evolution of discharge #180526. Line averaged density $\langle n_e \rangle$, injected power from neutral beams ($P_{\text{NBI}}$) and electron cyclotron ($P_{\text{ECH}}$) systems, normalized pressure ($\beta_N$), confinement enhancement factor ($H_{98,y2}$), $D_\alpha$ signal.

Figure 8. Total stored energy as a function of net auxiliary power for NegD discharges heated with NBI (H-mode case #180520 ⪯ and L-mode case #180523 ⪯) and with a mixed NBI and ECH (L-mode case #180533 ⪯). Experiments are compared to the corresponding values expected by the ITER-89P scaling law (▷). Symbol $\gamma$ indicates a best fit estimate of the power degradation of stored energy, corresponding to a power degradation of confinement equal to $\gamma - 1$. 

$W_{\text{tot}} = \alpha P^\gamma$
used Miller equilibrium parametrization
\[ R(\theta) = R_0 + r \cos[\theta + \arcsin(\delta) \sin(\theta)] \]
\[ Z(\theta) = Z_0 + \kappa r \sin(\theta), \]
where \( r \) is the minor radius, \( \kappa \) the elongation, \( \delta \) the triangularity and \( (R_0, Z_0) \) the geometric center of the flux surface. The boundary between the bad and good curvature regions is located at \( \theta = \pm \pi/2 \), for which the corresponding radial location is \( R = R_0 - r\delta \). At those locations, the confining toroidal magnetic field takes the value
\[ B = \frac{B_0}{1 - \epsilon \delta}, \]
where \( \epsilon \equiv r/R_0 \) is defined as the inverse aspect ratio and \( B_0 \) is the magnetic field at the geometric center of the flux surface. The fraction of particles that are constrained to never leave the bad curvature region is given by
\[ F_{BC} = \left( \frac{m}{2\pi T} \right)^{3/2} \int_0^\infty d\mu B \int_{-v_{||,max}}^{v_{||,max}} dv_{||} e^{-\frac{mv_{||}^2 + 2\mu B}{2T}}, \]
where \( m \) is the mass of the given particle and \( T \) is the temperature of the Maxwellian distribution it belongs to. The maximum parallel velocity \( v_{||,max} \) of particles always confined into the bad curvature region is found by setting
\[ \mu B_{\theta=\pi/2} = \frac{1}{2} m v_{||,max}^2 + \mu B_{\theta=0} \]
where \( B_{\theta=0} \) is the magnetic field on the outboard midplane and \( B_{\theta=\pi/2} \) is that at the boundary between bad and good curvature regions. This in turn gives
\[ v_{||,max} = \pm \sqrt{\frac{2\mu B_0}{m} \left( \frac{1}{1 - \epsilon \delta} - \frac{1}{1 + \epsilon} \right)} = \pm \sqrt{\frac{2\mu B_0(1 + \delta)}{m(1 + \epsilon)(1 - \epsilon \delta)}}. \]
The function \((1 + \delta)/(1 - \epsilon \delta)\) decreases monotonically at decreasing triangularity, resulting in a lower fraction of particles confined to the bad curvature region for given values of \( B_0 \), \( \epsilon \), \( m \) and \( \mu \).

The second reason for which the power degradation of confinement is stronger than that observed in previous IWL experiments might be related to the much lower impurity content in the diverted discharges as compared to the IWL counterparts, resulting in \( Z_{\text{eff}} \simeq 1.5 \) vs \( 2.5 - 3 \). It is well known that impurities exert a stabilizing effect on electrostatic instabilities such as the Ion Temperature Gradient ITG via main ion dilution, and collisionless Trapped Electron Modes (TEM) via collisionality; as the impurity content is varied between the two scenarios, the relative strength of ITG and TEM modes can also vary, thereby potentially altering the dominant mode at ion gyro-radius scales. Indeed, linear gyro-kinetic simulations indicate that, while the IWL discharges were dominated by TEM at most radii and wavenumbers [12], ITG modes dominate the ion-scale spectrum in diverted plasmas. More specifically, the growth rate of the most unstable mode at ion scales is mostly sensitive to the ion temperature inverse scale length, \( \nabla T_i/T_i \), while that at higher wave-numbers responds more promptly to varying \( \nabla T_e/T_e \) and \( T_e/T_i \). Additionally, while plasmas in the IWL
Figure 9. Cartoon illustrating the comparison of the trajectory of a given trapped particle (in green) in two cases at positive (top) and negative (bottom) values of triangularity $\delta$. While for $\delta > 0$ the trapped particle is constrained to the bad curvature region, for $\delta < 0$ the same trapped particle has its turning points in the good curvature region.

scenarios were heated with 3.5 MW of EC power, only 1.5 MW was available to the diverted experiments; at similar neutral beam heating, this substantially modifies the ion to electron heating power ratio. This is confirmed by a linear stability survey of a number of discharges: while the wave-number spectrum is ITG dominated up to $k_y \rho_s \simeq 0.5$ in the early part of the discharge, during which the beam power is relatively low compared to that from the ECH system, at higher beam heating power ITG modes are the dominant ion scale instability in the region extending to $k_y \rho_s \simeq 1$. While negative triangularity has a proved stabilizing effect on TEM turbulence [10, 36, 37], it is not yet clear to what extend it stabilizes plasmas where TEM are subdominant to ITG modes. Therefore, when the dominant ion-scale mode is ITG, as in the diverted discharges, the confinement scaling might as well differ from that in TEM dominated regimes, as in the IWL experiments.

ELM-free scenarios in future reactors need to efficiently exhaust impurities and helium ash. The impurity particle confinement time was measured by laser ablating suitable Al and F targets during selected constant heating phases in all discharges. The particle to energy confinement time ratio, $\tau_P/\tau_E$, was measured in all cases to be of order unity, in quantitative agreement with results obtained in the IWL scenario. As opposed to H-mode regimes, for which such ratio is typically measured in the range $2 - 4$, this scenario makes impurity retention less problematic. We note, however, that fuelling in future reactors might be more challenging at NegD, depending on how the main ion particle transport scales with respect to that of impurities. It is worth noting that inferred values of $\tau_P/\tau_E$ are quantitatively similar in the H-mode discharge #180520 and the other L-mode discharges, consistent with the common observation
that impurity retention becomes less severe in standard H-mode regimes characterized by small and rapid ELMs. At the highest power level, the bootstrap current accounts for approximately 20% of the total plasma current, roughly consistent with the previous experiments in the IWL configuration [12].

3.3. Scrape-off layer properties

One of the main reasons that motivated this experiment is to probe divertor power handling at negative triangularity. Discharges were carried out in an open divertor (see Fig.1) near half the Greenwald density, resulting in both divertor legs being attached. The experiment did not focus on divertor detachment, due to the novelty of the scenario, so no attempt was made to seed impurities in the divertor region or to operate near the Greenwald limit to facilitate detachment. The heat flux width in the scrape-off layer was reconstructed from Infrared thermography measurements collected at the divertor plate [38], and subsequently mapped onto the mid-plane where values were found consistent with those inferred using electron density and temperature profiles measured across the separatrix. The inter-ELM power fall-off in the only H-mode plasma obtained (#180520) is consistent with the value predicted by the Eich multi-machine scaling law [39] within its 20% statistical uncertainty (2 mm vs 2.4 mm), as well as with other DIII-D discharges with similar lower-half plasma shape. All the remaining L-mode plasmas featured, on average, up to 50% wider $\lambda_q$ as compared to the H-mode data-point, as displayed in Fig. 10. Comparisons with multi-machine scalings indicate that the $\lambda_q$ values measured in negative triangularity L-mode discharges are, respectively, wider and narrower than those predicted for positive triangularity H- and L-mode plasmas, consistent with the corresponding $\lambda_{Te}$ values measured on the outboard mid-plane [40]. The substantial radial sweep described above also made it possible to measure the
electron density and temperature profiles using the Divertor Thomson system and map them onto the nominal equilibrium. The density profiles, displayed in Fig. 11 show the formation of an extremely high density region located on the high field side of the X-point, consistent with the direction of $E \times B$ drifts which, in a configuration where the ion $\nabla B$ drift is directed towards the X-point, circulates particles into the inner side of the SOL where they accumulate. A similar phenomenon is also observed on the TCV tokamak, although only during EC heating [41].

4. Conclusions

Recent experiments on DIII-D extended high confinement L-mode plasmas at negative triangularity to a diverted configuration. Plasmas routinely sustain H-mode grade confinement and pressure levels, reaching $H_{98,32} = 1$ and $\beta_N = 3$, while maintaining robust ELM free operation. As compared to previous inner-wall limited discharges, these plasmas maintain a steady line averaged density while being heated with up to five neutral beam sources delivering 9 MW and are characterized by low impurity content, with $Z_{\text{eff}}$ averaging 1.5 across the minor radius. The ratio of the impurity to energy confinement time is measured to be of order unity; in contrast to H-mode regimes, for which such ratio is commonly measured in the range $2 - 4$, this scenario is characterized by low impurity confinement. The L $\rightarrow$ H power threshold is postulated to drastically increase at negative triangularity, being around 2 MW at $\delta_u = -0.18$ and exceeding 10 MW at $\delta_u = -0.4$. The reason for such increase is modelled to be due to the stability of $n = \infty$ ballooning modes that, in the shape with higher negative triangularity, close access to the 2nd stability region. The H-mode pedestal is rather shallow, with small ELMs that increase in frequency with increasing auxiliary power. The inaccessibility to the H-mode regime, the weak power degradation and the wider $\lambda_q$ could make L-mode reactors at negative triangularity a viable, robust, solution in future reactors. As it was pointed out in previous work [21, 42], reactors will benefit from having strike
points impinging at outer radii because of a larger separatrix wetted area, less divertor maintenance overhead and lower background magnetic field for internal poloidal field coils. While, at positive triangularity, strike points can be redirected to the low-field-side of the machine using dedicated poloidal field coils, plasmas at negative triangularity automatically feature this property thereby simplifying hardware requirements.

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Diverted NegD plasmas on DIII-D: high confinement without a pedestal

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