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Abstract. New off-axis neutral beam injection (NBI) capability on DIII-D has expanded the range of achievable and sustainable current and pressure profiles of interest for developing the physics basis of steady-state scenarios in future tokamaks. Using off-axis NBI, plasmas have been produced with q_{\min} between ~ 1.3 and ~ 2.5 to evaluate the suitability for steady-state operation ($f_{\text{NI}}=I_{\text{NI}}/I_{\text{p}}=1$). These plasmas typically have broader current and pressure profiles and increased stability. Nearly stationary plasmas were sustained for two current profile relaxation timescales (3 s), with $q_{\min}=1.5$, $\beta_{\text{N}}=3.5$, $f_{\text{NI}}=70\%$, and performance that projects to $Q=5$ in an ITER-size machine. The duration of the high β_{N} phase is limited only by the available NBI energy. Low-order tearing modes are absent and the predicted ideal-wall $n=1$ kink β_{N} limit is >4 . This demonstrates performance close to that required for the ITER and FNSF-AT steady state missions with margin for further improvement. Furthermore these plasmas have been shown to be compatible with a divertor heat flux reduction technique that relies on neon injection into the private flux region to enhance radiated power. To achieve higher f_{NI} , the bootstrap current fraction must be increased, and achieving higher β_{N} and higher q_{\min} is expected to do this. High $q_{\min}>2$ and β_{N} near 5 are also required for a DEMO steady state power plant solution. Experiments to produce plasmas with $q_{\min}>2$ showed that the use of off-axis NBI results in higher sustained q_{\min} , with q_{\min} at a larger radius (i.e. a broader current profile), and a broader pressure profile. These changes increased the predicted ideal-wall $n=1$ kink mode β_{N} limit from $\beta_{\text{N}}\approx 3.5$ to $\beta_{\text{N}}\approx 4$. These plasmas typically achieved a maximum $\beta_{\text{N}}=3.2$ limited by the available NBI power and reduced confinement ($H_{89}<2$) relative to similar plasmas with lower q_{\min} and only on-axis NBI. Enhanced fast ion loss at high q_{\min} is the likely cause.

1. Introduction

Fully noninductive operation ($f_{\text{NI}}=I_{\text{NI}}/I_{\text{p}}=1$) is planned for many next-step tokamaks, including ITER, FNSF-AT [1], and DEMO. One possible scenario for achieving high fusion gain, high bootstrap current fraction ($f_{\text{BS}}=I_{\text{BS}}/I_{\text{p}}$) operation is to operate with elevated minimum safety factor (q_{\min}) and high normalized β_{tor} (β_{N}), since $f_{\text{BS}}\propto\beta_{\text{pol}}\propto q\beta_{\text{N}}$. ITER, the FNSF-AT, and DEMO are all expected to use an elevated q_{\min} scenario with high β_{N} to achieve steady state operation. The approximate expected values are: ITER $\beta_{\text{N}}\approx 3$, FNSF-AT $\beta_{\text{N}}\approx 3-4$, and DEMO $\beta_{\text{N}}\approx 5$.

A goal of the DIII-D program is to develop the physics basis of steady state tokamak operation for the next-step devices. On DIII-D, neutral beam injection (NBI) and electron cyclotron (EC) waves are used for heating and current drive. While EC current drive is predicted to be capable of producing elevated q_{\min} plasmas with reactor relevant $T_{\text{e}}=T_{\text{i}}$, at present NBI remains the primary tool for going to high β_{N} . However, high power on-axis NB current drive (NBCD) results in peaked current density profiles and low q_{\min} . Therefore one of the four beamlines was upgraded in 2010-2011 to provide a flexible injection angle between 0° and 16.5° to horizontal (Fig. 1). When the magnetic field pitch is aligned with the

beam injected at 16.5° significant off-axis current drive was predicted and confirmed to exist [2]. This current density is distributed widely about the half-radius.

Compared to on-axis heating, off-axis heating reduces the on-axis pressure and current density, effectively broadening both profiles which is known to increase the β_N limits to ideal-wall kink modes [3]. For steady state scenario research, the new off-axis NBI capability was used in experiments that had two goals. The first was to access and take to $\beta_N > 4$ plasmas with $q_{\min} > 2$ and broad current and pressure profiles. These are predicted by modeling to have high β_N limits due to increased wall stabilization, and good confinement due to a large volume of weak or negative magnetic shear [4]. The second goal was to extend high performance, quasi-stationary elevated q_{\min} operation to multiple current profile relaxation time scales (τ_R) to confirm passive stability of tearing modes. The first of these goals was meant to test access to the conditions required by a steady state DEMO, and the second goal was focused more on a demonstration of conditions that could be useful for ITER and FNSF-AT.

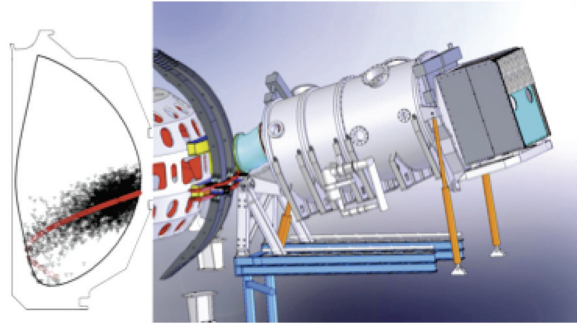


FIG. 1. 5 MW of off-axis neutral beam injection 16.5° to horizontal on DIII-D.

2. Exploration of Access to $q_{\min} > 2$, High β_N Operation

The first steady-state scenario experiments with off-axis NBI sought to demonstrate higher sustained q_{\min} with broader current and pressure profiles, and this was largely successful. In 2009, DIII-D steady-state scenario experiments using only on-axis NBI and ~ 2.25 MW of off-axis electron cyclotron current drive (ECCD) showed that it was difficult to sustain q_{\min} above 2 at $\beta_N \approx 2.7$, $B_T = 2.1$ T, and $q_{95} \approx 6.7$. In 2011 these plasma conditions were reproduced with the following changes: (1) up to 4 MW off-axis NBI, (2) an additional ~ 1 MW of EC power, (3) reversed toroidal field polarity to maximize off-axis NBCD; (4) slightly modified plasma shape: unbalanced double null biased away from the $B \times \nabla B$ drift direction and strike points a few centimeters farther from the divertor pumps to facilitate a match to the line-averaged density of the 2009 plasmas. Three of the six co- I_p beams, including both off-axis beams, were in feedback control mode to match a target β_N waveform. Figure 2 shows a comparison of the key equilibrium quantities obtained in the 2009 and 2011 experiments. The plasma heated by off-axis NBI was sustained with $q_{\min} \approx 2.4$ and $\rho_{q_{\min}} \approx 0.3$ at $\beta_N \approx 2.7$ for as long as NBI energy was available. The pressure profile peaking factor was also reduced from ~ 3.5 down to ~ 2.5 . The pressure profile broadening is due chiefly to a less peaked fast ion pressure profile and increased electron heating at mid-radius by the off-axis NBI and electron cyclotron heating (ECH), and somewhat by a higher pedestal density obtained by less aggressive divertor pumping.

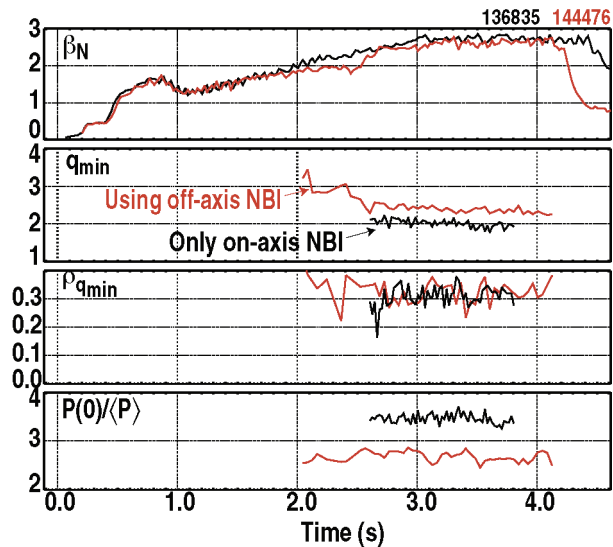


Fig. 2. Using off-axis NBI improves access to and sustainment of broad current and pressure profiles with $q_{\min} > 2$.

After confirming that off-axis NBI helps to obtain more advanced profiles at moderate $\beta_N \approx 2.7$, the next experiments targeted $\beta_N > 4$ by using the maximum available co- I_p NBI power. ECCD was applied at $\rho \approx 0.3-0.6$ to help maintain a broad current profile. Off-axis beams were in β_N feedback mode. Figure 3 shows the time traces of an example discharge operated in this mode. Under these conditions, the energy confinement time was typically less than or equal to that expected of H-modes based on the ITER89P L-mode [5] and ITER98y2 H-mode [6] scaling expressions: $H_{89} \approx 1.7$ and $H_{98} \approx 1$. Thus the maximum achieved β_N was 3.2 and limited primarily by the available power rather than MHD instability. With $q_{min} > 2$, $m/n=2/1$ tearing modes are completely absent. 3/1 modes were observed in the course of optimizing the discharge evolution, but these were ultimately avoided by making adjustments to the L-H mode transition timing, β_N ramp rate, and ECCD timing – all of which affect the current profile evolution. 7/2 and 5/2 tearing modes with an amplitude of 3–5 Gauss at the wall proved common and difficult to avoid when q_{min} passed through 7/2 or 5/2. These modes sometimes disappeared but often remained unstable throughout the remainder of the shot. Based on the calculated island width and the Chang-Callen tearing mode belt model [7] the 5/2 modes in such plasmas are estimated to reduce the energy confinement time by $\sim 18\%$. A known way to increase energy confinement time is to form an internal transport barrier. However, ITB's were avoided in these experiments because of previous work that showed the difficulty of avoiding pressure profile peaking and low ideal MHD β_N limits [8].

Figure 4 shows a comparison of two similar plasmas that were set up to achieve high q_{min} and high β_N using the maximum available power. The case using off-axis NBI maintains $q_{min} > 2$ with $\rho_{q_{min}} \approx 0.5$ and $P(0)/\langle P \rangle < 3$. Consistent with ideal MHD modeling, the ideal-wall $n=1$ β_N limit increases, in this case to over 4.

Since the achievable β_N is limited by the available co- I_p neutral beam power, it is important to understand why the observed energy confinement time (τ_E) is relatively low in cases with off-axis NBI and $q_{min} > 2$. A small reduction in τ_E was shown to occur just by replacing on-axis NBI with off-axis NBI in an otherwise identical discharge with $q_{min}=1.1$ and $\beta_N=3.5$. In this demonstration, both discharges were formed the same way with a mix of on- and off-axis beams, but in one of them the off-

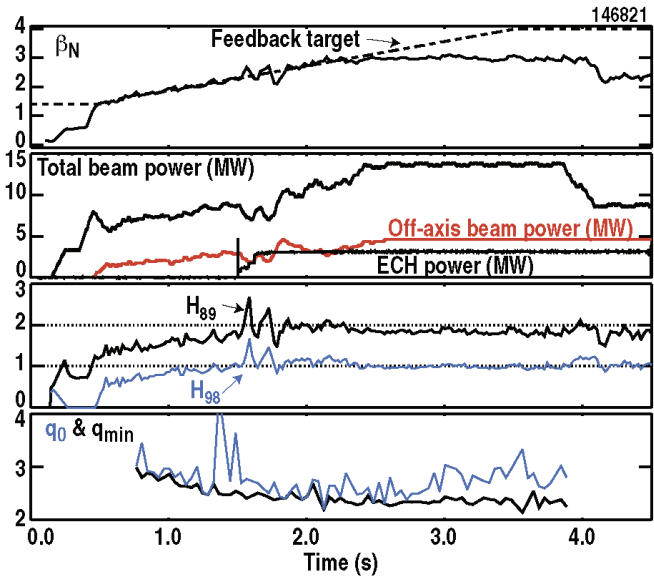


Fig. 3. With $q_{min} > 2$ the maximum achievable β_N was limited by transport and the available power.

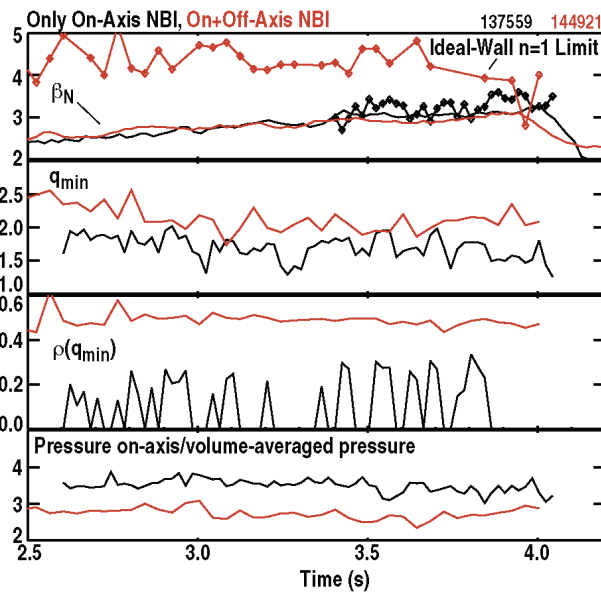


Fig. 4. Off-axis NBI broadens profiles, sustains $q_{min} > 2$, and increases the predicted ideal-wall β_N limit (symbols).

axis beams were turned off once β_N reached 3.5. Subsequently 13% less total power was required to maintain this β_N . Thus τ_E was $\sim 13\%$ less with off-axis NBI, and H_{89} about 6% less. This makes sense when one considers that thermal conductivity typically increases with radius, so unless this profile is changed significantly, applying power at larger radius results in a faster heat loss.

However, this level of confinement reduction is not enough to explain the reduced confinement at $q_{\min} > 2$. Figure 5(a,b) shows the H_{89} and H_{98} confinement scale factors for plasmas with and without off-axis NBI as a function of the average q in the range 0–0.3 (q_{core}). Since strong shear reversal was typically avoided, q_{core} is close to q_{\min} in most cases. These plasmas have β_N within 2.6–3.9 and q_{95} within 4.5–6.8. Here, the thermal stored energy comes from the fitted density and temperature profiles, and the fast ion stored energy comes from using the NUBEAM code within ONETWO. The two plasmas with off-axis NBI and q_{core} below 2 fall generally within the middle of the scatter of H -factors, with $H_{89} > 2$ and $H_{98} > 1$, indicating that these plasmas have confinement typical of similar plasmas without off-axis beams. For $q_{\text{core}} > 2$, the thermal energy confinement factor H_{98} remains above 1, while the total energy (thermal plus fast ion) confinement factor H_{89} falls below 2. This suggests that the relatively low global energy confinement observed in the $q_{\min} > 2$ plasmas with off-axis beams is due to a difference in the fast ion confinement. Fast ions may be lost from the plasma more readily at high q_{\min} due to increased Alfvén eigenmode activity which is observed to increase with q_{\min} . Unfortunately direct measurements of fast ion loss are not available while using the B_T direction required to maximize off-axis NBCD.

Previous experiments on DIII-D transiently accessed good τ_E with $q_{\min} > 2$ and β_N near 4 using a toroidal field ramp to produce a very broad current profile [9]. B_T -ramps induce current in a broad region centered near $\rho \sim 0.7$, which is beyond the peak in the off-axis NBCD. A new experiment tested the stability and confinement of even broader current profiles by adding off-axis NBI to the B_T -ramp. The goal of the experiment was to use the ramp to evolve to an equilibrium with the same B_T , q_{\min} , and q_{95} obtained in a previous plasma without a ramp, but with a broader current profile, meaning lower l_i and/or higher ρ_{qmin} . After this time B_T was held constant and the obtainable β_N assessed. Figure 6 compares B_T , q_{\min} , l_i , and β_N for a control discharge without a ramp and one with a ramp. The minimum B_T was 1.4 T, and the maximum available co- I_p NBI power was ~ 10 MW. The high β_N phase of both plasmas was ultimately terminated by an $n=1$ tearing mode after q_{\min} evolved to less than 2. While q_{\min} was still greater than 2, the control plasma achieved an average $\beta_N \sim 3.5$ and the B_T -ramp plasma achieved $\beta_N \sim 3.8$ with $\sim 7\%$ less beam power, indicating improved confinement. However, during the high β_N phase, l_i was about the same for both plasmas. During the low β_N phase, the B_T -ramp lowered l_i to ~ 0.5 and helped to maintain high q_{\min} . But

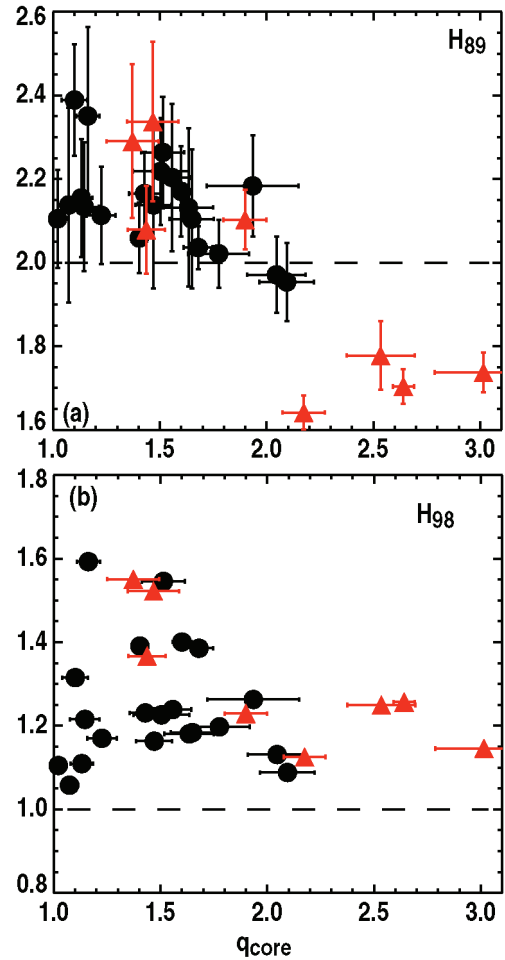


Fig. 5. (a,b) At the highest q_{\min} , thermal energy confinement matches $H_{98} \approx 1$ scaling, but total energy confinement falls below typical H-mode levels ($H_{89} < 2$). Circles are plasmas without off-axis NBI, and triangles are with off-axis NBI.

when q_{\min} passed through 4 and 3 an instability occurred that led to momentary collapses of rotation and β_N , and which rearranged current to arrive at a new, higher l_i equilibrium. Measurements of the radial magnetic flux associated with this mode show that the growth time was about 3 ms, which is roughly the wall time, and therefore this mode is an $n=1$ resistive wall mode. Future experiments with $l_i < 0.6$ will need to improve RWM stability possibly by optimizing the q -profile evolution, increasing rotational stabilization, and optimizing error field correction and RWM magnetic feedback stabilization.

To summarize so far, attempts to demonstrate DEMO-relevant $\beta_N \sim 5$ with $q_{\min} > 2$ using off-axis NBI on DIII-D have successfully broadened the current and pressure profiles and moved the predicted ideal-wall β_N limits closer to the goal. But at $q_{\min} > 2$ there is an apparent trade-off between high β_N potential and confinement, and specifically fast ion confinement. Further current profile optimization such as that done in the B_T -ramp experiments may eliminate this trade off.

3. Extension of High Performance, Quasi-Stationary Operation to $2\tau_R$

Off-axis neutral beam injection has proved very beneficial for achieving and sustaining discharges with modest q_{\min} (1.3-1.8) to optimize profiles for stability and sustain them for a suitable duration. Plasmas with modest q_{\min} and moderately high β_N ($\sim 3-4$) have been shown [10] on DIII-D to be promising candidates for long pulse or fully noninductive operation on an ITER-sized machine with projected fusion gain $Q \approx 5$. The same parameter range is relevant for the present FNSF-AT concept that is smaller than ITER and which only requires $Q \approx 4$ [1]. With q_{\min} and β_N in these ranges, tearing modes with $m/n=3/1$ and $2/1$ are the most common instabilities that can terminate good performance, and these are sensitive to the current profile and the proximity to the ideal-wall kink mode β_N limits [11]. The demonstrations of nearly or fully noninductive operation on DIII-D have been limited to durations less than $1\tau_R$, where $\tau_R(s) = 0.17$ [major radius (m)]/resistance ($\mu\Omega$) is the lowest order spatial eigenmode of the current evolution equation with a constant I_p constraint [12]. Furthermore some of these demonstrations are at β_N very close to predicted ideal MHD limits [13]. When operating close to ideal or resistive stability limits one must evaluate the evolution of the current profile to a stationary state over several τ_R to demonstrate both access to and robustness of the target equilibrium. Better still is to also adjust the profile such that the required operating pressure is farther from a stability limit.

With 5 MW off-axis NBI and ~ 3 MW of off-axis ECCD aimed between $\rho \sim 0.2-0.6$, nearly stationary plasmas were sustained for $2\tau_R = 3$ s, with $q_{\min} \approx 1.5$, $\beta_N \approx 3.5$, $f_{NI} = 70\%-75\%$, and equivalent fusion gain $G = \beta_N H_{89} / q_{95}^2 = 0.3$ that projects to $Q = 5$ in an ITER-size machine with the same double-null shape and aspect ratio (Fig. 7, red traces). This surpasses earlier results in similar plasmas lacking off-axis NBI and with less ECCD power that were

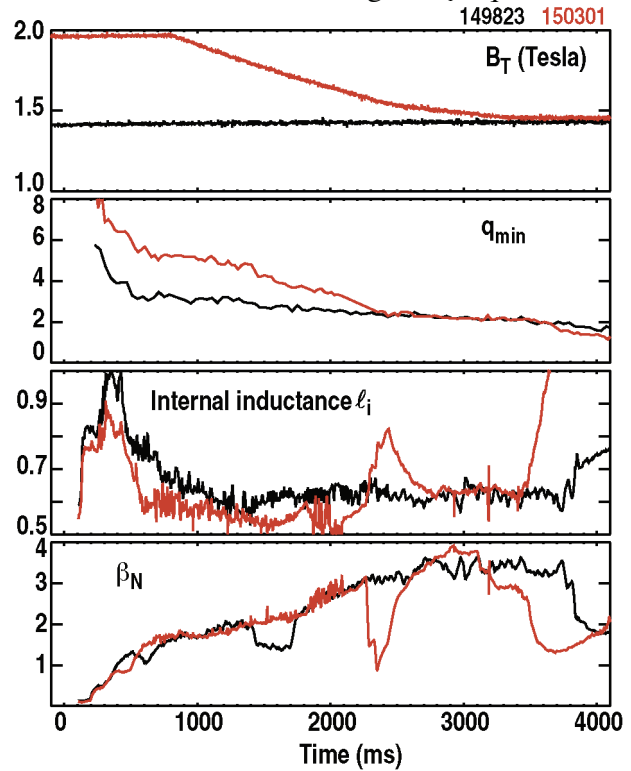


Fig 6. Exploration of very low l_i , high q_{\min} using additional off-axis current drive from a B_T -ramp found RWM instability at $l_i < \sim 0.6$ when $q_{\min} = 4$ or 3 (e.g., $t = 2200$ ms in the discharge with a B_T -ramp above).

stationary for $1\tau_R$ (Fig. 7, black traces). The duration of the high β_N phase was limited only by the available NBI energy, and low-order tearing modes were absent. The longer, $2\tau_R$ stable high β_N operation reduces the likelihood that the current profile will continue to evolve to one that is unstable to a tearing mode at $\beta_N=3.5$.

Measurements and stability analysis indicate that these plasmas exceed the $n=1$ no-wall limit β_N while being 12%–30% below the predicted $n=1$ ideal wall limit — thus leaving open the possibility for higher β_N and f_{NI} operation. Dynamic error field correction was required to minimize the resonant $n=1$ component of the error field. Active MHD spectroscopy measured a nearly linear plasma response amplitude with increasing β_N . The response increased faster above $\beta_N\approx 3.1$, indicating a proximity to the no-wall $n=1$ limit. This is in good agreement with a calculation of the $n=1$ no-wall limit between $\beta_N=3$ and 3.4 using the CORSICA and DCON codes shown in Fig. 8. The predicted ideal-wall $n=1$ β_N limit is typically between 4 and 5. A broad pressure profile with $P(0)/\langle P \rangle$ less than 3 likely contributes to the relatively high stability limits.

The noninductive current fraction f_{NI} is 70%–75% in these plasmas. This is based on two analysis techniques. The first technique determines the parallel electric field from the time derivatives of a series of equilibrium reconstructions, and then the inductive current density is formed by the product of $E_{||}$ and the neoclassical conductivity [14]. The second

technique uses the ONETWO transport code to compute the bootstrap current using the Sauter model [15], the NBCD using NUBEAM [16], and the ECCD using TORAY-GA [17]. The bootstrap fraction is ~45%–50%; the NBCD supplied ~20% of I_p , and ECCD supplies the remaining ~5%. The relatively small fraction of current driven by ECCD is due to the use of a lower toroidal field ($B_T=1.65$ T). In this case the EC resonance shifts to larger radius where it cannot drive as much current. The NBCD fraction is relatively low because of an inferred fast ion loss. A uniform anomalous fast ion diffusion of $1\text{ m}^2/\text{s}$ was needed in the NUBEAM calculation to eliminate an ~8% overestimate of the total stored energy relative to that determined by the equilibrium reconstruction.

This scenario is close to satisfying the requirements for an ITER steady state scenario, but further improvement is needed. While the equivalent fusion gain projects to the $Q=5$ ITER target, this demonstration was done in a double null diverted, high triangularity shape not possible in ITER. Such a shape is planned for FNSF-AT. No attempt has yet been made to produce this scenario with the heating and current drive tools presently anticipated for ITER.

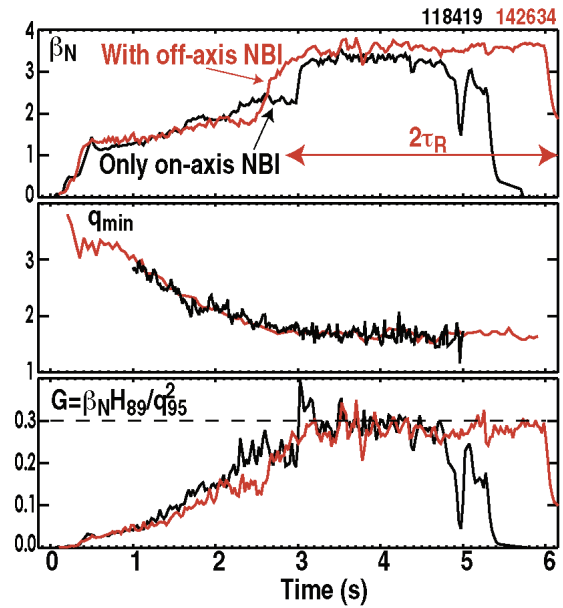


Fig. 7. High performance quasi-stationary plasma duration extended by using off-axis current drive.

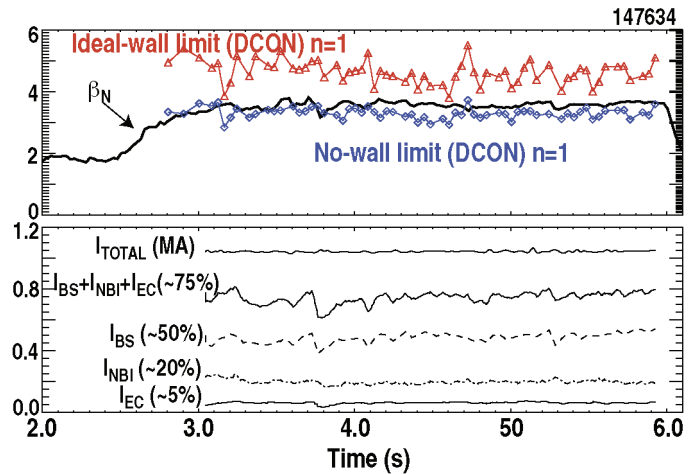


Fig. 8. Stability and current drive properties of the quasi-stationary plasma in Fig. 7.

To achieve $f_{\text{NI}}=100\%$ with the same q -profile, higher β_{N} is needed to increase the bootstrap current, and ideal stability analysis indicates that there is margin for this. In this scenario the choice of $B_{\text{T}}=1.65$ T to sustain a target $\beta_{\text{N}}=3.5$ with the available NBI power resulted in a low amount of current drive by EC. Further optimization to avoid fast ion loss, or increased NBI power would allow the use of higher B_{T} and thus higher EC and NB current drive levels. Of course any solution must avoid further peaking of the pressure and current profiles to maintain high stability limits.

4. Divertor Heat Flux Reduction in a High Performance, Quasi-stationary Plasma

Future tokamaks operating high power steady-state scenarios may experience divertor target damage if the power falling onto these surfaces is not reduced. One technique is to radiate away much of the power in the scrape off layer and private flux region by injecting a noble gas into this region, but this runs the risk of contaminating the main plasma and ruining the core scenario performance.

To test this technique, neon was injected into the private flux region of the lower divertor in the middle of quasi-stationary plasmas like those shown in Fig. 7. The time-averaged heat flux at the lower outer divertor target was measured by an IR camera, and the radiated power was measured by bolometry. Two neon injection rates were tested with the injection timed to match the beginning of the high β_{N} phase. 1.2 torr-l/s produced a relatively modest increase in total radiation and decrease in heat flux. Core plasma performance was unaffected. 5.2 torr-l/s resulted in a factor of ~ 2 higher total radiated power and a $\sim 40\%$ reduction in the divertor peak heat flux before main plasma performance was significantly degraded. The outer strike point did not detach, and before 4.75 s there was little change to the H-mode pedestal pressure compared to a case without neon. To accommodate the IR camera measurement, the location of the outer strike point in these plasmas was not at the optimal pumping location. Improvements to the IR camera field-of-view will allow a more optimal placement of the outer strike point closer to the pump and thus allow more effective methods of radiating away divertor power flow, e.g. “puff-and-pump”.

5. Discussion and Summary

Recent steady state scenario experiments on DIII-D with a new off-axis neutral beamline highlight a number of trade-offs related to the choice of q -profile. The off-axis beams improve access to $q_{\text{min}}>2$ where instabilities associated with a $q=2$ surface are not observed, and the calculated ideal-wall $n=1$ β_{N} limit is relatively high (≥ 4). But the lower energy confinement ($H_{89}<2$) observed in these plasmas prohibits testing these limits with the available heating power. Analysis indicates that in the plasmas generated so far with $q_{\text{min}}>2$, the thermal transport is about the same as and no better than standard H-mode transport, but fast ion confinement is apparently worse. This is a potential concern for any future device trying to obtain high β_{N} with a high q_{min} scenario and dominant beam or fusion alpha heating. Further study is needed to see if a high q_{min} scenario can be optimized to avoid this loss, possibly by adjusting the q and magnetic shear profiles. Nearly stationary

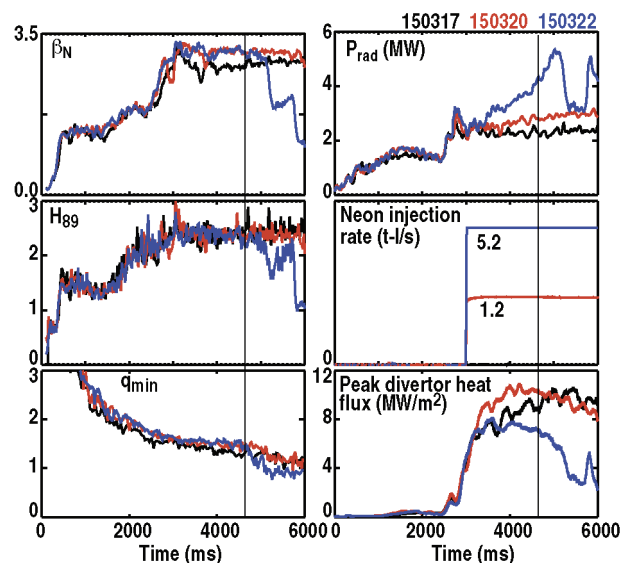


Fig. 9. A neon radiating mantle reduced the peak divertor heat flux by $\sim 40\%$ in a high performance plasma before evidence of main plasma degradation was observed.

plasmas with good confinement ($H_{89} > 2$), a high noninductive current fraction, and q_{\min} near 1.5 have been shown to be stable to tearing modes at $\beta_N = 3.5$. These have equivalent fusion gain performance that projects to $Q = 5$ in an ITER sized machine, and initial studies suggest that these plasmas can tolerate significant levels of neon injection useful for reducing divertor heat flux. More external and/or bootstrap current drive is needed to achieve $f_{NI} = 1$ and the latter implies higher β_N is needed. The calculated ideal wall $n=1$ kink β_N limit for this scenario is between 4 and 5, so there is margin for improvement. This work has demonstrated a scenario that is close to meeting the needs of ITER and FNSF-AT steady state scenarios, but even the predicted β_N limits of various experimental equilibria still fall short of what will be needed for robust $\beta_N = 5$ operation in a DEMO. Nevertheless progress has been made on DIII-D moving towards broader current and pressure profiles with increasingly higher predicted β_N limits, as shown in Fig. 10. Future steady state scenario experiments on DIII-D will address ITER more directly by increased fidelity to ITER constraints (e.g. shape, heating tools). Planned increases in the NBI and ECH power, and a possible second off-axis beamline will allow access to an even broader range of equilibria to further increase both the stability limit and obtainable β_N .

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References

- [1] CHAN, V.S., *et al.*, Fusion Sci. Technol. **57** (2010) 66
- [2] PARK, J.M., Bull. Am Phys. Soc. **56** (2011) 96.
- [3] FERRON, J.R., *et al.*, Phys. Plasmas **12** (2005) 056126.
- [4] TURNBULL, A.D., *et al.*, Phys. Rev. Lett. **74** (1995) 718.
- [5] YUSHMANOV, P.N., *et al.*, Nucl. Fusion **30** (1990) 1999.
- [6] ITER Physics Groups, Nucl. Fusion **39** (1999)
- [7] CHANG, Z. and CALLEN, J.D., Nucl. Fusion **34**, 1309 (1994).
- [8] LAO, L.L., *et al.*, Phys. Plasmas **3** (1996) 1951.
- [9] GAROFALO, A.M., *et al.*, Phys. Plasmas **13** (2006) 056110.
- [10] MURAKAMI, M., *et al.*, Phys. Plasmas **13** (2006) 056106.
- [11] BRENNAN, D.P., *et al.*, Phys. Plasmas **10** (2003) 1643.
- [12] MIKKELSEN, D.R., Phys. Fluids **B 1** (1989) 333.
- [13] HOLCOMB, C.T., *et al.*, Phys. Plasmas **16** (2009) 056116
- [14] FOREST, C.B., Phys. Rev. Lett. **73** (1994) 2444.
- [15] SAUTER, O., Phys. Plasmas **6** (1999) 2834.
- [16] PANKIN, A., Comput. Phys. Commun, **159** (2004) 157.
- [17] LIN LIU, Y.R., Phys. Plasmas **10** (2003) 4064.

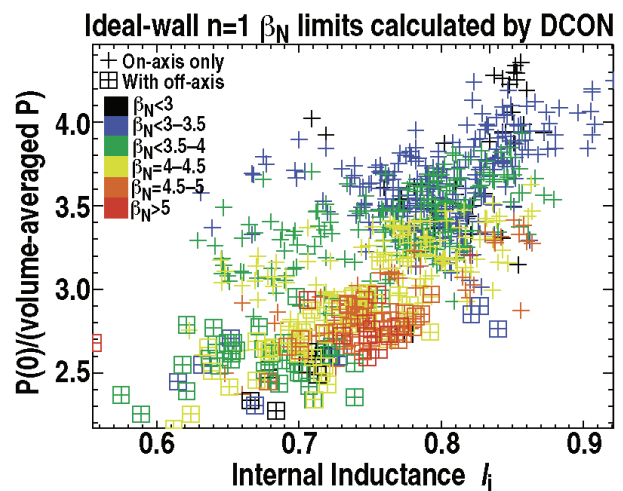


Fig. 10. Calculated ideal-wall $n=1$ β_N limits of experimental DIII-D equilibria show off-axis injection leads to broader profiles and higher β_N limits.