Neutral Beam Heating on the TCV Tokamak

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Abstract:

For over two decades, TCV has exploited high power density microwave heating from many steerable launchers for auxiliary heating and plasma control. To extend TCV's principal goal in developing the ITER physics basis, the first of two Neutral Heating Beams has been installed and operated in 2016. This paper presents some of the first results and observations of TCV operation with NBH together with a comparison of experimental kinetic profiles with predictions from the design study proposal.

1 Introduction

TCV's principal goal is to explore and develop the physics basis for ITER exploitation and to aid in the development of DEMO. The TCV tokamak was designed to examine the effects of magnetic confinement configuration shape on energy and particle confinement. To this end, it is equipped with a highly elongated vacuum vessel, a generous array of poloidal shaping coils and high resolution profile diagnostics of the main plasma kinetic parameters. To facilitate exploration of highly shaped configurations, each shaping coil is equipped with a separate power supply unit such that, within time scales dictated by plasma response and field penetration through the metal vessel, the magnetic configuration can be evolved at will. In the initial design, a combination of X2 and X3 ECH power was installed to provide precision auxiliary heating. For X2, this is delivered through lateral steerable mirrors and for X3 vertically, to compensate the absorption efficiency by maximising the resonant path through the plasma. With a nominal $< 1.5T$ toroidal field strength, X2 heating remains limited to electron densities $<\sim 4 \times 10^{19}/m^3$ and X3 to electron densities $<\sim 1 \times 10^{20}/m^3$. The extremely high ECH power density available has been exploited over the decades to not simply heat target plasmas but, using the high spatial localisation of X2 absorption, to tailor plasma current and temperature profiles and directly interact with deleterious plasma instabilities. (cf. many past papers at this conference series). Furthermore, the introduction of real-time (RT) magnetic equilibrium and plasma profile tracking coupled with RT reaction to the steering mirrors, has extended these capabilities into RT plasma discharge assistance where, for example, modes such as NTMs have been recognised, targeted, and reduced by precision X2 ECH and/or ECCD [1]. Several realms of plasma operation, pertinent to our stated goals of ITER and DEMO relevance, remained, however, only marginally accessible. H-mode discharges on TCV are only accessible to X2 ECH in the edge and only some configurations achieved sufficient electron temperature to obtain strong X3 heating, vital to the study of H-mode pedestals and ELMs. This, and many other higher density experiments, could only harness edge heating through X2 and core ohmic heating that, although extremely efficient, also affects the magnetic configuration and becomes less effective at higher temperatures. Divertor studies are a strong component of TCV's present and future research plan and, as for the H-mode discharges, high plasma densities are common and the presently available power can be insufficient to reach discharge parameters that are not strongly influenced by an unfavourable power balance. This is only exacerbated when puffing radiating impurities into TCV for divertor radiation experiments, or when the wall conditioning worsens, sometimes resulting in a plasma radiation collapse before, say, high power divertor detachment. Finally, when ECH power alone heats the plasma, electron-ion collisional equipartition decreases, resulting in extremely hot electrons with the ion temperature trailing, on TCV, by a factor that can exceed 30. [2]

For all these reasons, a phased upgrade program [A.Fasoli at this conference], is un-

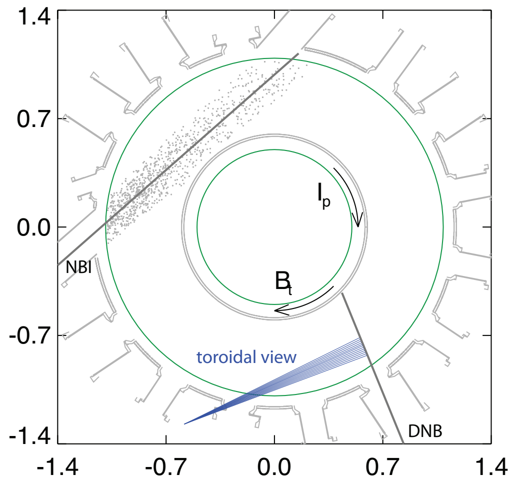

FIG. 1: Plan view of TCV showing Heating beam (NBI) and Diagnostic neutral beam (DNB) orientations. (Toroidal field (B_t) and Plasma Current (I_p) shown for co-injection)

derway on TCV that not only extends the power range of X2 and X3 heating but also introduces direct ion auxiliary heating using state of the art Neutral Beam Injection. In a first stage, a 1MW, tangentially launched, neutral beam of Hydrogen or Deuterium is installed and reported in this paper.

2 First Comparison with Scoping Predictions

The potential effect of NBH on TCV was scoped during a feasibility design phase. At that time, ion temperatures had never exceeded 1keV and, for high power microwave heating where the ion-electron collisional equilibration reduced, were often far lower. Emphasis was placed on obtaining NBH power access at X2-ECH compatible core plasma densities without excessive shine-through, and core heating at higher densities. It was concluded [3], that a relatively low energy ($\leq 30keV$) particle energy was necessary, with the possibility of injecting 15keV neutrals at the lowest TCV densities. Furthermore, a tangential injection, with the beam passing tangentially through the core plasma, $Fig.1$, would be required to obtain central heating for a wide range of plasma densities. With TCV's compact aspect ratio, (a=0.25m/R=0.88m), beam access between the toroidal field coils was extremely challenging. Two main conclusions resulted: consequential modifications to the vessel ports would be required and severe constraints on the beam divergence would be necessary. To respond to this need, two new tangential ports were grafted onto the existing vessel requiring in-vessel machining and welding and a slotted beam ion optics was developed to provide a beam with less vertical than horizontal divergence to fit between the coils. The divergence exigence proved accident prone [3, 4] so, to date, although the first beam can reliably provide $> 1MW$ of power for over 2s (∼the order of TCV's shot duration), operations were limited to 0.5−0.7MJ into a plasma discharge, mainly due to overheating worries in the beam to TCV duct[4].

TCV performance scoping with a single 1MW beam and up to 3MW of electron

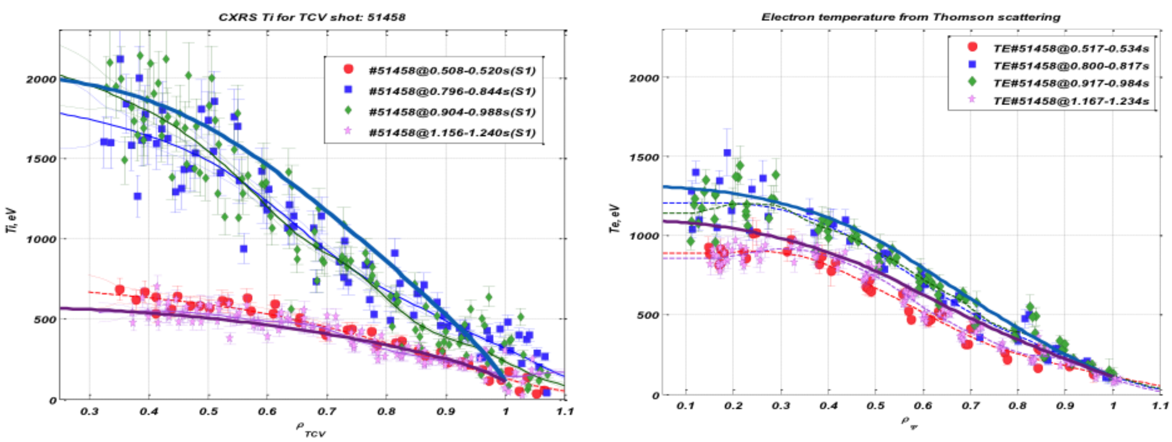

FIG. 2: Measured Ion Temperatures (left) and Electron Temperatures (right) at four times during the same discharge with thin lines drawn to aid the eye. Thick lines are comparable ASTRA modelling predictions (simulations for individual power sources in[3])

heating were modelled with the ASTRA code [5]. The simulations showed that $T_i(0) = T_e(0)$ ∼2.2 keV could be achieved with ∼0.8 MW of NBI and 1.3 MW X3-ECH. Furthermore, access to $T_i/T_e > 2$ should be attainable at increased (≥2MW) NBH or reduced ECH power. This simulation corresponded to plasma conditions of density $\sim 8 \times 10^{19}/m^3$ with 25keV neutral beam energy i.e. the kind of parameters associated with an ELMing

H-mode showing that, with the heating beam, ITER-relevant electron/ion temperature ratios and regimes can be explored. This effort used typical ion and electron confinement times from published scalings with low estimated losses from charge exchange and small fast ion orbit losses. The plasma parameters and confinement models (neoclassical) for this calculation are listed in [6]. With first full NBH injection into a plasma, we could compare these predictions with measurements. Fig.2 shows the measured plasma kinetic profiles for ions and electrons for one of these first TCV discharges with the full 1MW NBH injected power. Solid lines show the scoping results from [3, 6]. They are in excellent agreement, particularly if one considers that the ASTRA calculations only use the modelled and not the experimental plasma parameters. There is considerable experience in fusion research of ∼1MW injected into a machine with a ∼1m major radius that was used in the scoping. The agreement with the scoping values shows that the power is indeed reaching the plasma and that, for this relatively simple case, TCV performance is in line with well established knowledge. The remarkable agreement for both the electron and ion kinetic profiles indicates that the NBH power is indeed deposited, as modelled, in the plasma and that the simple loss assumptions in the scoping were adequate.

3 Ion Temperature and Toroidal Rotation

TCV has a long history in measuring intrinsic ion temperature and impurity rotation profiles using a diagnostic neutral beam (DNB, Fig.1) that is virtually torque and power injection free[7]. It was chosen to install the NBH ∼180° from the DNB to preserve their independence. Powerful NBH simultaneously provides strong ion heating and strong co-beam injection torque. Fig.3 and Fig.4 show examples of the ion temperature and toroidal rotation profiles for a TCV discharge with the full 1MW of NBH. These profiles

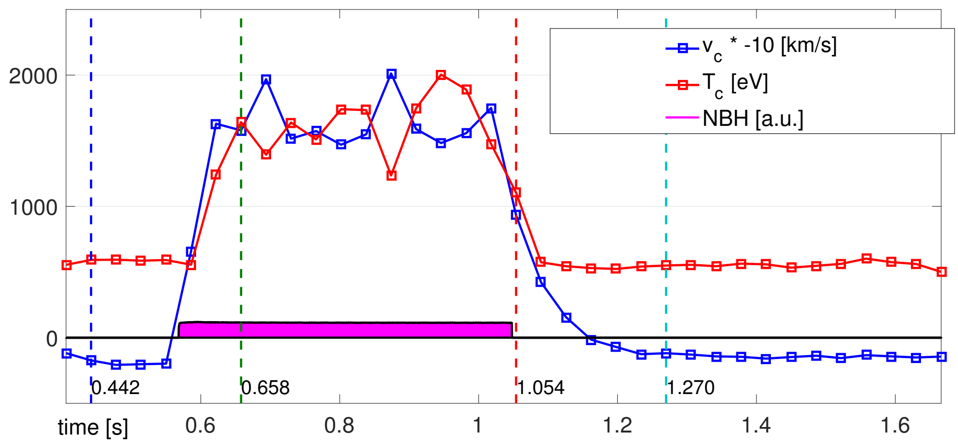

FIG. 3: Time history of core ion temperature and toroidal rotation(reversed in the figure). NBH duration is indicated as horizontal bar close to the axis. #51458, ($I_p = 300kA$)

should be compared to typical maximum measured toroidal core rotation of ∼30km/s [7] and ion temperatures that have a mostly parabolic shape peaking at ∼600eV. Since these measurements are performed with the DNB system, ion temperature relaxation to direct ion heating and intrinsic momentum balance state can now be measured directly. The core kinetic values take ∼100ms to reach a new steady state following NBH injection with T_i virtually tripled while the core rotation (shown inverted in the figure for compactness) starts counter-I_p but then accelerates strongly in the beam momentum injection direction to absolute values close to 6x the intrinsic rotation value. The combination of a diagnostic measurement system completely separated from a strong momentum injection source is quite unique and many exploratory experiments designed to disentangle the numerous possible contributions to the momentum balance become possible.

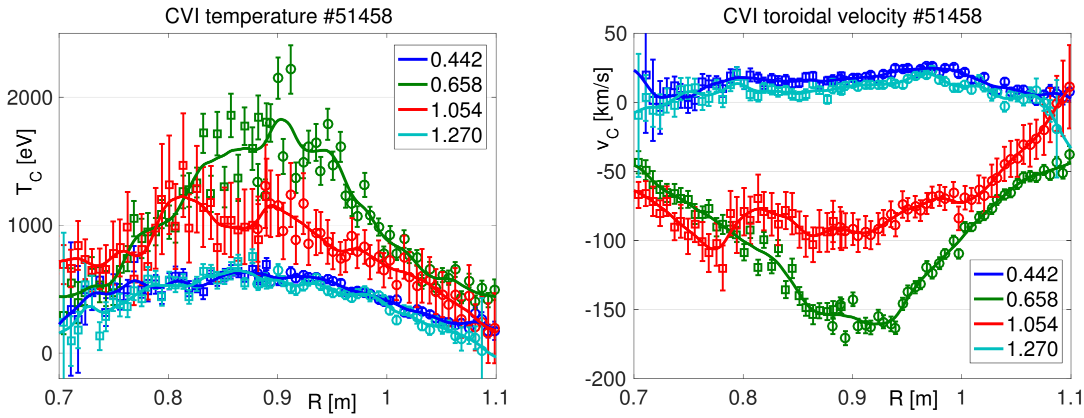

FIG. 4: Measured ion temperature (left) and toroidal rotation (right) profiles at the four times indicated above.

4 Sawteeth Stabilisation

Fast ions have been predicted and seen to affect MhD plasma modes such as the sawtooth (ST) [8, 9]. TCV has amassed considerable experience in the stabilisation and destabilisation of ST using high precision ECH power deposition inside or outside the $q = 1$ rational surface. On a medium density TCV plasma with q_{edge} ∼3 or lower, typical natural ST periods are only ∼1ms. By slowly scanning the ECH deposition across the region of the $q = 1$ surface, the ST can be routinely stabilised with periods extending to 30ms and above sometimes leading to NTM triggering [10]. Within the ST stability model [8, 9] fast particles will also stabilise ST, resulting in a longer ST-periods. Furthermore, co-beam injection (1MW at 25keV directly results in a 40kA current assuming complete ionisation with no losses), the plasma current density profile, and thus q profile, will be affected. $Fig.$5 shows an example of the ST period where ∼0.5MW of X2 ECH deposition is scanned from the HFS to the LFS towards the q=1 position. As the ST is stabilised, the ST-period lengthens creating longer and more powerful ST crashes [1, 10] and, in this case, a full plasma current disruption.

1MW of NBH power resulted in ST-periods up to 18ms and 0.5MW of NBH only resulted in ST-periods up to ∼8ms well demonstrating fast-ion ST stabilisation on TCV. It is interesting to note that, whereas for ECH heating a precise deposition location is imperative for ECH stabilisation, (mm accuracy on TCV is important), fast-ion ST

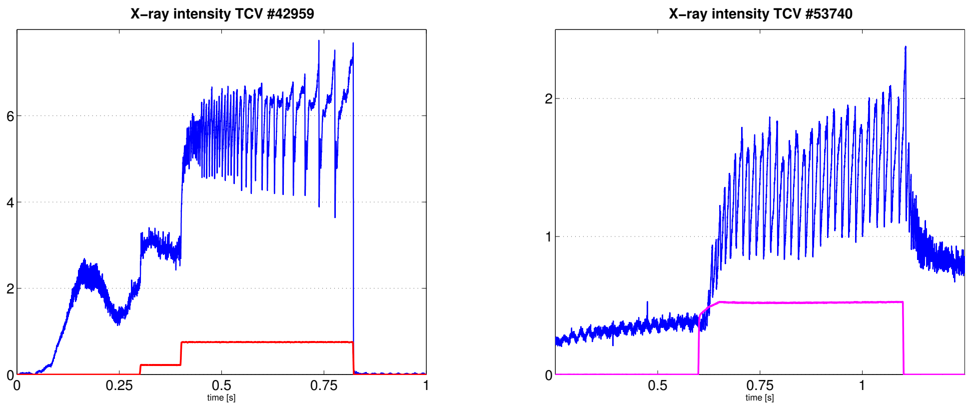

FIG. 5: Left: X-ray intensity showing ST period lengthening as X2 ECH swept to the plasma q=1 surface. Right: X-ray intensity with 1MW of NBJ injected into the plasma.

stabilisation using NBH only requires a fast ion population within the q=1 radius, which on TCV only requires that the beam reach the plasma. It is expected that the NBH and ECH ST-stabilisation techniques can be combined to examine the ST crash as a function of heating profile and/or ST-mixing radius [8]. To successfully heat near the q=1 position it is necessary to compensate for any change in the plasma q-profile, due to additional NBH heating changing the plasma temperature and thus plasma conductivity, and fast-ion current drive. As TCV does not have a direct q-profile measurement, and little experience in q-profile changes with NBH, several ECH X2 wide position scans were attempted for a range of NBH powers. The most promising result to date is shown in Fig.6 The result is somewhat perplexing. Although the X2 ECH power is sufficient to

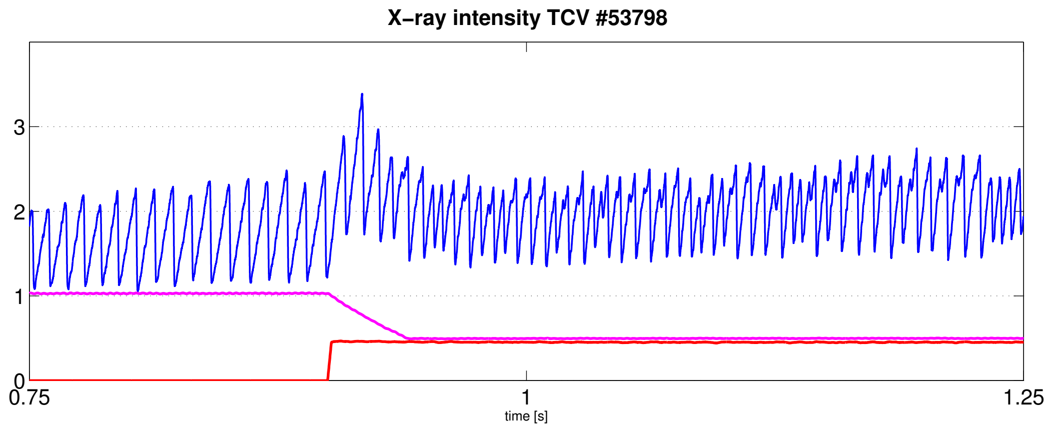

FIG. 6: X-ray intensity with 1MW of NBH until 0.85s where NBH power is reduced to 0.5 and a 0.5MW scan across the expected position of the q=1 surface ($I_p = 300kA$). At ∼1.2s there is a small increase in the ST period but the increase is modest.

result in long ST-periods and NBH should have, at least partially, already helped stabilise the ST, the measured ST-period does not change significantly over the entire scan and the obtained ST-period stabilisation is moderate. Further investigation of this situation is planned in the near future.

5 Co and Counter Plasma Current Injection

By moving the plasma in the vessel, TCV can trivially probe the effect of on and off-axis beam deposition into the same plasma configuration. The effect on the FIDA spectrum is shown by B. Geiger et. al. at this conference so here, the fast ion behaviour is demonstrated from the neutral emission spectra recorded by a compact neutral particle analyser (CNPA)[11]. This CNPA is placed nearly co-linearly $\sim 10°$ with the NBH injection after traversing the plasma and is mostly sensitive ions that have only suffered little scattering.

From the left hand of $Fig.7$, for co-current injection ($I_p = 240kA$), there is a increase in the emitted neutral signal below the highest relevant (23keV) neutral channel on the CNPA i.e. that closest to the 25keV neutral injection energy. This is the signature of a gradual slowing-down within the plasma, consistent with fast-ion confinement. The neutral flux with counter-injection, however, decreases below the beam energy until $\sim$15keV, indicating that these ions are lost more quickly than their confinement (on average). For lower energies, the signal again increases but at a much lower level than for co-current injection. The off-axis spectra are somewhere between these two cases with even stronger ion losses for off-axis, counter-current, beam injection. This is highlighted in the right hand $Fig.7$ where all the spectra are normalised to their values at 23keV (the energy of the closest NPA energy channel to the neutral beam energy). The effects described above appear more distinct with much stronger fast-ion losses for counter beam injection.

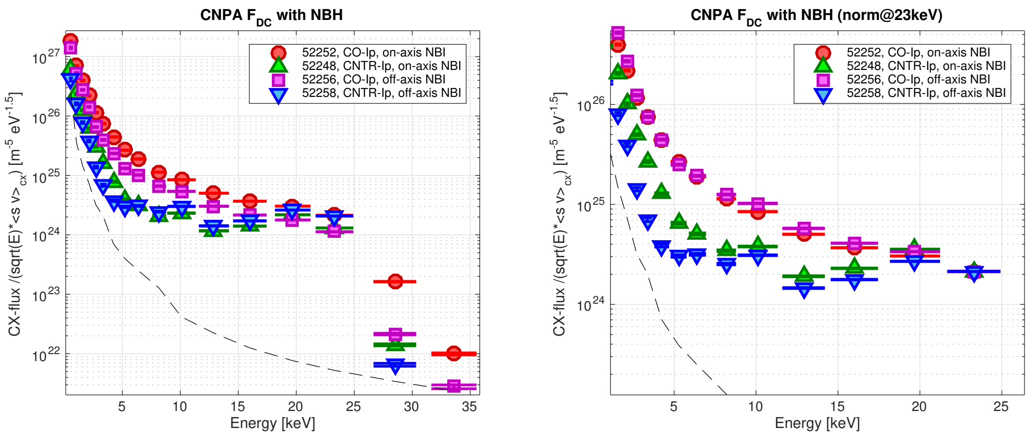

FIG. 7: Left: Measured NPA profiles for co-IP, counter-IP and on and off-axis beam injection (Bulk ion temperature can be estimated from the curve slopes ONLY at the lowest energies). Right: spectra normalised to the 23keV CNPA energy channel

6 Conclusions and Outlook

A high power neutral beam has been successfully installed and operated on TCV. From the first, the electron and ion kinetic profiles agreed well with scoping models implying the installation of the upgrades in ECH and Neutral Beam Heating (See A.Fasoli at this conference), can progress with confidence. A higher beam divergence than designed and initial design flaws in the beam coupling to TCV resulted in limited power allowed, at present, into the plasma. To protect TCV's integrity, several security systems are operated. Beam injection is halted if, the plasma current is too low (disruptions), the plasma density is too low (monitored by a FIR interferometer) or when a set of pyrometers on the beam duct entrance or tiles on opposite side of the machine to the injection axis, overheat. Despite these precautions, the duct between the injector and TCV has suffered major failure and increased internal protection is being considered [4]. Access to $T_e/T_i{\sim}1$ and the presence of a high fast ion fraction [B. Geiger at this conference] are being used to leverage TCV's expertise in ECH heating and MhD tailoring into plasma shape and confinement regimes that are directly ITER relevant. Recently, TCV has acquired the ability to change the toroidal field direction between discharges whilst retaining the ability to also reverse the plasma current between discharges. With the internal carbon protection tiles and diagnostic systems designed without a preferential toroidal direction, a wide range of shot to shot varied geometries and plasma current, fast ion current, ECCD directions are accessible. With a second, up to 50keV, beam planned, almost exactly tangentially opposite in direction to the installed beam, balanced momentum injection and faster ions will be added to TCV's possibilities. With the resulting increased electron temperature, for a wider range of plasma densities than that available to X2 heating alone, and increased ion temperatures using NBH (and this with extremely high ion and/or electron heating power densities), TCV will be able to explore many ITER relevant and, still further, reactor relevant regimes.

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