Neutral beam injection for fusion reactors: technological constraints versus functional requirements

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Abstract

In this paper we look at the technological constraints of neutral beam injection (NBI) systems and compare them with the functional requirements that NBI has in the various envisaged plasma scenarios for tokamak fusion reactors of the DEMO and fusion power plant (FPP) class. We show in particular that there is an intermediate beam energy range in which beamlines are unattractive because of size. Furthermore, for scenarios that consider NBI only for ion heating during the ramp-up and heat-to-burn phase we show that the use of beam energies in the range of 100 to 200 keV, which could be produced from positive ion beams with a much simpler system, could be an attractive option that should be further investigated.

Keywords: DEMO, neutral beam injection, ramp-up

(Some figures may appear in colour only in the online journal)

1. Introduction

Neutral beam injection (NBI) is an established heating method on many existing magnetic confinement fusion devices. Also ITER will have two or optionally three NBI injectors, each providing 16.7 MW of heating power with 1 MeV deuterium beams [1, 2]. NBI is also considered for DEMO and fusion power plant (FPP) class devices (see e.g. references [3, 4]). Besides plasma heating for ramp-up and burn control, NBI can provide additional functions such as current drive and inducing plasma rotation [5–7]. While ECRH has the unique property that it can drive current locally with surgical precision and is therefore the actuator of choice for NTM control and precise control of the current profile, NBI is able to provide good current drive efficiency at any radius [4]. NBI is therefore considered one of the preferred systems to provide the non-inductive and non-bootstrap fraction of the plasma current necessary to achieve steady-state tokamak operation. While current drive requires high beam energy because the slowing down path length of the circulating fast ions increases with beam energy, driving plasma rotation favors lower beam energies because the momentum input per beam power increases with decreasing kinetic energy. This indicates that the required specifications of an NBI system differ substantially depending on the function that it is meant to provide. In this article we will first investigate the technological constraints of both positive ion based and negative ion based NBI systems. We will then compare the required specifications for different NBI functions with these constraints and assess whether and how these requirements can be met. In the final section we explore the potential of positive-ion-based NBI to provide ion heating in the plasma ramp-up phase, motivated by the open question whether ECRH alone is sufficient to drive enough initial fusion power for the alpha particle heating to take over and bring the plasma to full fusion power [8].

2. Technical constraints of an NBI system

Most currently operated NBI systems create neutral hydrogen (H, D, or even T [9]) by neutralizing positive ion beams...
in a gas neutralizer (GN) via charge exchange. As the mean path length to ionization of the fast neutrals in the plasma is roughly proportional to the beam energy, bigger devices require accordingly higher beam energies. For ITER a beam energy of 1 MeV D was chosen [2]. As the neutralization yield of positive ions in the GN decreases with energy and becomes vanishingly small above 200 keV amu\(^{-1}\), such high beam energy requires the use of negative ion beams for which the neutralization yield remains approximately constant at around 55% above this energy. Such systems are referred to as NNBI as opposed to positive ion based systems, in contrast denoted as PNBI. There are several key differences between NNBI and PNBI systems:

- The creation of negative ion beams with high extracted current density is much more complicated than the generation of positive ions. Contemporary negative ion NBI sources use caesium evaporated into the source to reduce the work function of the plasma grid surface in order to enhance the surface production of D [10]. Caesium evaporation is usually continuous during source operation to obtain stable performance. The transport and distribution of Cs in the source is governed by a variety of mechanisms. It is also different comparing phases without and with beam extraction, as during the latter sputtering of Cs by positive ions back-accelerated into the ion source becomes important. Degrading or suboptimal caesium conditioning first shows up by an increase of the co-extracted electron current. These electrons are magnetically deflected and dumped on the extraction grid after acceleration by only about 10 kV. Nevertheless, the resulting heat load on the extraction grid limits the permissible electron-to-ion extraction ratio, e.g. for the ITER system to \(j_e/j_i < 1\) [10, 11]. The NBI system for ITER is currently still in development with a prototype system under construction and related radio-frequency ion source test stands being operated at IPP Garching [10] and RFX Padova [11] in order to establish reproducible and long-pulse-stable operating procedures for the ITER NBI ion sources, which is still a major challenge. On the contrary, PNBI sources are routinely and stably operated at their nominal parameters.

- The current density that can be accelerated from an NNBI ion source is approximately a factor of ten lower than that of PNBI sources. ITER’s challenging requirement is 200 A m\(^{-2}\) (accelerated current per net extraction area, i.e. aperture area), while for comparison the PNBI ion sources of ASDEX Upgrade’s injector 1 achieve 1980 A m\(^{-2}\), and this is not the ultimate limit.

- Both PNBI and NNBI systems usually have GNs. For PNBI the optimal neutralization yield, i.e. the fraction of fast neutrals in the beam leaving the neutralizer, is theoretically obtained for infinite gas target thickness and determined by the ratio of the energy dependent cross sections for charge exchange and ionization. For NNBI neutralization happens via stripping of the additional electron and the optimal neutralization yield of \(\approx 55\%\) is reached at an optimal target thickness before too many fast neutrals are ionized further to positive ions. The yield is practically constant at higher energies.

- Although NNBI also works at beam energies beyond 1 MeV, every further increase of the applied voltages increases the technological risk or requires alternative accelerator concepts [12] that differ substantially from the ITER beamlines. Hence there is a practical interest in avoiding excessively high beam energies.

For both types of systems the neutralization yield is the dominant limitation to the overall beamline efficiency. The relevant figure of merit is often called wall-plug efficiency, defined by the beam power injected into the torus divided by the total electrical power required to drive the system. Power losses occur due to stripping losses in the accelerator (NNBI only), geometrical beam transmission losses, reionization of a fraction of the beam as well as the limited, albeit high efficiency of power supplies as well as the power consumed by auxiliary systems. For the ITER NBI the sum of these losses amounts to approx. 45% [13]. While it may be possible to reduce all of these losses by some factor, the most relevant efficiency improvement could be achieved by addressing the losses associated with the non-neutralized beam fraction. There is a number of concepts for achieving this. The first two mentioned here, which are advanced neutralizers, can be applied to negative ions only. The third one works for negative and positive ion beams.

- Photoneutralization [12] that uses the photo-detachment of the negative ions’ additional electron has no theoretical yield limit, as the photon energy can be chosen such that it cannot produce positive ions. However, a yield close to unity requires high photon density in the overlap of the laser beam with the ion beam, which for reasonable laser power can only be achieved with optical cavities with very low losses. Such systems are currently far from being technologically ready for application and are therefore not considered as an available option in this paper.

- Plasma neutralization [14] is an improved version of gas neutralization in which the neutralizer gas is ionized to form a sufficiently dense plasma. In collisions with electrons or ions of this neutralizer plasma the cross sections for single stripping of the fast beam ion (D\(^+\) \(\rightarrow\) D\(^0\)) are larger than in collisions with neutral hydrogen, while the cross sections of D\(^+\) forming reactions (D\(^0\) \(\rightarrow\) D\(^+\)) do not increase by the same factor. Thus the fraction of neutral beam atoms at the exit of the neutralizer is increased. Achieving the required high ionization degree of the order of 10% and plasma density of the order of 10\(^{18}\) m\(^{-3}\) in a neutralizer volume of several m\(^3\) is a challenge in itself. All-around magnetic cusp confinement is therefore applied to minimize wall losses. In the beam driven plasma neutralizer (BDPN), first suggested by Surrey and Holmes [15], the plasma is generated by the passing beam itself, avoiding added system complexity. Predictions of the performance of the BDPN currently rely on a mostly zero-dimensional model by Surrey and Holmes [15] and Turner and Holmes [16], that has recently been extended by Starnella et al [17] to also include a rate
equation model for the ion-species composition and the effect of dissociative recombination of electrons with the molecular ions. We use this model in the following to estimate the neutralization efficiency achievable with limited system complexity.

- Residual ion energy recovery (ER) [18] is another option to reduce neutralizer power losses. As opposed to advanced neutralizers, this concept does not increase the neutral beam power for a given accelerated beam power as it does not increase the neutralization yield, but it reduces power losses by reclaiming a large fraction of the residual ions’ kinetic energy. The concept is simple. The two polarities of the residual ions (for PNBI only one) are deflected in opposite direction by a transverse magnetic (or electric) field and electrostatically decelerated before being collected on dump electrodes. The recovery of the energy from the ions that still have the polarity that they were extracted with is straight forward, as the collector electrode can be connected via a small bias power supply to the ion source potential. For the opposite polarity, i.e. positive residual ions for NNBI, it requires a more advanced technology, referred to as resonant converters or ER modules [18]. In particle tracking simulations we found that up to 95% of the kinetic energy can be recovered while still collecting all ions. We use this number to estimate the maximally achievable energy efficiency for the BDPN combined with ER.

In figures 1 and 2 we have estimated a number of quantities assuming an NBI beamline that delivers 25 MW of heating power in deuterium beams to the torus. In order to estimate these quantities we have to make some assumptions. We assume a beamline transmission including reionization losses downstream of the neutralizer of 0.8, like on ITER [13]. To take into account the losses in the accelerator and neutralizer (stripping, secondary electron acceleration and geometric losses) we assume that 1.2 times the current entering the neutralizer has to be accelerated, which corresponds to an optimistic 1.65 fold reduction of these losses compared with ITER. Such a reduction could possibly be achieved through more effective pumping of the volume between the grids, which could be achieved by increased grid transparency and improved lateral gas conductance, as outlined in reference [13]. For positive ions we assume only a factor 1.07 instead of 1.2, as these beams do not suffer stripping. We also count the RF power to the ion sources, for which we scale that of the ITER source for NNBI and the AUG RF sources for PNBI by the accelerated current. For other auxiliaries together we assume, in the absence of data available to us, a power that is equal to the RF power and therefore also scales with accelerated current. The efficiency of all power supplies is taken as 0.9. For simplicity we assume for PNBI that the beam has only atomic ions at full beam energy, neglecting the half and third energy components that come from the extracted molecular ions and typically constitute up to 30% of the beam power.

Figure 1 shows the maximum neutralization yield as a function of energy for PNBI with a GN (blue), NNBI with a GN (black), and NNBI with a BDPN as estimated with our revised BDPN model [17]. At low energies negative and positive ion beams have similar yields, but the yield for PNBI drops dramatically above 100 keV, while that of NNBI saturates at around 0.55, as calculated using the cross sections from Barnett et al [19]. With a BDPN the predicted efficiency is substantially higher and almost constant at 0.68 in the energy range between 300 and 2000 keV, limited by the achievable ionization degree. The non-smooth nature of this curve is due to a detail of how the model calculation is performed: as the extracted current density is assumed to be constant, the net beam size varies according to the extracted current; in order to constrain the distance between the cusp magnets on the entrance and exit sides the number of the bars containing these magnets is adapted to the beam size, leading to step-wise changes of the cusp loss area at certain energies. In figure 2 we consider only PNBI for energies below 300 keV and NNBI above. Figure 2(a) shows the extracted current necessary to deliver the requested power of 25 MW to the torus. For NNBI it is shown for a GN as well as a BDPN. The thin dashed line shows the injected (neutral) current for comparison, which for the assumed constant power falls as $1/E_{\text{beam}}$. Because of the dramatic decrease of the neutralization yield for PNBI the required accelerated current increases above $\approx 100$ keV and reaches more than 1500 A at 300 eV. At the same energy NNBI requires a factor of ten less accelerated current. The necessary accelerated current is also reflected in the cross-sectional source area shown in figure 2(b). To calculate this area for NNBI we have scaled the gross cross-sectional area of the ITER NBI source ($\approx 2 \text{ m}^2$) with the accelerated current. For PNBI we have divided this area by 10 as the extracted current...
density is ten times higher. At 300 keV this compensates the factor of ten difference in the neutralization yield and required extracted current and PNBI and NNBIs sources have practically the same size. With around 8 m² they are impractically big, making the beam energy region around 300 keV rather unattractive for high power beamlines with either beam polarity. It should also be pointed out, although not shown here, that with increasing source and neutralizer cross section the neutralizer gas flow also increases, necessitating much larger getter pumps in order to keep the background gas pressure at an acceptable level. For 25 MW NNBI at 300 keV we estimate >100 m² of required pump surface, about three times larger than in the ITER beamlines. The gas consumption of NBI also has implications for the fuel cycle, which we plan to address in a separate paper. We conclude that there is an energy range between about 200 and 450 keV in which the required source and beamline size would shrink and that owing to a higher current drive efficiency the required power would be lower. On the other hand, as mentioned in section 2, exceeding the already challenging 1 MV acceleration voltage of ITER comes with additional technological risk.

As a current drive system would be continuously operated, the electrical power needed to run the system is an economically important factor. For illustration, an FPP with the parameter set given in reference [21] (3.5 GW fusion power, major radius \( R = 9.28 \) m, bootstrap fraction \( f_{\text{bs}} = 0.62 \), average electron density \( \langle n_e \rangle = 8.5 \times 10^{19} \) m\(^{-3}\), NBCD efficiency \( 0.3 \times 10^{20} \) A m\(^{-2}\) W\(^{-1}\)) would reinvest

Another parameter that deserves consideration is the divergence of the beamlets that the beam is composed of. If all individual beamlets are steered such that they intersect at the end of the beamline, the required port size is determined by the local beamlet diameter which is determined by the beamlet divergence. The divergence of a 100 keV PNBI beam is approximately twice that of a 1 MeV NNBI beam: the ITER NBI beams are specified to have a beamlet divergence <7 mrad [20] while the divergence of the ASDEX Upgrade 93 keV beams has been measured as 14 mrad. Thus, if the beamline length is identical, either the port needs to be twice as large for the lower beam energy or the transmission losses will be higher. Tritium breeding could also be negatively affected, if more breeding blanket modules are affected due to the larger port size.

3. Considerations for the use of NBI for different functions

3.1. NBI for current drive

NBI can drive a current in the plasma that is carried by the neutral-beam-born fast ions as they circulate toroidally around the torus and are not fully shielded by the electrons. This requires tangential beam injection in co-current direction. Generally, neutral beam current drive (NBCD) profits from high beam energy as the slowing down path length increases with increasing fast ion energy, increasing the number of toroidal circulations and thereby the amplification of the injected current. A particular advantage of NBCD compared with wave-based current drive methods is that its efficiency remains high at larger deposition radius \( r \), as the higher fraction of trapped electrons at larger \( r \) reduces the electron shielding current. This makes NBCD particularly well-suited for off-axis current drive that allows one to maintain an elevated central \( q \) in advanced scenarios. Scenario simulations for DEMO machines such as FlexiDEMO [21] typically assume beam energies between 1000 and 2000 keV [7, 22] for high NBCD efficiency and calculate a required NBI power between \( \approx 100 \) and 200 MW to achieve fully non inductive current drive at a bootstrap fraction of about 60%.

Given the required beam energies NBCD can only be provided by an NNNB system. At least four of the discussed 25 MW beams or one to three accordingly bigger sources would be required to deliver the necessary power. Beam energies in excess of 1 MeV would have the advantage that the required source and beamline size would shrink and that owing to a higher current drive efficiency the required power would be lower. On the other hand, as mentioned in section 2, exceeding the already challenging 1 MV acceleration voltage of ITER comes with additional technological risk.

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**Figure 2.** Estimated characteristics of a 25 MW beamline as function of beam energy. Results for \( E_{\text{beam}} \leq 300 \) keV are for PNBI with GN and and those for \( E_{\text{beam}} \geq 300 \) keV are for NNBIs with GN or BDPN. (a): accelerated current (thick lines with symbols) and injected current (dashed line); (b): required ion source size \( A_{\text{source}} \); (c) wall-plug efficiency with and without ER as indicated.
roughly half the primary electrical power it produces into its NBCD system if this system had an ITER-like wall-plug efficiency around 25%. Therefore an NNBI system designed for current drive on a steady-state tokamak needs to exploit all available technologies to improve the wall-plug efficiency, such as the plasma neutralizer in combination with ER, or—if then available—photoneutralization.

3.2. NBI for burn control

Using NBI for burn control may require less power than current drive [23]. Depending on the chosen operational point NBI could either be operated on standby only to be rapidly available should the fusion rate drop due to a transient event, or NBI might be operated continuously at an intermediate power level to allow for control in both directions with this single actuator. In the latter case the wall plug efficiency would be similarly important as for NBCD. In the former case the unpredictable timing and in both cases the large power variations make stable Cs conditioning even more challenging. As burn control would best be achieved by central power deposition, the required beam energy would also be in the range accessible only by NNBI.

3.3. NBI for rotation

Scenarios have also been studied that use NBI for inducing toroidal plasma rotation in order to improve confinement [5, 6] or to give access to the quiescent H-mode [7]. Because the momentum and hence torque input per beam power increases with decreasing beam energy, these scenarios use beam energies in the range from as low as 35 keV [7] to \( \approx 100 \text{ keV} \) [5] and powers between 10 [5] and 76 MW [7]. This energy range is clearly only possible with PNBI. As this function would have to be provided continuously, wall-plug efficiency is also of importance here. PNBI has the potential to provide good wall-plug efficiency in this energy range, particularly when equipped with ER. The combination of very low beam energy (e.g. 35 keV) with high power (e.g. 76 MW) would lead to very large total source size, approx. 15 m\(^2\) with the above assumptions.

3.4. NBI for ion heating during ramp-up

The current base-line conceptual design for the European DEMO is a pulsed tokamak that does not aim at fully non-inductive current drive and hence does not consider NBI for this function. It currently explores ECRH only to provide all heating system functions. However, it has been observed in ECRH-only heated L-mode discharges on the W7-X stellarator and the ASDEX Upgrade tokamak alike that the ion temperature remains clamped below about 1.5 keV independent of ECRH power applied as turbulent ion transport increases with \( T_n/T_i \) [24]. Both direct ion heating and the transition to H mode can help to overcome this clamping. It is therefore not sufficiently clear yet whether or not direct bulk ion heating is required in the ramp-up phase of DEMO [8]. If required, both ICRF and NBI are candidates for this function and the technical complexity of the system will be one of the criteria for the decision between the two. In the following section we attempt to shed some light on the question in how far a PNBI system might be adequate for this limited application of NBI.

4. Positive-ion-based NBI for plasma ramp-up

If for a given DEMO scenario the only function of NBI is to bring up the ion temperature in the initial phase in order to kick-start fusion until alpha particle heating takes over, the technical complexity that a negative ion based NBI system would bring in appears disproportionate for its limited function and time of operation. It is therefore reasonable to explore in how far a much simpler positive ion based system operating at beam voltages below 100 keV amu\(^{-1}\) could sufficiently serve the limited purpose. For assisting the L–H transition one would expect that the deposition radius does not have a major impact as the threshold power was found to depend on the ion heat flux across the separatrix [25], independent of how or where inside the pedestal the heating power was deposited. For ion heating and the resulting \( T_n \) profiles the deposition profile will matter, however it may not be ultimately required to achieve ion heating profiles that are peaked in the center. The lower beam energy would have the additional advantage that the fraction of NBI heating power delivered to the bulk ions is higher than at higher beam energy. With a deuterium beam energy of 150 keV ion heating would exceed electron heating already at an electron temperature above 3.6 keV. This could be particularly interesting early in the current ramp-up phase, when the electron temperature is still relatively low.

In this section we do not attempt the modelling of plasma ramp-up with low energy beams. Instead, we restrict ourselves to give an indication of the heating profiles that one could obtain with lower-energy PNBI beams. To this end we have estimated the fast ion birth profiles for a range of low, PNBI-compatible beam energies. We assume deuterium injection. As PNBI systems extract \( D_2 \) and \( D_3 \) molecular ions along with the atomic \( D^+ \) ions, there are three components of the neutralized beam at \( E_0, E_0/2 \) and \( E_0/3 \), where the full beam energy \( E_0 \) is given by the elementary charge times the acceleration voltage. We take typical power fractions of 0.7 for the full, 0.2 for the half and 0.1 for the third energy component, although higher full energy power fractions may be achievable. We calculate a separate birth profile for each energy component. The birth profiles, defined as the number of ionized fast neutrals per \( \rho_{pol} \) interval \( [\rho_{pol}, \rho_{pol} + d\rho_{pol}] \) normalized to the plasma volume corresponding to this interval, generally give a good indication of the expected heating power profiles. In order to give a total birth profile of all energy components that is closer to the resulting heating power density profile, we weigh the birth profiles of the three energy components with their power fractions instead of their flux fractions. The profiles therefore represent fast ion power per unit volume.

Figure 3 shows the assumptions regarding the target plasma. We use a simplified equilibrium without x-point that emulates the DEMO 1 equilibrium presented in reference [26] in order to relate space coordinates to \( \rho_{pol} \) values and calculate flux surface volumes. The density profile in figure 3(b) was taken from the same reference and fit with a polynomial. Figure 3(c) shows the assumed beam geometry in a top
view. We have chosen relatively radial injection with a tangency radius of 4 m because more tangential injection would further increase the path length to the center while the mean free path to ionization is already low for the PNBI beams. The chosen geometry resembles beam geometries found on many tokamaks, such as ASDEX Upgrade. As the normalized gyroradius and the normalized banana width of the NBI fast ions are much smaller in DEMO than in medium-sized tokamaks for similar beam energies, one would not expect excessive fast ion losses due to the high fraction of trapped fast ions created with this injection geometry. For simplicity, the beam was not given a finite width in \( z \) direction, but extended from \( z = -0.3 \) m below to +0.3 m above the midplane. Injection is horizontal and the beam has no divergence.

The deposition calculation was carried out along multiple parallel sub-beams, each propagating along \( x \) at fixed \( z \). Fast ion deposition was then calculated in equally spaced \( \rho_{\text{pol}} \) intervals. The cross section for ionization by collisions with plasma ions and by charge exchange were taken from Janev et al.\cite{27}. Ionization by electrons was neglected as its cross section is always much smaller than the aforementioned. The effect of multi-step ionization and ionization due to impurities was modelled by multiplying the total cross section with a factor of 1.5, which should slightly overestimate the effective cross section and underestimated rather than overestimate the mean free path (see the calculations for \( Z_{\text{eff}} = 2 \), \( n_e = 10^{14} \text{ cm}^{-3} \) between 50 and 100 keV amu\(^{-1}\) by Janev et al.\cite{28}). The calculated birth profiles as defined above are shown in figure 4. In sub-figure (a) the beam energy is 150 keV, only slightly more than the 140 keV of the JET neutral beam injectors. The solid black line is the total profile, while the dashed lines show the contributions of the full, half, and third energy fractions. While the profile is clearly peaked at the edge, it does still not vanish in the center. The little second peak in the center is a consequence of the volume going to zero in the center. Figure 4(b) compares the birth profiles for different beam energies. At 100 keV NBI does not reach the center at all, while at 180 and 200 keV central deposition is substantially increased with respect to 150 keV beam energy.

When using NBI to assist ramp-up, one may also consider early heating before reaching current flat top. During the current ramp the plasma density is still lower, increasing beam penetration. In order to illustrate this effect, we have simply scaled the density profile shown in figure 3 with a factor of 0.5. Figure 4(c) compares the resulting birth profile (red line) with that for the full plasma density (black line), both at for 150 keV beam energy. As can be seen this should lead to a centrally peaked heating profile.

In order to estimate the heating power densities to ions and electrons we have multiplied the fast-ion birth profiles with the ion and electron heating fractions \( G_i(\rho_{\text{pol}}) \) and \( G_e(\rho_{\text{pol}}) \) and total injected NBI power \( P_{\text{NBI}} \) according to

\[
p_i(\rho_{\text{pol}}) = G_i(\rho_{\text{pol}}) b(\rho_{\text{pol}}) P_{\text{NBI}}, \tag{1}
\]

In this equation \( b(\rho_{\text{pol}}) \) is the total fast-ion birth profile in units of m\(^{-3}\) normalised such that

\[
\int_0^1 b(\rho_{\text{pol}}) \frac{dV(\rho_{\text{pol}})}{d\rho_{\text{pol}}} d\rho_{\text{pol}} = 1, \tag{2}
\]

where \( V(\rho_{\text{pol}}) \) is the volume inside the \( \rho_{\text{pol}} \) flux surface. According to Stix \cite{29} the local heating fractions to ions can be calculated by

\[
G_i = \frac{E_i}{E_\text{c}} \int_0^{E_\text{c}/E_i} \frac{dy}{1 + y^{3/2}} \quad \text{and} \quad G_e = 1 - G_i, \tag{3}
\]

where

\[
E_i = 14.8 T_e \left( \frac{A_j^{3/2}}{n_e} \sum n_j Z_j^2 \right)^{2/3}
\]

is the critical energy. Here \( A \) is the atomic mass number of the NBI ions and \( A_j \) and \( Z_j \) are the mass number and nuclear charge of plasma ion species \( j \). We have assumed a helium fraction of \( n_{\text{He}}/(n_0 + n_T + n_{\text{He}}) = 0.1 \) and a tungsten concentration of \( 1.7 \times 10^{-4} \) chosen such that \( Z_{\text{eff}} = 2.0 \). \( T_e \) and \( n_e \) were taken from the polynomial fit shown in figure 3(b). \( T_e, n_e, E_i, G_i, G_e \) are all functions of \( \rho_{\text{pol}} \). In this treatment we assume that the total power deposition is identical in shape as the fast-ion birth profile, neglecting radial broadening of the profile due to finite orbit width and scattering. However, the banana widths and passing orbit shifts even for 1 MeV deuterium ions are only of the order of 1/15 of the minor radius. The comparison of birth profiles and heating power profiles calculated with the METIS code in reference \cite{26} confirms that the shapes of these profiles are very similar. Using for benchmarking purposes the same tangency radius of 7.09 m and beam energy of 1 MeV as in reference \cite{26} we find generally good agreement with the power deposition profiles calculated by METIS, confirming the validity of our simplified approach.

Figure 5 shows the total heating power density profile and the profiles of the power to ions and electrons of a 150 keV beam for two cases. The solid lines are for the full flat-top plasma density and temperature. Owing to the high electron temperature and low NBI energy ion heating dominates at all radii, and inside \( \rho_{\text{pol}} = 0.6 \) the heating power goes exclusively to the ions. For the dashed lines we have emulated an early phase during ramp up by scaling the density profile again with a factor of 0.5 and the \( T_e \) profile with 0.1. In this low \( T_e \) case a 1 MeV beam would deliver more than 85% of its power to the electrons in the centre. In contrary, for the 150 keV beam the power is approximately equally shared between electrons and ions.

Summarizing, positive ion based NBI could have the potential to provide substantial ion heating in the plasma ramp-up phase, particularly when applied early while the electron density and temperature are still low. The penetration range of the low energy beams is increased at low density and at low electron temperature lower beam energy is beneficial as beams in the 1 MeV range would almost exclusively heat the electrons in this situation. As the system size for 100 to 200 keV PNBI and 500 to 1000 keV NNBi systems for the same NBI power is roughly the same (see figure 2), the required number of beamlines should also be identical.
Figure 3. Parameters assumed for the NBI deposition calculations. (a) Poloidal cross section of the simplified equilibrium the shown flux surfaces range from $\rho_{\text{pol}} = 1.0$ to 0.1 in steps of 0.1. (b) Flat top density and electron temperature profiles (polynomial fits). (c) NBI injection geometry with a tangency radius of 4 m.

Figure 4. Fast ion birth profiles for deuterium injection. All profiles are normalized to a peak value of 1. (a) For 150 keV beam energy; the dashed lines show the contributions of the full, half and third energy component. (b) For various beam energies. (c) For 150 keV and plasma density as in figure 3 (black) and for the same density profile scaled by a factor of 0.5 to mimic a lower-density transient ramp-up phase (red).

5. Summary and conclusions

We have investigated key figures of an NBI beamline for a given NBI power as a function of beam energy, particularly the parameters related to beamline size and efficiency. We have shown that reasonable wall plug efficiencies > 50% could be achievable with positive ion based systems in the energy range up to about 100 keV and for negative ion based systems at any energy up to > 2000 keV. In the energy range between approx. 200 and 450 keV both PNBI and NNBI sources—and with them the whole systems—become very large owing to the low neutralization yield for PNBI and low accelerated current density for NNBI combined with relatively low particle energy, making powerful NBI systems rather unattractive in this energy range.

We have compared these constraints with the requirements that NBI has to fulfil for different NBI functions on a DEMO or fusion power plant class device. The application of NBI as a steady-state tokamak current drive system and the use for burn control both require high beam energies in the 1 MeV range, accessible only to NNBI, and—at least for current drive—continuous operation. This means that the wall plug efficiency needs to be maximized, which would require advances neutralizers such as photoneutralization or the (beam driven) plasma neutralizer, the latter ideally in combination with residual ion ER. The application of NBI to induce toroidal rotation requires much lower beam energies below 100 keV, in the range that can only reasonably be accessed by PNBI, as NNBI injectors would be impractically large and unnecessarily complex. Whether a fusion reactor then needs either NNBI or PNBI or both depends on the list of NBI functions, which can be very different for different devices; for instance, the current European DEMO baseline design is a pulsed tokamak [30], that does not require fully non-inductive current drive, eliminating one reason why high-energy neutral beams and thus
an NNBI system would be required. Whether all heating functions can be sufficiently provided by other heating systems is currently being studied.

For the possibly only remaining function of an NBI system to provide ion-heating during ramp-up in order to kick-start alpha heating on an otherwise ECRH-only tokamak we propose to investigate the application of positive ion based NBI in scenario simulations. We have shown that, although the NBI birth profiles are peaked at the edge, core deposition is still present at 150 keV and above. In the 150–200 keV range such a system could also still achieve reasonable wall-plug efficiency, especially when ER is employed, but efficiency is anyway of less concern for this application. Owing to PNBI’s higher current density the system size could also be reasonable and comparable to a 1 MeV NNBI system of the same power. For this limited application during ramp-up PNBI would provide a much cheaper and technically simpler system, even if primarily chosen for heating during ramp up, could also be used for other necessary functions during the flat top phase. If these functions include current drive or central heating, e.g. for burn control, then of course negative ion based beamlines would be required.

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References

[1] Hemsworth R. et al 2009 Status of the ITER heating neutral beam system Nucl. Fusion 49 045006
[2] Singh M.J., Boilson D., Polevoi A.R., Oikawa T. and Mitteau R. 2017 Heating neutral beams for ITER: negative ion sources to tune fusion plasmas New J. Phys. 19 055004
[3] Liu L., Kessel C., Chan V., Guo Y., Chen J., Jian X., Mao S. and Ye M. (CFETR Physics Team) 2018 The time-dependent simulation of CFETR baseline steady-state scenarios Nucl. Fusion 58 096009
[4] Mikkelsen D.R., Kessel C.E., Poli F.M., Bertelli N. and Kim K. 2018 Survey of heating and current drive for K-DEMO Nucl. Fusion 58 036014
[5] Chen J., Jian X., Chan V.S., Li Z., Deng Z., Li G., Guo W., Shi N. and Chen X. (CFETR Physics Team) 2017 Self-consistent modeling of CFETR baseline scenarios for steady-state operation Plasma Phys. Control. Fusion 59 075005
[6] Jian X. et al 2017 Optimization of CFETR baseline performance by controlling rotation shear and pedestal collisionality through integrated modeling Nucl. Fusion 57 046012
[7] Vincenzi P., Artaud J.-F., Fable E., Giruzzi G., Siccino M. and Zohm H. 2021 Neutral beam injection for DEMO alternative scenarios Fusion Eng. Des. 163 112119
[8] Suárez López G., Fable E., Poli E., Tardini G. and Zohm H. 2020 On the feasibility of $T_e$ collisional heating and $T_i/T_e$ control on DEMO using ECRH Virtual 62nd Annual Meeting of the APS Division of Plasma Physics (9–13 November 2020)
[9] Hawryluk R.J. 1998 Results from deuterium–tritium tokamak confinement experiments Rev. Mod. Phys. 70 537
[10] Fantz U. et al 2017 Towards powerful negative ion beams at the test facility ELISE for the ITER and DEMO NBI systems Nucl. Fusion 57 116007
[11] Serianni G. et al 2020 First operation in SPIDER and the path to complete MITICIA Rev. Sci. Instrum. 91 023510
[12] Simonin A. et al 2016 Negative ion source development for a photoneutralization based neutral beam system for future fusion reactors New J. Phys. 18 125005
[13] Sonato P. et al 2017 Conceptual design of the DEMO neutral beam injectors: main developments and R & D achievements Nucl. Fusion 57 056026
[14] Berkner K.H., Pyle R.V., Savas S.E. and Stadler K.R. 1980 Plasma neutralizers for H− or D− beams 2nd Int. Symp. Production and Neutralization of Negative Ions and Beams (291 Upton, NY, 6–10 October 1980) (https://www.osti.gov/biblio/6790798)
[15] Surrey E. and Holmes A. 2013 The beam driven plasma neutralizer AIP Conf. Proc. 1515 532
[16] Turner I. and Holmes A.J.T. 2019 Model for a beam driven plasma neutraliser based on ITER beam geometry Fusion Eng. Des. 149 111327
[17] Starnella G., Hopf C. and Maya P. 2021 On suitable experiments for demonstrating the feasibility of the beam-driven plasma neutraliser for neutral beam injectors for fusion reactors Nucl. Fusion (submitted)
[18] McAdams R., Holmes A., Porton M., Benn A., Surrey E. and Jones T.T.C. 2013 Advanced energy recovery concepts for negative ion beamlines in fusion power plants AIP Conf. Proc. 1515 559
[19] Barnett C.F., Hunter H.T., Fitzpatrick M.I., Alvarez I., Cisneros C. and Phaneuf R.A. 1990 Atomic data for fusion, Volume 1: collisions of H, H2, He and Li atoms and ions with atoms and molecules ORNL-6086 Oak Ridge National Laboratory
[20] Hemsworth R.S. et al 2017 Overview of the design of the ITER heating neutral beam injectors New J. Phys. 19 025005
[21] Zohm H., Triebel F., Bie1 W., Fable E., Kemp R., Lux H., Siccino M. and Wenninger R. 2017 A stepladder approach to a tokamak fusion power plant Nucl. Fusion 57 086002
[22] Giruzzi G. et al 2015 Modelling of pulsed and steady-state DEMO scenarios Nucl. Fusion 55 073002
[23] Wenninger R. et al 2017 The physics and technology basis entering european system code studies for DEMO Nucl. Fusion 57 016011
[24] Beurskens M. et al (The Wendelstein 7-X team, The ASDEX Upgrade Team, The MSTI Team) 2021 Confinement in electron heated plasmas in Wendelstein 7-X and ASDEX Upgrade; necessity to control turbulent transport Virtual
[25] Plank U. et al 2020 H-mode power threshold studies in mixed ion species plasmas at ASDEX Upgrade Nucl. Fusion 60 074001

[26] Vincenzi P., Agostinetti P., Artaud J., Bolzonella T., Kurki-Suonio T., Mattei M., Vallar M. and Varje J. 2021 Optimization-oriented modelling of neutral beam injection for EU pulsed DEMO Plasma Phys. Control. Fusion 63 065014

[27] Janev R.K., Reiter D. and Samm U. 2003 Collision Processes in Low-Temperature Hydrogen Plasmas (Jül-4105: Forschungszentrum Jülich)

[28] Janev R.K., Boley C.D. and Post D.E. 1989 Penetration of energetic neutral beams into fusion plasmas Nucl. Fusion 29 2125

[29] Stix T.H. 1972 Heating of toroidal plasmas by neutral injection Plasma Phys. 14 367–84

[30] Federici G. et al 2019 Overview of the DEMO staged design approach in Europe Nucl. Fusion 59 066013