PAPER

Heating neutral beams for ITER: negative ion sources to tune fusion plasmas

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Abstract

Neutral beam injection (NBI) based on a negative ion source is one of the basic heating and current drive systems designed for ITER required to reach its goals of the operation with high fusion power, \( P_{\text{fus}} \sim 500 \text{ MW} \) with fusion gain, \( Q = 10 \) for 400 s in a baseline scenario, and \( P_{\text{fus}} > 250 \text{ MW} \), \( Q = 5 \) operation for 3600 s in an advanced scenario. A total power of 33 MW from the two heating neutral beam (HNB) injectors is envisaged in the present scenario. The scope of the present paper is to provide an overview of the main aspects of the interaction of the HNBs with the ITER plasma. Various operational scenarios with different mixtures of the main ion species, He, H, DD and DT, foreseen at different phases of the ITER operation are considered.

1. Introduction

High fusion power, \( P_{\text{fus}} \sim n_D n_T T_i^{3/2} (\alpha \sim 3 - 1.5) \) [1] requires operation with high fuel densities and temperatures. For ITER parameters, major radius \( R = 6.2 \text{ m} \), minor radius, \( a = 2 \text{ m} \), plasma volume, \( V \sim 800 \text{ m}^3 \), the baseline target parameters can be reached at \( n_D + n_T \sim 10^{20} \text{ m}^{-3} \). \( n_D \) and \( n_T \) refer to the deuterium and tritium ion densities and average ion temperature, \( \langle T_i \rangle \sim 20 \text{ keV} \). Ohmic heating alone is far insufficient to heat plasma to the required temperatures due to the decrease in plasma resistivity with temperature and the corresponding reduction in fusion power. The ITER design foresees three auxiliary heating systems required to achieve \( Q = 10 \), \( P_{\text{fus}} \sim 500 \text{ MW} \) operation.

The systems include two heating neutral beam (HNB) [2] injectors each capable of delivering 16.5 MW, 20 MW of electron cyclotron resonance heating (ECRH) [3] and 20 MW of ion cyclotron resonance heating (ICRH) [4]. Apart from heating the ITER plasma, each of the systems is foreseen to play specific roles to achieve the desired goals. Future upgrades to each of these systems are possible, allowing for a combination of either an additional HNB injector, an extra 20 MW of ECRH or 20 MW of ICRH. These systems have been designed while taking future upgrades into consideration. Table 1 summarises these systems in terms of power deliverables and the function each system is expected to perform during the different operation phases.

In addition to the above systems is a diagnostic neutral beam line (DNB) [5], which shares the port with the HNB-1 and delivers 100 keV H beams to the machine. The DNB is a probe beam to be used to diagnose the ion temperature and the He ash content in the plasma core using the charge exchange recombination spectroscopy (CXRS) technique. The capability of each of the four systems for the various intended uses has been demonstrated on tokamaks worldwide. This paper however concentrates on describing the anticipated role of only the HNB in the ITER plasmas for the various scenarios and phases of the ITER machine listed in Table 1.

Over the years as tokamaks have grown in size and density, neutral beams also have gone through considerable phases of development. In the past a large number of injectors around the world used positive ion sources to produce accelerated ion beams which were subsequently neutralised using gas neutralisers in the range of energies, \( E_{\text{NBI}} < 120 \text{ keV} \), where such neutralisation is sufficiently efficient. However with an increasing demand on the beam energy due to the increase in the plasma size and density, injectors based on positive ion sources are less efficient because of the steep reduction in neutralisation cross sections resulting in...
reactor size ITER-like machines penetration of the NBI to the plasma centre requires NBI energies of increase in plasma size and density, reduced power deposited by the beams in the plasma. Note that the NBI penetration depth drops with the $A_{\text{NBI}}$ corresponds to the mass of the neutral beam species used, $n'$ corresponds to the plasma density and $a'$ corresponds to the minor radius. Thus, for the reactor size ITER-like machines penetration of the NBI to the plasma centre requires NBI energies of $E_{\text{NBI}} \sim 1 \text{ MeV}$ (0.5–1 MeV/amu). This limitation requires injectors based on negative ion sources, for which the neutralisation efficiencies are as large as 60% at the energies of interest. It may be noted that the baseline negative ion sources for ITER are the multidriver RF based sources developed by IPP Garching [6].

The simulation carried out for the plasma parameters expected in ITER show that the energies, $E_{\text{NBI}} \sim 1 \text{ MeV}$ are sufficient for the NBI to penetrate to the plasma centre and be trapped by plasma due to multi-step ionisation processes driving the plasma current and producing plasma rotation due to beam trapping and the subsequent coupling of energy and momentum to the plasma [7]. The use of NBI heating in ITER will be subject to the restrictions associated with the localised deposition of non-thermalized NBI ions on the plasma facing components (PFCs) which may reduce their lifetime in a similar qualitative fashion to today’s experiments, although quantitatively different due to the higher NBI energy, long pulse operation requiring water-cooled PFCs, larger plasma dimensions, etc. In addition the high energy of the NBI ($E_{\text{NBI}} \sim 1 \text{ MeV}$ for D beams) leads to the fast ions originating from the NBI to be super-Alfvenic, as is the case in spherical tokamaks such as MAST [8] which increases the risk of excitation of Alfvenic eigenmodes [9] with a possible reduction in NBI heating and current drive efficiency and increased loss of fast ions onto PFCs. In order to reduce the localised deposition of non-thermalized NBI ions onto PFCs, the following design features and operational strategies have been adopted for ITER.

- Loss of fast NBI ions due to ripple loss. This issue was identified as leading to significant localised losses for the $Q = 5$ steady-state scenario due to the relatively low plasma current ($I_p = 9 \text{ MA}$) and large banana orbits of the 1 MeV ions for an intrinsic ripple of 1.16% associated with the 18 TF coils in ITER. The value of this ripple has been reduced by the use of ferromagnetic inserts down to 0.3% in most of the plasma toroidal perimeter, except where the NBI ports are located in which it increases to 0.55% [10]. The target value of the ripple reduction was driven by consideration of H-mode performance rather than localised fast particle losses [11] and at these levels of ripple localised fast particles losses from NBI ions trapped in the ripple wells are negligible even for the 9 MA $Q = 5$ scenario [12].

- Loss of fast ions due to the excitation of Alfvenic eigenmodes. This is determined by the value of the fast ion population pressure gradient across the magnetic surfaces. To decrease the risk of triggering these instabilities the NBI particle deposition profile can be broadened by steering the sources up or down with respect to the tilted beam axis so that the overlap of both deposition profiles is minimised (see figures 1(a), (b)).

- Shine-through loads. The NBI configuration in ITER is nearly tangential to the magnetic axis, with a tangency radius of $R_b = 5.3 \text{ m}$ and directed in the same direction as the toroidal magnetic field and plasma current (figures 1(a) and (b)). Such configuration minimises the orbital loss of fast ions caused by toroidal effects and minimises shine-through loads so that the NBI energy coupling to plasma is high (>95%) for a large range of plasma conditions. Despite this, the high energy of the NBI ions requires that the plasma density exceeds a minimum value of $2.2\text{–}4.4 \times 10^{19} \text{ m}^{-3}$ (depending on NBI isotope (H or D) and plasma species (H, He, DT)) for full power unrestricted application of NBI heating in ITER (i.e. 16.5 MW per NBI injector and long heating pulses (>100 s long). The implications of this limit for NBI operation in ITER on the basis of the present simulation results as well as the remaining uncertainties related to the evaluation will be described in this paper. In this respect it should be noted that the ITER simulations with NBI heating are performed for energies

| System | 1st Plasma | H-He/ DD/DT | Upgraded* (Potentially) | Function |
|--------|------------|-------------|--------------------------|----------|
| NB     | 0          | 33          | 50                       | Heating, current drive, current profile tailoring, plasma rotation |
| IC     | 0          | 20          | 40                       | Heating, Impurity control, MHD control |
| EC     | 6.7        | 20          | 40                       | Assist of plasma breakdown, heating, current drive, current profile tailoring, NTM stabilisation, ST control, impurity control |

* The maximum power injected in the vacuum vessel simultaneously by all upgraded systems is limited to 110 MW. MHD: magnetohydrodynamic; NTM: neoclassical tearing mode; ST: sawtooth
∼1 MeV, much higher than ever used in the real experiments with NBIs based on negative ion sources, \((E_{\text{NBI}} \sim 350 \text{ keV})\), and make use of NBI stopping cross sections derived theoretically \([13]\) but not checked against experiment at the level of energy foreseen for ITER. Therefore the verification of the validity of such cross sections through dedicated experiments to characterise NBI deposition profiles, driven current and shine-through losses is an important part of the activities related to the NBI commissioning with plasma considered in the ITER Research Plan \([7]\).

Another area where the neutral beams have been used effectively is the current drive, which is basically producing current parallel to the magnetic field in a tokamak plasma which replaces part of the current driven inductively. Note that in some cases the net current driven by all ITER auxiliary heating and current drive, CD, systems together with bootstrap current can fully replace the inductive current making possible the steady-state operation \([14]\). In the case of neutral beams it is the parallel velocity component of the suprathermal ion that is responsible for the toroidal current. The current is however strongly influenced by the presence of ions and electrons in the trapped banana orbits in tokamak plasmas of low collisionality. Part of the ion current is screened by the electrons reducing the net driven current, \(I_{\text{NBCD}} < I_{\text{NBI}}\). Despite this the NBI has the maximum current drive efficiency, \(I_{\text{CD}}/P_{\text{aux}}\) among all the ITER heating and CD systems. The NB driven current density has a radial profile which depends on the beam energy, the deposition profile and the injection angle, plasma density and temperature profiles. It is possible to optimise these parameters to some extent within the requirements for the ITER operational scenarios in order to achieve the desired NB driven current profile shape and to maximise the global current drive efficiency related to the injected power. In principle, the highest CD efficiency can be achieved by operating at the highest electron temperature and lowest density and by choosing the beam energy closest to the critical energy at which the effects of collisions with the thermal ions and electrons are approximately equal. In practice, however, the choice of beam energy is determined by the beam penetration requirements. It should be noted that although for a given beam species the NBI energy in ITER can be reduced by 20%, such reduction is accompanied by ∼50% reduction of the NBI power \((P_{\text{NBI}} \sim E_{\text{NBI}}^{2.5})\) as the perveance has to be matched for optimal beam optics.

Figure 1. (a) Top and (b) side view showing the HNB beam transmission at ITER.
Neutral beams also induce plasma rotation in ITER plasma. The rotation occurs because of the transfer of angular momentum of the ionised fast particles injected by NBI into the plasma particles. The NBI is the only external source of toroidal momentum in ITER. Plasma rotation in ITER produced by NB momentum input has been computed for various beam energies and target plasma densities. The results are described in section 2.

This paper is divided into two major sections. The first section is dedicated to a brief description of the important parameters required for neutral beams at ITER, and are covered in more detail in the paper by Hemsworth et al [15] on this issue. A comparison of heating and diagnostic beam parameters is given in this section only to provide the reader with an idea of the varied beam requirements for the two systems. The interaction of the heating beams with the ITER plasma is described in the second section of this paper and covers the aspects related to the use of NBIs to access and heat H-mode plasmas, current drive, plasma heating and plasma rotation. Some of the aspects have already been addressed in detail in [16]. In this paper we re-analyse issues related to the shine-through loads taking into account the detailed shape of the first wall panels and blanket shield modules on which these loads are deposited. The interaction of the diagnostic beam with the ITER plasmas however is not within the scope of this paper.

### 2. Heating and diagnostic beam lines

Two HNB and one DNB beam lines are planned for the H/He phase of the operation of ITER and are located in the NB cell of the tokamak complex as shown in figure 2. As seen in the figure the DNB has a crossover with the HNB-1 in the duct region.

HNBs have been described in detail in another contribution to this issue [15]. Table 2 summarises the beam parameter requirements for the HNB and DNB systems at ITER. Apart from the substantial differences in terms of the beam energy, current and delivered power requirements, the HNB and DNB beam lines in principle have the same layout and the beam line components are also similar. The design choice is made considering the

![Figure 2. Layout of the two HNBs and one DNB beam lines in the ITER NB cell.](image)

Table 2. Summary of the beam requirements for the two systems.

| Parameter | HNB | DNB |
|-----------|-----|-----|
| Injected power (H/He, DD/DT phase) (MW) | 16.7/beam line | 2 |
| Beam energy/species (H/He Phase) (MeV) | 0.87/H | 0.1/H |
| Beam energy/species (DD/DT Phase) (MeV) | 1/D | 0.1/H |
| Accelerated current (H/He Phase) (A) | 46/H | 60/H |
| Accelerated current (DD/DT Phase) (A) | 40/D | 60/H |
| Beamlet divergences (H/He, DD/DT phases) (mrad) | 3–7 | 3–7 |
| Pulse length/duty cycle (s) | 3600/25% | 3 s ON, 20 s OFF, 5 Hz |
| Total time of beam operation (s) | 2 × 10⁷ | 2 × 10⁷ |
| Horizontal focusing beamlet/beam group (m) | 7.2/25.5 | 20.67/20.67 |
| Vertical focusing beamlet/beam group (m) | Infinity/25.5° | 20.67/20.67 |
| NBI axis vertical inclination angle (mrad) | −49.2 | 15.5 |
| Beam axis vertical tilting angle (mrad) | +/−10 | 0 |
| Tangency radius (m) | 5.3 | |

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functionality of each of the components. On the other hand the choice of materials and manufacturing technologies is based on the consideration that ITER shall have a hostile neutron radiation environment requiring thereby remote maintenance of the components and that the components should survive the ITER lifetime of 20 years.

Figure 3 shows the layout for the HNB beam line. The beam line is a combination of the vacuum vessel housing the beam line components (BLCs), the exit scraper (ES), followed by a series of front end components (FEC) comprising a fast shutter (FS), absolute valve (AV), drift duct liner (DDL), vacuum vessel suppression system (VVPSS) box, connecting duct liner (CDL) with a liner and the duct liner (DL) made up of several modules. The end of the DL couples to the VV port. The BLCs consist of an ion source (IS) [17], a neutraliser (N), an electrostatic residual ion dump (ERID) and a calorimeter (C) [18]. The vacuum vessel for the HNB is a combination of the beam source vessel, BSV, and the beam line vessel (BLV). Coupled to the BSV is the high voltage bushing (HVB) from the top flange in the case of HNB. In the DNB case the bushing is mounted horizontally from the rear flange of the vacuum vessel. All the gas, hydraulic and electrical connections to the various components of the beam source from the high voltage deck transit through the bushing. The walls of the BLV are lined with cryopumps with a pumping speed of $10^6$ l s$^{-1}$. The ion beams produced and accelerated by the IS should not be deflected by the magnetic fields from the tokamak, the values of which can vary depending on the mode of operation of the machine. This is achieved by having a combination of passive magnetic shield (PMS) and a set of active correction compensation coils (ACCC). The PMS for the HNB and DNB beam lines not only provides the desired field compensation but also acts as a neutron shield. In the case of the HNB the PMS consists of two 75 mm thick layers of low C steel with a 100 mm thick polyethylene (PE) layer sandwiched between the two. This arrangement is further encased in a 24 mm thick lead layer for gamma shielding. On the other hand the PMS for the DNB consists of three 50 mm thick layers of low C steel separated by a 25 mm airgap. The set of ACCC is arranged in a distributed fashion outside the PMS for both the HNB (figure 3) and the DNB.

The beams with specifications listed in table 2 are produced with the help of the beam source consisting of a cesiated ion source coupled to an extractor and accelerator system. The 1 MV D or the 870 keV H beams are produced by eight driver RF based negative beam sources which use a seven grid five stage accelerator system whereas the 100 keV H beams for the DNB use a source similar to the HNB but with a three grid system with one acceleration stage. The details of the source design and the accelerators can be found in [15, 17, 19, 20].

Depending on the beam aiming requirements which are dependent on the plasma parameters for shine-through reasons and to mitigate Alfven eigenmodes (TAEs) [9] the HNB source can be tilted in the vertical direction with respect to the neutral beam axis within $\pm 1$ mrad. X in this case refers to any value between 0 and 9 mrad. It may be noted that the neutral beam axis is tilted with respect to the machine
centre line by 49.2 mrad. The two HNBs will be operated with the axis of one pointing upwards, $+X$ mrad, on the axis, and the other pointing downwards $-X$ mrad, off-axis. There is no such requirement of beam steering for the DNB case. The accelerated ion beams are neutralised, $\sim 56\%$, in the neutraliser with an active gas feed of $\text{H}_2/\text{D}_2$ gas, depending on the beam species. The neutraliser is a four-channelled five-walled structure. Forty-four percent of the ionic component of the beam is separated from the neutral part in the electrostatic residual ion dump. The electrostatic ion dump with a layout similar to the neutraliser has three of the five alternate walls at ground potential and the remaining two at $-20 \pm 5$ kV in the HNB case to deflect the positive and negative components of the ion beam from their neutral counterparts. For the DNB the applicable voltage is $-8 \pm 4$ kV.

Prior to injection into the machine the accelerated neutral beam is intercepted by a two armed V-shaped calorimeter which acts as a diagnostic to optimise and characterise the beams. Post optimisation, the calorimeter arms are opened to allow the beam passage to reach the plasma during which the beams pass through the front end components and the duct. The total flight path is $\sim 17$ m for the case of the HNB and $\sim 12$ m for the DNB case. Due to the beamlet divergence and the interaction of accelerated neutrals with the residual pressure in the beam lines [15], the beam traversal is accompanied by losses due to direct interception and reionisation. The reionisation losses are estimated to be $7\%$ for the HNB and $18\%$ for the DNB case. The heat loads have been estimated using a beam transport and reionisation (BTR) code [21], developed by the RRC ‘Kurchatov Institute’ in Russia and have formed the basis of the design of all the beam line and front end components and the panels of the duct liner. The calculations and the results are discussed in detail in [22] and have formed the basis of the design of the components in order to comply with the ITER design guidelines and also survive the lifetime of ITER. For the sake of completeness the summarised results on the exit scraper (ES), FEC, and the DL, for the HNB-1 and DNB beam lines are shown in figure 4.

3. Heating beams in ITER plasma

Studies related to the injection and interaction of neutral beams with ITER plasmas are based on model calculations and cover the important aspects of determining the plasma densities required for the unrestricted full power operation of NBI in ITER with acceptable shine-through loads on the first wall and shield blanket modules, the use of NBI to access and sustain H-mode plasmas, the non-inductive current driven by the beams in ITER plasmas, the power deposited by the beams in the plasma and plasma rotation due to the beams. These aspects are described briefly in the following subsections.

3.1. Model calculations

For the calculations described in this work, two ITER operational scenarios are considered, namely the 7.5 MA/2.65 T, which is foreseen to study the access to the H-mode plasmas in an initial phase, and 15 MA/5.3 T which are those of the reference $Q = 10$ ITER scenario. In addition, operational integration constraints such as those associated with acceptable power loads and to keep a sufficient margin of the edge power flow for sustainment of high confinement H-modes are considered in our evaluations [11]. These integration constraints correspond to the maximum peak power heat load on the divertor plate being lower than $d_{\text{pk}} < 10$ MW m$^{-2}$, the density being in the interval determined by the minimum required for acceptable shine-through loads for full power unrestricted application of NBI heating and the Greenwald limit, $n_{\text{NBI,shine}} < n < n_G = I_p/\pi a^2$, [10$^{20}$ m$^{-3}$, MA, m], and sustainment of high quality H-mode $P_{\text{loss}}/P_{\text{H}} > 1.3$ (for H-mode scenarios) [16]. There is also a
requirement on the fusion gain $Q \geq 10$ for the baseline 15 MA inductive scenario. For completeness we note that the Greenwald limit at 7.5 MA in ITER is $6 \times 10^{19}$ m$^{-3}$ and $1.2 \times 10^{20}$ m$^{-3}$ at 15 MA; the ITER plasma minor radius is $a = 2$ m, the major radius, $R = 6.2$ m and the plasma surface area, $S$ is 680 m$^2$.

To determine the required power to access the H-mode in ITER we apply the empirical scaling in Martin et al [23], which is reproduced below for readers ease

$$P_{\text{thresh}} = C_M 0.0488 e^{0.0570 e_{2D}} n_{e2D}^{7.171 \pm 0.035} B_T^{0.803 \pm 0.032} S^{0.941 \pm 0.019}$$

(1)

where $P_{\text{thresh}}$ is the threshold power to access the H-mode in MW, $n_{e2D}$ is the line average electron density in $10^{20}$ m$^{-3}$, $B_T$ is the magnetic field in Tesla and $S$ is the plasma surface area in m$^2$. The uncertainties to the exponents of each of the contributing terms correspond to the standard errors in their determination.

The mass factor ($C_M$) is $C_M = 1$ for DD plasmas with atomic mass, $A_D = 2$. For hydrogen and tritium plasmas $C_M = A_D/A_i$ where $A_i$ corresponds to the mass of the specie considered. For the case of He plasmas the relation of the H-mode threshold to that of DD plasmas depends on device and experimental conditions. Most experiments (including JET, DIII-D and Alcator C-Mod) [23–26] find that the H-mode threshold of He plasmas is higher than for D plasmas although the ratio between the two depends on plasma density. On the other hand, experiments in ASDEX-Upgrade find that the H-mode threshold in D and He plasmas is the same. The difference between the experiments is likely to be related to the material of the first wall (Carbon versus W) and divertor geometry. For ITER modelling, we adopt the value $C_{M,\text{He}} = 1.4$, which is consistent with the JET findings and a reasonable conservative (i.e. pessimistic or leading to larger power requirements to access the H-mode) assumption for the required input power to access H-mode in He plasmas when the experimental results and remaining uncertainties are considered altogether.

In summary for the ITER modelling studies presented in this paper we use:

$$P_{\text{LH}} = 0.0488 \ C_{M,\text{He}}^{0.717} B_T^{0.803} S^{0.941}$$

(2)

where $C_{M,\text{H}} = 2$, $C_{M,\text{He}} = 1.4$, $C_{M,\text{DD}} = 1$ and $C_{M,\text{DT}} = 0.8$.

The assessment of the L–H threshold discussed above is based on scaling expression (equation (2)) [23] with monotonic increase of the threshold with plasma density. In many experiments this trend does not continue to low densities; instead it is found that under a given density the power to access the H-mode increases with decreasing plasma density [23–26], figure 5. The point at which the trend of the H-mode threshold changes with density determines the minimum power required to access the H-mode for given plasma conditions ($I_p, B_T$, divertor geometry, etc) and we denominate the corresponding density $n_{\text{LH}}$. There is presently no universal picture that describes the experimental trends seen regarding $n_{\text{LH}}$; experiments in ASDEX-Upgrade indicate that this is linked to the thermal decoupling of the electrons and ions at the plasma edge [24, 25], while experiments at JET indicate that the presence of low Z impurities is key [26, 27]. In our modelling we will consider the approach in [24] noting that this is not universally consistent with all experiments. Following [25], $n_{\text{LH}}$ can be evaluated for ITER by applying the following scaling

$$n_{\text{LH}} = 0.07 I_p^{0.34} B_T^{0.62} \left( \frac{R}{a} \right)^{0.4} a^{0.95}$$

(3)

![Figure 5. Qualitative representation of the comparison of $P_{\text{thresh}}$ obtained from the scaling law, equation (2), solid line and the experimental observations with dashed line related to the turn over density.](image)
where $n_{\text{HH}}$ is $10^{19}$ m$^{-3}$, $I_p$, $B_T$, $R$ and $\phi$ correspond to the plasma current (MA), toroidal magnetic field (T) and the major and minor machine radii (m). For ITER, this value corresponds to $\sim 33\%$ of $n_{\text{CH}}$ for plasmas with $I_p = 7.5$–15 MA and $\phi_{DS} = 3$.

It should be noted that significant uncertainties remain regarding the evaluation of the additional heating level required to access the H-mode regime in ITER and its dependence with density at low densities. These include the presence of low Z impurities (C or N) which increases the requirements to access the H-mode with respect to metallic walls (i.e. ITER-like) as found in ASDEX-Upgrade and JET, the role of the divertor geometry which increases the power for vertical plate divertors (i.e. ITER-like), etc. The assumptions used for ITER above represent a reasonably conservative evaluation of the required power for H-mode access; they are based on carbon-PFC dominated devices [23], which overestimate the values of power required in ITER. However the scaling does not take into account the effect of the vertical divertor geometry which increases the H-mode power threshold compared to horizontal divertors as seen at JET [28]. Obviously depending on the final additional heating power that will be required in ITER to access the H-mode, some of the detailed quantitative considerations described below may need to be re-adjusted.

The 1.5D transport simulations of ITER performance have been carried out with the ASTRA code [29] using a semi-empirical approach for transport coefficients [30]. The basic description of the neutral beam heating and current drive block of the ASTRA code can be found in [31]. The beam input to the code corresponds to the beam profile obtained at the exit of the duct for 1280 beamlets each with 5 mrad divergence and with focussing parameters defined in table 2. The ionisation cross sections correspond to those in [15]. The detailed description of the heat and particle transport models with consistent pedestal parameters and boundary conditions is described in [32].

3.2. Electron and ion density and temperature profiles
The electron and the ion density and temperature profiles have been calculated using ASTRA code for different values of the plasma density for both, $B_T = 2.65$ T, $I_p = 7.5$ MA and $B_T = 5.3$ T, $I_p = 15$ MA cases. To vary the temperature values and profiles in a realistic manner in our sensitivity studies we have considered different combinations of one or two NBIs at full power each and EC heating, varying the total input power from 16 MW (1 NBI) to 53 MW (2 NBIs with $P_{\text{NB}} = 33$ MW and $P_{\text{EC}} = 20$ MW) together with location of the heating and CD, foreseen in the ITER design. For full field (5.3 T) operation for H and D plasmas we added 20 MW of the central ICRH to the 33 MW of NB and 20 MW of ECRH to make it 73 MW power in total. It should be noted that in our simulations we consider two assumptions regarding the core density profiles: one is that they are flat due to the lack of NBI sources and the other that there is a moderate density peaking due to an anomalous inwards particle pinch. The latter is consistent with experimental observations and predictions of turbulent ITG transport [33] in ITER [34].

The impurities considered for all the cases mentioned above correspond to 2% Be in addition to Ne, 0.2%–2%. For ICRH full field operation for all species we include 3% of He$^3$ minority while for half field operation with ICRH we consider 3% of hydrogen minority.

Figure 6 shows the typical profiles corresponding to density and temperature for the L- and H- mode plasmas considered in the study without anomalous density pinch. The auxiliary power input for the L-mode plasma is 33 MW of NBI whereas for the H-mode the auxiliary power input considered in 20 MW of EC in addition to the 33 MW of NB. The H-mode plasmas exhibit a sharp gradient of the edge temperatures and densities which correspond to the edge transport barrier in the H-mode plasmas. In the L-mode plasma the density gradient is caused by deeper penetration of the fuelling source for low edge temperatures.

3.3. Shine-through limits
One of the factors which restricts the application of the neutral beam injection (NBI) in the ITER plasmas as in all other tokamaks is the value of the plasma density. The plasma density has to be high enough so that the associated shine-through power fluxes on the ITER first wall and blanket modules are acceptable, which depends on the NBI injected power and the duration of the NBI heating pulse. The shine-through power fluxes correspond to those neutral particles of the beam which are not ionised during their passage through the plasma and hit the plasma facing components. At ITER, the shine-through neutrals will hit the first wall panels 15S and 16S and a small portion will also penetrate into the poloidal horizontal gap, 10 mm, between the two and reach the blanket shield block 16DS (figure 7). If the beam axis coincides with the NB axis (i.e. no tilt to the ion source) the peak of the shine-through beam is incident on this poloidal gap and on the shield block. When the source axis is tilted with respect to the NB axis the peak of the shine-through power moves along FW 15S or FW 16S depending on the direction of the tilt. As the tilt is increased, only the tail of shine-through beam profile intercepts the shield block 16DS for which the power density is much smaller than for the no-tilt case. A representative case of the heat load pattern on the panel walls 15S and 16S and on the shield block 16DS is shown in figure 7 for the case of the non-tilted source axis and source axis tilted by $\pm 10$ mrad. As seen from the figure
the peak NB heat flux for the case of non-tilted beams falls on the shield block 16DS with an estimated power density of \( \sim 5\, \text{MW m}^{-2} \) normal to the beam axis. Therefore, the surface of the shield block receiving this heat load has been extended, for preventing the shine-through particles from reaching the vacuum vessel, which is an important protection component. The slanted shield block surface intercepts the beam at an angle so that the incident power density on its surface is reduced to \( 1.9\, \text{MW m}^{-2} \) as shown in figure 8.

The blanket module shield blocks are not designed to be replaced routinely during ITER’s lifetime, except in the case of malfunction. Therefore it is required that the shield block’s lifetime is appropriate to sustain the 30 000 beam cycles of operation expected during the ITER lifetime. At the level of \( 1.9\, \text{MW m}^{-2} \), detailed thermo-mechanical and fatigue calculations show that the shield block lifetime would be 1000 cycles of operation with 2 sec in duration, as shown in figure 9. In order to have a lifetime of 30 000 cycles in the thermal steady state (i.e. a duration of 100 s or longer), the heat flux in the thermal steady state should be \( \leq 0.3\, \text{MW m}^{-2} \) on the slanted edge, 16DS, which corresponds to \( \leq 0.8\, \text{MW m}^{-2} \) normal to the beam axis using the conversion
factor (5/1.9) mentioned above. Larger power fluxes may be allowed provided that their duration is short and
the number of cycles with these larger fluxes is moderate. This is a similar situation to present experiments where
larger shine-through power fluxes for transient phases are allowed than for longer phases, such as in the
transition from L-mode to H-mode in which the plasma density increases in time following the H-mode
transition.

The allowed level of power density obtained from the above calculations thus becomes the basis to
determining the level of plasma density required to avoid excessive shine-through loads that would limit the
shield block lifetime when NBI is used to heat the plasma in ITER. For simplicity, we will evaluate the value of
this density for the case of unrestricted NBI use, i.e. full power and long pulse length (>100 s) and discuss the
implications of shorter phases with higher shine-through loads which may take place during transients. Given
the power flux limit normal to the beam of 0.8 MW m$^{-2}$ for long pulse operation due to the shield block’s
lifetime, the use of untitled beams leads to more restricted density range to reduce the shine-through loads, as
the maximum of the shine-through power flux falls exactly on the horizontal gap between the 15S and 16S
first wall panels onto shield block 16DS.

However, as seen from figure 7, tilting of the source by ±10 mrad allows the peak of the shine-through
power flux to move onto the first wall panels 15S or 16S so that only the peripheral power flux (a factor of 2.8
lower than the peak) reaches shield block 16DS. This means that when the beams are tilted by ±10 mrad, the
maximum shine-through power flux normal to the beam on the first wall panels 15S and 16S which is
compatible with the lifetime of shield block 16DS for long pulse operation is 2.2 MWm$^{-2}$, (i.e. 0.8 MWm$^{-2}$ on
the shield block 16DS multiplied by 2.8 as the shield block is in the periphery of the shine-through power flux
patch). From this evaluation it is clear that operation with beam source tilted by ±10 mrad is the most favourable
choice to allow operation at low plasma densities and high NBI powers; in addition the −10 mrad, +10 mrad
configuration is also advantageous to minimise Alfvén eigenmodes excitation. Long pulse NBI operation with
smaller tilts will necessarily require operation at higher densities or a reduction of the injected NBI power
compared to ±10 mrad to ensure that the shine-through power flux intercepting the shield block 16DS does not
exceed the limit of 0.8 MW m$^{-2}$ normal to the beam required by the shield block lifetime.
The density required for acceptable shine-through power fluxes on the first wall and blanket shield block for unrestricted application of NBI to plasma heating in ITER with ±10 mrad tilted beams. The value of \( n_{\text{GW}} \) at 7.5 MA and 15 MA is shown for illustration. The 870 keV H beams are proposed to be used during the He and H phases at ITER and the 1 MeV D beams during the D0 and DT operational phases.

Oikawa et al. [35] performed a detailed study of the expected power densities at FW for different plasma scenarios including 7.5 MA/2.65 T He and hydrogen L-mode plasmas and 15 MA/5.3 T DT L- and H-mode plasmas. The study includes the effects of various impurity concentrations and \( Z_{\text{eff}} \). The results of these studies are summarised in figure 10 where the maximum shine-through power flux on the first wall of 2 MW m\(^{-2}\) is considered, i.e. 10% lower than the maximum allowable for the shield block 16DS for long pulse NBI operation of 2.2 MW m\(^{-2}\), to calculate the minimum density with acceptable shine-through loads for unrestricted heating with NBI (i.e. full power and long pulse (>100 s)). As can be seen from this figure 10 operation with H\(^6\) NBIs requires higher densities to achieve the same shine-through loads because of the higher velocity of the accelerated neutrals compared to D and the associated lower ionisation efficiency of the plasma. The data in this figure can also be used to evaluate the minimum plasma current required for long pulse operation with NBI under the assumption that a plasma density of \( n_{\text{GW}} \) can be achieved in all H-mode conditions in ITER. This corresponds to 2.4 MA in DT, 3.6 MA in He and 5.2 MA in H. It also should be noted that the minimum density for unrestricted use of NBI heating is always above that for minimum H-mode threshold power (\( \sim 0.3 n_{\text{GW}} \)) for 7.5 MA plasma operation in ITER. Note that the maximum current, \( I_p \sim B/\dot{q} \), is determined by the edge safety factor MHD stability limit, \( \dot{q} \geq 3 \).

The density required for long pulse NBI heating of ITER plasmas can be decreased by decreasing the beam energy. This decreases the velocity of the injected neutrals, increases the plasma ionisation efficiency and decreases the shine-through loads. Operation at lower beam energy has the disadvantage that the power injected in the plasma decreases strongly (as \( \sim E_{\text{NB}}^{-2.5} \)) compared to full energy due to the need to operate the ion source in the permeance matched conditions for optimal beam optics. Figure 11 shows the dependence of the minimum density required for acceptable shine-through power fluxes (peak power density perpendicular to the beam of 2 MW m\(^{-2}\)) on the beam energy for H beams in H and He L-mode plasmas, and for D beams in DT H-mode plasmas. The large decrease of the required density for acceptable shine-through in long pulse NBI operation with decreasing beam energy is due, to a large degree, to the fact that the total power injected by the NBI decreases significantly when the beam energy is reduced (16.5 MW at 870 keV to 4 MW at 500 keV per beam box for H beams).

3.4. Access and sustainment of H-modes

Auxiliary power is necessary to transit from the low confinement mode (L-mode) to the high confinement regime (H-mode), \( P_{\text{aux}} + P_\alpha + P_{\text{OH}} - P_{\text{rad}} > P_{\text{LH}} \), which provides the energy confinement required for ITER to achieve its \( Q = 10 \) goal. At ITER the input power required for such a transition will be provided by the available power from the planned heating and current drive systems, viz. neutral beam (HNB), electron cyclotron (EC) and ion cyclotron (IC) systems with 73 MW in total. As shown in table 1 should the need be, these systems will be upgraded to provide additional 57 MW power depending on the need; it should be noted that the maximum power injected in the vacuum vessel simultaneously by all upgraded systems is limited to 110 MW.

Figure 12 shows the threshold power as a function of the electron density for the D, DT, He and H plasma scenarios of ITER. The two curves for a given scenario shown on the figure refers to the high and low values of the \( P_{\text{thresh}} \) calculated using the upper and lower values of the exponents in equation (1) and represent the window where the threshold for the L–H transition should lie for a given plasma density.
Figure 11. Required density for acceptable shine-through (2 MW m\(^{-2}\) on the FW panels and 0.8 MW m\(^{-2}\) on the shield block) in stationary NBI operation versus beam energy for (a) H beams in H and He L-mode plasmas and (b) for D beams in DT H-mode plasmas. Also shown is the injected power into the plasma for each of the two beams assuming a 7 mrad divergence of the beamlets.

Figure 12. H-mode power threshold as a function of the plasma density for the 5.3 T/15 MA DD, DT and 2.65 T/7.5 MA scenarios for the H, He ITER plasmas.
Table 3. L–H transition values for various cases at the minimum density required for unrestricted use of NBI for plasma heating with acceptable shine-through power fluxes for pure plasma w/o impurities.

| Specie | SHT density ($\times 10^{19}$ m$^{-3}$) | $B_1$ (T) | Scaling factor $C_M$ | $P_{LT, \ max}$ (equation (1)) (MW) | $P_{LT, \ min}$ (equation (1)) (MW) | $P_{LT}$ (equation (2)) (MW) | $P_{LT}$ (nL–H) (equation (3)) (MW) | $P_{LT}$ (nL–H) (equation (2)) (MW) |
|--------|------------------------------------|-----------|---------------------|-----------------------------------|-----------------------------------|----------------------------|---------------------------------|---------------------------------|
| He     | 3.0                                | 2.65      | 1.4                 | 35                                | 25                                | 30                         | 2.1                             |                                 |
| H      | 4.4                                | 2.65      | 2                   | 66                                | 46                                | 56                         | 2.1                             |                                 |
| H      | 4.4                                | 5.3       | 2                   | 120                               | 80                                | 97                         | 4                               |                                 |
| D      | 2.2                                | 2.65      | 1                   | 20                                | 14                                | 17                         | 2.1                             |                                 |
| D      | 2.2                                | 5.3       | 1                   | 35                                | 24                                | 29                         | 4                               | 45                              |
| DT     | 2.2                                | 5.3       | 2/2.5               | 28                                | 19                                | 23                         | 4                               | 36                              |

* This value of the density is lower than that estimated ($4 \times 10^{19}$ m$^{-3}$) for the scaling laws in equations (1) and (2) to be applicable according to the assumptions in [25].
As mentioned in the previous section, a minimum plasma density is required to avoid excessive shine-through for unrestricted use of neutral beams for plasma heating in ITER. Table 3 shows the estimates on the L–H threshold power requirements at these densities for a range of plasma conditions with the max values corresponding to the positive signs of the exponents in equation (1) and min values corresponding to the negative values of the exponents together with the average values from equation (2). It should be noted that, with the exception of 15 MA/5.3 T plasma in DD and DT, the minimum shine-through density is above the density for minimum H-mode threshold power evaluated according to [25] equation (3) and therefore the H-mode power requirement derived from these equations is expected to apply.

The values quoted in the above table are the threshold powers needed to access the H-mode at the density required for the unrestricted NBI heating of ITER plasmas with acceptable shine-through loads. However, it should be noted that, following the H-mode transition, the plasma density and temperature increase. This density increase, which in ITER is more moderate than in present experiments due to the low core fuelling efficiency by recycling neutrals [36–38], together with the need to maintain a margin above the H-mode threshold in order to obtain high confinement H-mode plasmas implies that the power required to sustain high confinement H-mode plasmas is larger than the L–H power thresholds given in the estimates above. Indeed the requirements regarding the minimum density for unrestricted NBI heating of ITER plasmas with acceptable shine-through loads apply to the H-mode stationary phase and not to the L–H mode access phase. During this H-mode access phase, in which ITER is estimated to take ∼4 s (i.e. 2 $\tau_{H-\text{stationary-modes}}$) for H and He plasmas [37], shine-through power fluxes a factor of two larger than those for long pulse operation, are compatible with an acceptable shield block lifetime, as shown in figure 9. The final choice of the H-mode operating density in ITER during the non-active phase will also have to take into account other physics processes associated with the possible detrimental effects of instabilities caused by fast particles from the NBI, which can lead to a substantial fast ion pressure at low densities $\beta_{\text{fast}}/\beta_{\text{thermal}} \sim 20\%–25\%$ [39]. Therefore, the values in table 3 above based on the stationary shine-through requirements together with those related to the applicability of the scaling in equations (1) and (2) should be used to only provide guidance to evaluate the minimum power/density required to sustain high confinement H-modes in ITER (i.e. 1.3 $P_{1\text{H}}$ in table 3) while the final values will have to be determined experimentally taking into account several other scenario integration issues (i.e. ELM control, fast particle effects, etc). In this respect, already from the values in table 3, it can be concluded that H-mode access/sustainment in pure hydrogen plasmas is not likely in ITER. At 5.3 T the expected required power is higher than the available baseline heating power of 73 MW, while at 2.65 T there is no suitable scheme for ICRH heating so that the available power is only 53 MW. H-mode access experiments (not sustainment) could be envisaged at 2.65 T by operating with 20 MW of ECRH and 33 MW of NBI for short periods (few seconds) at lower densities than those in table 3, but above 2 $10^{19}$ m$^{-3}$ (minimum H-mode threshold), for which the shine-through loads compatible with an acceptable shield block lifetime are larger (see figure 9).

The results from the ASTRA calculations for the margin to the H-mode power threshold in He, H, D and DT plasmas at different densities up to $n_{GW}$ are shown in figures 13(a)–(d) respectively. Also shown in figures 13(a) and (b) are the time durations for which the shine through corresponding to full power NBI heating of the shield block is allowed for a given plasma density. For He and H plasmas it is assumed that only 33 MW of NBI is applied and the results show that indeed this may be marginal to access the H-mode in He plasmas at 7.5 MA/2.65 T and certainly not sufficient to sustain it so that the additional heating provided by ECRH and/or ICRH will be required for this. For H-mode plasmas, as mentioned before, even with the addition of 20 MW of ECRH the access to H-mode at 7.5 MA/2.65 T will be marginal and only possible for few seconds if densities lower than 4.2 $10^{19}$ m$^{-3}$ are required for this due to shine-through power limitations.

Two scenarios are considered for DD plasmas in figure 13(c). One corresponds to 7.5 MA/2.65 T for which the auxiliary power used for the calculations is 53 MW, 33 MW NB and 20 MW of EG, and the other scenario relates to 15 MA/5.3 T for which the auxiliary power considered is 73 MW, 20 MW of IC in addition to 53 MW from the NB and EC. The calculations cover the range from the minimum density for validity of the H-mode threshold scaling law ($\sim 0.3 n_{GW}$; i.e. 2 $10^{19}$ m$^{-3}$ for 7.5 MA and 4 $10^{19}$ m$^{-3}$ for 15 MA) up to $n_{GW}$. It should be noted that the density required for acceptable lifetime of the shield block under the shine-through loads with full power NBI and long pulse in these conditions is 2.2 $10^{19}$ and thus practically all points in this figure are compatible with long pulse full power NBI heating. The results in this figure show that with respect to the density the operational space of DD plasma for 7.5 MA/2.65 T with high confinement H-modes is very large and can reach up to $n_{GW}$ if 53 MW of NBI are applied while it is very restricted (i.e. $n_e \sim 4.0\;10^{19}\;\text{m}^{-3}$) for 15 MA/5.3 T even when 73 MW of additional heating is applied.

The calculations for 15 MA/5.3 T DT plasmas with 33 MW of NB and 20 MW of ECRH are shown in figure 13(d). In contrast to H, He, and DD plasmas the ratio $P_{1\text{H}} = P_{\text{sat}}/P_{1\text{H}}$ in DT plasmas increases with the density, which is due to the increased contribution of the alpha heating, $P_{\alpha}$, with the increasing density in DT operation, as shown in figure 14. The increase in $P_{\alpha}$ with increasing density is caused by the increased fusion reactivity of the DT plasma which depends both on density and temperature and which is highest for densities
Figure 13. Margin above the H-mode threshold $P_{th} = P_{sol}/P_{LH}$ and time duration for full power NBI heating leading to shine-through loads compatible with an acceptable lifetime of the shield block for (a) He and (b) H plasmas with H NBI 33 MW heating only and 7.5 MA/2.65 T, $P_{aux} = 33$ MW, (c) DD plasmas with $P_{NB} = 33$ MW D NBI + $P_{EC} = 20$ MW for 7.5 MA/2.65 T and with an additional 20 MW of ICRH for 15 MA/5.3 T and (d) DT plasmas 5.3 T 15 MA cases with $P_{NB} = 33$ MW D NBI + $P_{EC} = 20$ MW for 15 MA/5.3 T. For (c) and (d) the shine-through loads allow application of full power NBI without restrictions associated with the lifetime of the shield block; in these two cases the shine-through loads reaching the shield block (in % of the maximum allowed for 100 s full NBI power operation) are provided in this figure. In all cases the impurity in the core plasma considered is Be with a concentration $n_{Be}/n_e = 2\%$. 
close to \( n_{\text{GW}} \) at 15 MA in ITER (i.e. the \( Q = 10 \) reference scenario). For conditions in which the core plasma radiation can be kept low (such assumed in this study with only Be impurity), the \( P_{1H} \) can reach values as high as 1.8 in \( Q > 10 \) plasmas. Indeed alpha heating is an essential contributor to the access and sustainment of 15 MA/5.3 T H-mode plasmas with high \( Q \) in ITER. Studies in [38] have shown that it is required to control the density evolution by gas puffing and pellet fuelling to optimize the path from L-mode to H-mode at 15 MA/5.3 T by maximising the ratio of \( P_{\alpha} \) to \( P_{1H} \) to ensure robust access to \( Q = 10 \) H-mode in ITER with an additional heating level of 33 MW.

3.5. Effect of impurities on shine-through loads and consequences for H-mode access power requirements

The shine-through loads are determined by the effective ionization capability of the plasma for fast neutrals injected in the plasma by the NBI. This includes not only ionization events of the fast ions by H, He, D or DT but also by impurities with the effective ionization of the plasma increasing with impurity density. Thus the presence of impurities (beyond Be) in the plasma decreases the density for which unrestricted long pulse NBI heating can be applied in ITER with acceptable shine-through loads. This can allow, in principle, the reduction of the density and stationary H-mode operation with lower values of the plasma density than those determined by the acceptable shine-through loads. This is of particular interest for the cases in which the density for stationary H-mode operation with acceptable shine-through loads is above the density for a minimum H-mode power threshold such as for H plasmas at 7.5 MA/2.65 T and to a lesser extent He plasmas at the same current and field as the shine-through density is lower.

While increased impurity densities may allow stationary H-mode operation at lower densities compatible with acceptable shine-through loads, they also lead to higher plasma radiation and thus can reduce the effective power deposited in the plasma and impair H-mode access for a given power level. In order to quantify how these two effects affect H-mode access/sustainment for 7.5 MA/2.65 T H plasmas, we have carried out the 1.5D transport analyses with consistent fuelling, boundary conditions for the L-mode operation following the model described in [31] and including Ne in the simulations, which is one of the extrinsic impurity species considered for divertor power load control in ITER. The additional heating considered is \( P_{\text{NB}} = 33 \) MW plus \( P_{\text{EC}} = 20 \) MW (note that the ICH is not efficient at 2.65 T for hydrogen plasmas in ITER) and we have varied the Ne concentration in the main plasma \( n_{\text{Ne}}/n_e \) assuming that it has a density profile similar to that of the electron density [40]. It should be noted that medium-Z impurities such as Ne tend to have hollow density profiles in H-mode in present experiments even when the electron profiles are peaked; this may lead to an underestimation of the core plasma radiation in the real ITER H-mode plasmas compared to our estimates. The results of the simulations shown in figure 15 show that it is possible to operate with unrestricted long pulse NBI heating with acceptable shine-through loads and access the H-mode for \( n_e < 3.6 \times 10^{19} \) m\(^{-3}\) (or 0.66 \( n_{\text{GW}} \)) provided that the neon fraction is greater than 0.7%. To have more margin so that H-mode confinement can be sustained in hydrogen plasmas with acceptable stationary shine-through loads would require a neon fraction in the core greater than 2% and operation with a density of \( n_e = 2.7 \times 10^{19} \) m\(^{-3}\) (0.45 \( n_{\text{GW}} \)) for which \( P_{\text{aux}} / P_{1H,H}(n_{e,\text{min}}) \sim 1.24 \) for the auxiliary heating of 33 MW. It is important to note that this density value is above that required for the H-mode threshold scaling in equations (1) and (2) to apply (\( n_e = 2.0 \times 10^{19} \) m\(^{-3}\) (0.33 \( n_{\text{GW}} \))).
3.6. NB current drive

Besides providing heating, neutral beams have also been used to drive current in the plasmas to replace part of the inductive current in present tokamaks. As mentioned above the NB driven current has a radial profile dependent on the beam energy, injection angle and plasma parameters. Figure 16 shows the variation in the NB induced ion current profile for a 9 MA/5.3 T DT Q = 5 ITER plasma scenario with one injector operated with 1 MeV D beam and with the ion source tilt varied from −9 mrad to +9 mrad with respect to the neutral beam axis using the ASTRA code. Figure 17 shows similar calculations for the 15 MA/5.3 T scenario with 33 MW from two 1 MeV D injectors with one operating at +9 mrad and the other at −9 mrad for the driven current, taking into account the screening by electrons.

Figure 18(a) shows the beam driven current density ($J_{NB}$) and total current density ($J_{tot}$) profiles obtained for the 7.5 MA/2.65 T H-mode plasmas in DD for two plasma densities $n/n_G = 0.62$ and $n/n_G = 0.92$ with 33 MW of NBI and with one beam on-axis and one off-axis. The peak in $J_{tot}$ at the edge of the plasma is due to the edgebootstrap driven by the density and temperature gradients in the H-mode pedestal together with the of plasma resistivity towards the separatrix. Figure 18(b) shows the ratio of the NBI driven current density to the
total plasma current density, $J_{\text{NB}}/J_{\text{tot}}$, at $\rho = 0.1$ versus the plasma density for 7.5 MA/2.65 T and 15 MA/5.3 T for DD and DT H-mode plasmas. At $n_e = 0.62 \, n_{\text{GW}}$ the NBI driven density fully or almost fully replaces the inductive current density, $J_{\text{NB}}/J_{\text{tot}} \sim 0.8–1$ for 7.5 MA plasmas at $\rho = 0.1$. At higher plasma densities and for 15 MA/5.3 T plasmas the fraction of the NBI current density in the centre is typically much smaller. A similar qualitative trend is reflected in figure 19 which shows the fraction of the NBI driven current ($I_{\text{NB}}/I_{\text{tot}}$) as a function of the plasma density. Only for 7.5 MA/2.65 T plasmas and moderate densities ($n_e \sim 0.62 \, n_{\text{GW}}$) does the NBI driven current reaches values close to 50% of the total plasma current.
3.7. NB heating

Post injection collisions between the neutral beams with the plasma ions and electrons lead to a multi-step ionisation process of the injected neutrals. Coulomb interactions with plasma ions and electrons slow down fast ions to the thermal energies, thus heating the ions and electrons in the plasma. The power deposition profile depends on the ionisation cross section, the plasma electron density profiles and the plasma main ion and impurity species.

Power deposition profiles by the ITER NBI beams have been studied for 33 MW 870 keV H beams to be applied to heat He and H plasmas in the non-active operations phase and 33 MW 1 MeV D beams to be applied to heat DD and DT plasmas in the active operations phase. The studies consider a range of plasma densities and different combinations of NBI with EC and IC heating as mentioned in the previous sections.

Figure 20 shows typical power deposition profiles from the three ITER heating systems for a 15 MA/5.3 T DD plasma simulated by ASTRA.

The power from the neutral beams is shared between the plasma ions and electrons and this sharing depends on plasma parameters (electron temperature) that control the slow-down of plasma ions. Figure 21 shows the typical NBI power density profiles deposited on the plasma thermal ions and electrons for the DT 15 MA/5.3 T ITER baseline H-mode scenario with $Q = 10$ ($P_{\text{tan}} = 500$ MW) (pedestal temperature of 3.5 keV and $\langle T_e \rangle$ 9.5 keV have been used in these simulations). In these conditions the central power deposited is equally shared between ions and electrons (which contributes to the increase of the fusion reactivity by higher ion temperature) although the total deposited power on the plasma electrons is higher by factor of two than that on the ions for these conditions.

Figure 22 shows the variation of the total power deposited on the ions and electrons by the neutral beams for 7.5 MA/2.65 T DD and 15 MA/5.3 T DT plasmas versus the volume averaged value of the electron temperature. The power deposited on the ions increases with increasing electron temperature, which as mentioned above, is
beneficial for the increase of fusion reactivity in DT plasmas. If high central/average temperatures can be achieved in ITER for 15 MA/5.3 T operation then \( P_{\text{NBI}}/P_{\text{NBE}} \approx 1 \) for \( T_e \approx 14 \text{ keV} \) will be achieved, while values of \( P_{\text{NBI}}/P_{\text{NBE}} \approx 0.5 \) are expected for 7.5 MA/2.65 T plasmas for \( T_e \approx 7–10 \text{ keV} \).

### 3.8. Plasma rotation

Some level of toroidal plasma rotation [41, 42] is required in tokamaks to avoid the development of locked modes which typically lead to disruptions. In addition, shear of the toroidal plasma rotation is also found experimentally to reduce turbulence and improve the plasma energy and particle confinement, in agreement with theoretical understanding. Although the turbulent processes in the plasma are found to produce toroidal plasma rotation, the prediction of the magnitude of this ‘intrinsic’ rotation in ITER remains very uncertain and, thus, the NBI remains the only system in ITER that can provide a controlled external source of toroidal plasma rotation. To quantify this, detailed studies of the achievable level of plasma rotation have been performed for H and He plasmas at 7.5 MA/2.65 T and for 15 MA/5.3 T plasmas in DT for L-mode and H-mode plasmas with the ASTRA code. In the studies we considered \( ^{1} \text{H}^0 \)-NBI with \( E_{\text{NB}} = 870 \text{ keV} \) (for H and He plasmas), \( ^{3} \text{He}^0 \)-NBI with \( E_{\text{NB}} = 1 \text{ MeV} \) (for DT plasmas), one and two NBI sources with a power of 16.5 MW in each beam. In addition, for the H-mode plasmas we studied the sensitivity of the achieved rotation speed to pedestal parameters, ratio of \( \chi_i/\chi_i \), density peaking level, confinement enhancement level over the ITER(98, y2) scaling and combination of NBI heating with other RF schemes.

To quantitatively compare the results of our 1.5D self-consistent simulations studies we consider the equation for the volume integrated momentum evolution, which is described by:

\[
\langle \Sigma M_i n_i dV_i/dt \rangle = S_\phi - \langle \Sigma M_i n_i V_\phi \rangle / \tau_\phi
\]

where \( M_i, n_i \) are the mass and density of the plasma ions, \( S_\phi \) is the momentum source from the beam, \( V_\phi \) is the toroidal rotation velocity and \( \tau_\phi \) is the momentum confinement time.
In stationary conditions the toroidal rotation velocity is

\[ V_\phi = S_\phi \tau_f / \langle \Sigma M_i n_i \rangle. \]  

(5)

Assuming a low impurity concentration \( Z_p n_p \sim n_o \) where \( n_p \) is the main ion density and \( Z_p \) its charge, so that \( \langle \Sigma M_i n_i \rangle \sim n_o M_p / Z_p \) and \( V_\phi = S_\phi \tau_f (Z_p / M_p) / \langle n_o \rangle \), where the ratio \( Z_p / M_p \) is smallest for a DT plasma (compared to H and He).

The momentum source from the NBI depends on the NBI power \( (P_{\text{NB}}) \), energy \( (E_{\text{NB}}) \), and isotope \( (M_{\text{NB}}) \), mass, \( S_\phi \sim P_{\text{NB}} (R_{\text{BT}} / R) (E_{\text{NB}} / M_{\text{NB}})^{-1/2} \), where the \( R_{\text{BT}} \) is the tangency radius of the NBI defined in earlier sections and \( R \) is the major plasma radius. The momentum source is higher for the \( D^0 \)-NBI than for the \( H^0 \)-NBI.

In addition to the momentum source and the ion charge to mass ratio, the toroidal rotation velocity increases with momentum confinement time. It has been experimentally found that there is an overall correlation between energy and momentum confinement where for most plasma conditions \( \tau_f \sim K \tau_E \), with \( K = 0.5-2 \) \(^6\), \(^{43}\) and we have considered this range in our studies to evaluate toroidal rotation in ITER. On the other hand the sensitivity of the modelled plasma rotation to the choice of pedestal parameters (we scanned the pedestal temperature by a factor of \( \sim 2 \)) appears to be weak.

In figure 23 we show the modelled toroidal plasma rotation profiles (assuming \( \tau_f = \tau_E \)) for a set of 15 MA/5.3 T plasmas in a density range covering \( n = 4-11 \times 10^{19} \text{ m}^{-3} \) (or 0.34–0.93 \( n_{\text{GW}} \)). This is the typical density range covered from initial gas fuelled conditions to pellet fuelled dominated plasmas in the access phase to stationary H-mode in the 15 MA/5.3 T DT Q = 10 scenario in ITER \(^{31}\) with 33 MW of \( D^0 \)-NBI. To illustrate the effect of different plasma species and plasma and beam isotopes we also display the toroidal rotation velocity calculated for 7.5 MA/5.3 T H and He plasmas with 33 MW of \( H^0 \)-NBI. The highest toroidal rotation velocities correspond to DT plasmas at 15 MA (larger \( \tau_f \)) with low density as expected from equations (4) and (5).

The modelled central \( (\rho = 0.1) \) toroidal velocity depends linearly on momentum confinement, as also expected from equations (4) and (5), which is illustrated in figure 24.
It should be noted, that in spite of the larger mass of ITER plasmas, due to their larger volume, and the unfavourable dependence of the momentum source on the NBI energy, $\sim 1/\sqrt{E_{\text{in}}}$, the maximum rotation speed predicted for ITER is of the same order of magnitude $\sim 100 \text{ km s}^{-1}$ at the plasma centre as in present experiments with similar levels of injected NBI power, although the resulting angular rotation and the value of the velocity normalised to the plasma central Mach number is typically low ($M < 0.1\text{–}0.2$) compared to present experiments. Such a high values of the toroidal rotation velocity are due to the much larger momentum confinement time predicted for the ITER plasmas and the almost tangential injection, $R_{\text{RT}}/R > 0.8$ leading to a large contribution of the NBI to toroidal momentum input.

4. Summary

This paper describes in detail the various aspects of HNBs to ITER plasmas. While 870 keV H beams shall be used for H and H plasmas in the non-active phase, 1 MeV D beams are planned for D and DT plasmas in the active phases of ITER operation. 33 MW of injected neutral beam power is expected from the two heating beam lines and the presently foreseen configuration is that one beam line will be operating on-axis and the other off-axis. The design aspects for the production of such beams presented in section 1 include the beam parameters, beam production and transport and optimal heat loads on beam intercepting components of the beam line to survive the ITER lifetime.

Studies on the expected shine-through loads on the first wall panels and shield block intercepting the beams and their dependence on plasma density, plasma specie and isotope and beam isotope have been performed. It is found that due to the magnitude of these loads the configuration which is most favourable to provide the widest density operational range while decreasing the risk of triggering Alfvén instabilities due to the NB injected fast particles is with one ‘on-axis’ beam and the other one ‘off-axis’. The studies show that unrestricted application of long pulse NBI heating in ITER to the highest power (16.5 MW per beam line) with acceptable lifetime of the shield block modules intercepting the shine-through require highest plasma densities for H plasmas $4.2 \times 10^{19} \text{ m}^{-3}$ with 870 keV H beams. This value is reduced to $3.0 \times 10^{19} \text{ m}^{-3}$ for He plasmas with 870 keV H beams and furthers down to $2.2 \times 10^{19} \text{ m}^{-3}$ for DD and DT plasmas with 1 MeV D beams. For shorter pulses (few seconds) lower densities can be allowed, while being compatible with an acceptable lifetime of the shield block modules intercepting the shine-through. This is would be an option to consider when accessing H-mode plasmas from lower density L-mode plasma conditions in ITER, as the density increases following the H-mode transition leading to a decrease of the shine-through loads in a timescale of few seconds to acceptable levels for the subsequent longer phase of the NBI heating.

Studies of access and sustenance of H-mode plasma conditions taking into account the shine-through loads on the shield block and the dependence of the H-mode threshold on plasma density for H, He, DD and DT plasmas have been carried out. These studies show that H-mode operation in H plasmas is not likely in ITER for the range of toroidal fields 2.65–5.3 T due to the higher densities required for acceptable shine-through loads and the higher H-mode threshold. For He plasmas, H-mode operation is possible for 7.5 MA/2.65 T and plasma densities above $3 \times 10^{19} \text{ m}^{-3}$. In this case, the main open issue concerns the maximum achievable density in L-mode conditions by plasma fuelling and the subsequent increase of the plasma density after the H-mode transition, although this is presently found to exceed $3 \times 10^{19} \text{ m}^{-3}$ by integrated modelling studies. It should be noted that even if such value could not be obtained and the H-mode density would be limited to that typical of He L-modes, the maximum rotation speed $\sim 1 \text{ at the plasma centre as in present}$, higher than that required for shine-through, is the practical limit for the lower density range of H-mode operation. Due to this minimum power threshold density, the higher H-mode threshold for DD than for DT plasmas and the lack of alpha heating, the H-mode operational space for DD plasmas at 15 MA/5.3 T is very small (i.e. a single density value of $\sim 4.5 \times 10^{19} \text{ m}^{-3}$ for a total heating power of 73 MW). It should be noted that the evaluations above have been performed for plasmas with negligible impurity content. Including impurities into consideration relaxes the density required for acceptable stationary shine-through loads at full NBI power and can open the H-mode operational space as the reduction of the power required to sustain the H-mode through the decrease of plasma density is larger than the increased core radiation by the impurities in the plasma. This effect is more noticeable for hydrogen plasmas at 7.5 MA/2.65 T for which the possibility to sustain H-mode operation exists for Ne concentrations of 2% and heating power of 53 MW (33 MW of NBI plus 20 MW of EC). It may however be noted that in this paper we do not discuss the operational scenarios which depend on transport models, see for e.g. [44]. In our consideration we assess the
boundaries of the operational space where the use of the NBI is safe and does not depend on any specific transport model.

Neutral beam assisted current drive has also been studied for H, He, DD and DT plasmas in ITER for both 7.5 MA/2.65 T and 15 MA/5.3 T. For 7.5 MA plasmas and low density operation near either the stationary shine-through level or the minimum H-mode threshold power the fraction of the current density produced by NBCD can reach ~100% near the plasma centre and provide ~50% of total current increasing significantly the duration of the plasma that can be sustained with the available solenoid flux due to the reduction of the resistive flux consumption by a factor of 2. The percentage of NB driven current density, however, decreases with increasing plasma density and it is only 20% contribution for the 15 MA/5.3 T conditions required for the achievement of Q = 10 in ITER. On the other hand the contribution from the bootstrap current, due to the pedestal gradients, to the total plasma current increases with plasma density partially compensating the reduction of the NBCD.

Heating by neutral beams of the plasma ions and electrons are found to depend on the plasma species and isotopes, value of the electron temperature and on the neutral beam specie and energy. For H-mode plasmas at 7.5 MA/2.65 T electron heating by the beams is dominant whereas the proportion of the power transferred to the ions increases for 15 MA/5.3 T H-mode plasmas due to the higher electron temperatures. This is beneficial to increase the plasma ion temperature and fusion reactivity in DT plasmas. From the core plasma fuelling point of view the contribution from the NB in ITER is insignificant due to the high neutral energy.

Evaluation of the toroidal plasma rotation that can be sustained with NB in ITER has been performed for H, He, DD and DT plasmas in ITER. The resulting values of the toroidal plasma rotation velocity are comparable to those in present experiments, although the corresponding angular rotation frequencies and Mach numbers are lower, indicating that the application of NB in ITER will be beneficial to decrease core plasma turbulent transport and to decrease the risk of mode locking in a qualitatively similar fashion to present tokamak experiments.

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