Assessment of Plasma Parameters for Low Activation Phase of ITER Operation

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Abstract

An assessment of ITER plasma parameters is carried out for the low activation phase that is required for commissioning the basic ITER systems including plasma control, heating and current drive, etc. Such operation is analyzed for hydrogen, helium and deuterium plasmas for full field and current, as well as with magnetic field and plasma current reduced to half of their design values, B_0 = 2.65 T, I_p = 7.5 MA. Both hydrogen and deuterium neutral beam injection (NBI) are considered. We assess the possible operating domain for safe operation, and possible target plasmas for commissioning the NBI, electron cyclotron heating (ECH) and ion cyclotron heating (ICH) systems, taking into account the NB shine-through (NBST) loss, Greenwald limit and access to H-mode operation. Simulations with the Automated System for Transport Analysis (ASTRA) show that for 33 MW of NBI with 20 MW of ECH and 20 MW of ICH, H-mode access is marginal for hydrogen plasmas. Good H-mode confinement, expected at $P_{NB} + P_{EC} + P_{IC} > 1.5\ P_{L\text{-}H}$, is more likely for helium and deuterium cases. It is found that plasma parameters for full power/half field/half current operation can be similar to those required for DT long pulse operation. Preliminary assessment is also made of the upper limit of tritium and neutron yield for the deuterium phase of ITER operation.

1. Introduction

At present four phases of operation are foreseen in ITER. The first phase is a non-nuclear phase in which the tokamak and its various subsystems are commissioned, the licensing assumptions for proceeding to active operation are validated and the first steps towards developing plasma scenarios are taken. During the second, deuterium phase the scenarios required for DT operation have to be developed and brought to full performance. For the following DT operation two different phases are foreseen. The first DT phase has the goal of achieving the Q = 10 mission, of developing regimes that are compatible with true steady-state operation and of exploring a wide range of plasma physics issues in the burning plasma state; and a second DT phase with the goal of demonstrating technologies and operating regimes that will be used in a demonstration fusion reactor to follow ITER. In this paper we restrict our consideration to assessment of plasma parameters foreseen for ITER operation during the non-active and deuterium operation phases.

The operation sequence for DT plasmas – plasma current initiation, current ramp-up, formation of the divertor configuration and current ramp-down – in principle can be developed or simulated in hydrogen or helium plasmas in a non-nuclear environment. The divertor performance of DT plasmas can be also checked in the low activation phase, at least under L-mode conditions. However, the divertor performance under H-mode conditions can only be examined in H and He plasmas if there is sufficient auxiliary heating to compensate for the lack of fusion alpha-particle heating and higher power requirements for the L-H transition. Characteristics of electromagnetic loads due to disruption or vertical displacement events and wall heat loads due to runaway electrons in the low activation phase are basically the same as those of the DT phase. Careful studies in the low activation phase would significantly reduce the uncertainties of full DT operation.

2. Operational Space for Low Activation Phase of ITER

Following a period of Integrated Commissioning to demonstrate the readiness of major tokamak subsystems, demonstration of plasma operation will define the completion of the Construction Phase. First Plasma with hydrogen at the end of the construction phase can be considered essentially as a demonstration of the integrated operation of the tokamak subsystems to the level that allows plasma breakdown to be achieved.

During the following non-active phase, hydrogen and helium plasma scenarios will be developed to allow the full commissioning of all tokamak sub-systems (except systems involving the use of deuterium or tritium) with plasma. If injected power levels allow, initial H-mode operation (preferred in hydrogen, but more likely in helium plasmas) will be established and H-mode operation characterized. A key milestone for the non-active operation phase will be the demonstration of plasmas at the full technical capability of the device (15MA/5.3T). The plasma pulse length, including that in plasmas at reduced parameters, will likely be limited by the operating time available to develop long-pulse operation. By the end of the non-active operation phase, all experimental data required to ensure the granting of a license for nuclear operation of the facility should be available.

The H-mode operating space is restricted by the Greenwald limit, $n/n_G < 1$, by power loss across the separatrix exceeding the L-H power threshold, $P_{loss}/P_{L\text{-}H} > 1$, and by the NBI shine-through (NBST) limit, $P_{NB,shine} < 0.5$ MW/m^2 [1] (in the absence of additional armour on the inside wall, which is under consideration as a design change to raise the limit to 4 MW/m^2).

2.1 NBI Shine-Through Limit

The NBST wall load depends mainly on the NBI power density and energy, the species of the injected neutrals, the plasma density and its contamination by impurities. Hydrogen beams will be used in hydrogen and possibly helium plasmas to exclude the device activation. Full energy (870 keV) H^0 beams have higher penetration then 1 MeV D^0 beams because of their higher velocity. The shine-through loss is reduced by impurities. Calculations of the NBST loss are carried out with the ASTRA NBI module [2] and the ACCOME code [3] taking into account the multi-step NB stopping cross sections [4] for pure helium plasma and hydrogen plasma with 3% of the ^{3}He minority and 3% of carbon (Figure 1). For this analysis the edge plasma parameters are extrapolated from the results of B2-Eirene calculations for DT operation [5-6], i.e. it is assumed that the edge electron density is saturated at $n_e(\rho=1) = 3\times10^{19}$ m^{-3} as the edge gas puffing is increased, and the core particle source is saturated at the level $S_{core} < 15$ Pa·m^{3}s^{-1}. Therefore, in the absence of another source for core fuelling and without an anomalous particle pinch one can expect a flat density

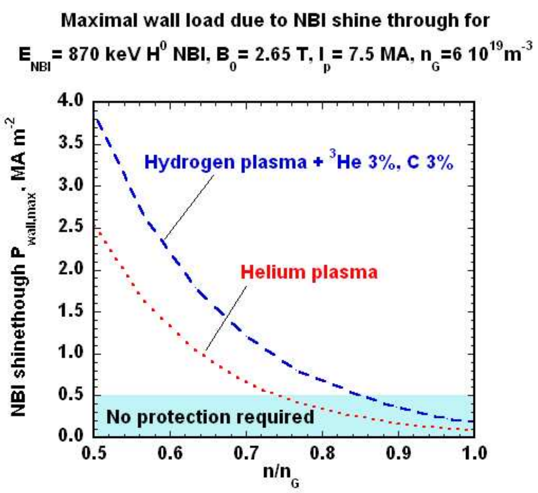

FIG. 1. NBI shine through loss for helium (dotted line) and hydrogen (dashed line) plasmas for 870 keV hydrogen NBI as a function of density for half field/half current operation B_0/I_p = 2.65T/7.5 MA.with n_G=6×10^{19}m^{-3}.

profile in helium plasmas with $n_e \sim 3\times10^{19}$ m^{-3}. This corresponds to half of the Greenwald density, $n/n_G = 0.5$ for half-current, full-bore, a = 2 m plasmas. For this low density both the ASTRA and ACCOME codes predict a shine-through loss at the level of 6.5% for helium plasma, which corresponds to a maximum power density at the wall of $P_{NB,shine} \sim 2.6$ MW/m^2 taking into account the inclination of the beam to the wall. For this high load case NBI commissioning at the full beam energy requires additional measures for wall protection. Suggested wall armouring will enable operation up to $P_{NB,shine} \sim 4$ MW/m^2. Therefore, hydrogen operation at low density $n_e \sim 3\times10^{19}$ m^{-3} is close to the NBST limit even in presence of impurities. Formally, helium operation looks more attractive for reduction of the NBST. However, in ITER plasmas there is no core helium fuelling so there is no way to control helium density.

Possible high density operation in helium plasmas may be provided by an anomalous particle pinch. Analysis of particle transport in helium experiments reveals that the pinch velocity for helium is noticeably higher than the neoclassical theory predictions [8]; i.e., it is strongly anomalous. It is known that in many experiments the anomalous pinch demonstrates a strong correlation with the safety factor q, so that: nq^{α} = const, with $\alpha = 0.5 – 1$, which corresponds to some theoretical predictions [9-10]. Taking into account the possible q values at the plasma centre and edge, q(0) ~1, $q_{95} \sim 3$, we can estimate the range of the central densities as: n(0)/n(1) = 1.7 – 3, n(0) = (5.2 – 9) $\times10^{19}$ m^{-3}. Thus, for assessment of the target plasma parameters with an anomalous pinch we use a simple approximation: $n_{He}(\rho) = (n(0)-n(1)) \times (1-\rho^2)^{0.5} + n(1)$, with boundary conditions $n(1) = 3\times10^{19}$ m^{-3} and central density in the range $n(0) = (5.2 – 9)\times10^{19}$ m^{-3}. The possibility of high density operation with helium plasmas will require further experimental and theoretical studies during the ITER construction phase.

It should be emphasized that uncertainties in the stopping cross-sections for such high beam energies, however, could either expand or reduce the operating space. Verification of theoretical predictions to reduce this uncertainty is suggested.

2.2 L-H Power Threshold

For hydrogen and helium plasmas the H-mode threshold power is expected to be higher than for deuterium plasmas [11] by factors of ~2 and ~1.42, respectively [7]. Figure 2 shows the operation regimes for H, D, and He plasmas in density and toroidal field space. In the figure, the red line represents the Greenwald parameter $n/n_G = 1$ as a function of the toroidal field with $q_{95} = 3$. When q_{95} is fixed, the Greenwald density is inversely related to the toroidal field, and the maximum operating density is a function of the toroidal field. The L-H threshold also depends on plasma density and magnetic field. Therefore, the highest operating density is determined by the maximal available power.

2.3 Power Limit

The low activation phase of ITER operation is required for commissioning the basic ITER systems including plasma control, full power heating, current drive, etc. Full power operation is also required to obtain the H-mode. This is possible only in some range of B_0 either near the full field and current or near the magnetic field and plasma current reduced to half of their design values, $B_0 = 2.65$ T, $I_p = 7.5$ MA, which is defined by the RF system frequencies: 20 MW ICRH with frequencies 40-55 MHz, 20MW ECRH with 170 GHz gyrotrons. ECRH operation is possible at the first harmonic in the range $B_0 = 2.3 - 2.8$ T, and for the second harmonic in the range $B_0 = 4.7 – 5.3$ T. ^{3}He minority heating by ICRH is possible in the range $B_0 = 3.7$-5.3 T. For helium and deuterium plasmas hydrogen minority heating is also possible in the range $B_0 = 2.5$ -3.8 T. Therefore, in the whole range $B_0 = 3.7 –$ 5.3 T the full power operation (20 MW of ECRH, 20 MW of ICRH and 33 MW of NBI) is limited to the ranges $B_0 = 4.7 – 5.3$ T (in H, D and He plasmas)and $B_0 = 2.5$ - 2.8 T (in D and He plasmas). Outside of these ranges only two of the heating methods can be applied (NBI+ECRH or NBI+ICRH), with the total available input power reduced to 53 MW. Appropriate ranges for the magnetic field are shown in the Figure 2 for the ECRH by blue and for ICRH by orange stripes.

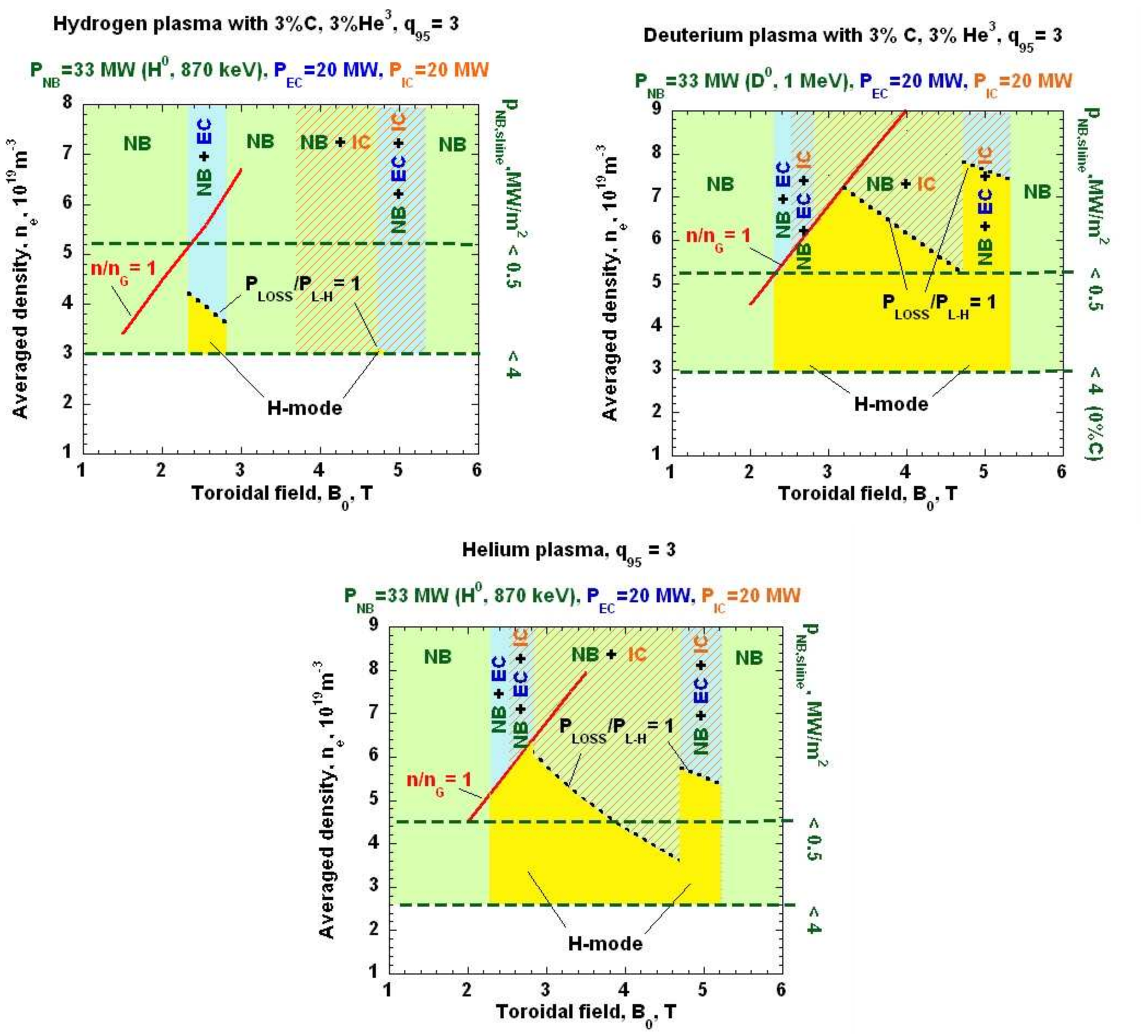

FIG. 2. Operation space in hydrogen (top left), deuterium (top right) and helium plasmas (bottom). The vertical divisions indicate the toroidal field ranges where each of the additional heating systems can operate. H–mode operation is indicated by the yellow shaded areas. The horizontal dashed lines represent limits without extra measures for wall protection ($P_{NB,shine} < 0.5$ MW/m^2) and with wall protection to provide $P_{NB,shine} < 4.0$ MW/m^2. The red line represents the Greenwald density limit, and the diagonal dashed lines represent the boundary for the L-H transition at 53MW (where EC is not available) and 73MW (where all three heating systems are available).

Figure 2 shows that with the additional wall armour H plasmas would possibly have a very small operating domain in H-mode, while for He plasmas the operating domain is greatly expanded, but still less than for D plasmas. The primary objective of He plasmas is the commissioning of the heating systems, but it also opens the possibility of investigating the L-

H transition and some H-mode physics if H-mode cannot be obtained in H plasmas. In general, at present it is not clear whether the specific H-mode issues such as ELM mitigation can be extrapolated from He to D, and DT operation. Therefore, the possibility of such extrapolation must be studied during the construction phase.

3. Assessment of Plasma Parameters in H-mode Operation

Achievement of the H-mode with type-I ELMs is an important goal of the low activation phase of ITER operation. This will enable studies of the edge pedestal parameters, demonstration of the ELM mitigation and control of high performance plasmas. In present day experiments, achievement of the Type-I ELMy mode operation requires an input power higher than the L-H power threshold, $P_{aux} = \alpha\ P_{L\text{-}H}$, with α = 1-1.3 in ASDEX-U and α > 1.4 in JET experiments. H-mode operation with Type-III ELMs, which can be reached at $P_{loss} = P_{L\text{-}H}$ has 15-20% lower confinement and different pedestal characteristics. Therefore, it is possible to expect that robust Type-I ELMy H-mode operation will be possible at $P_{loss} = 1.5\ P_{L\text{-}H}$. In general, it is possible to increase α by density reduction. However, the density reduction is limited by the shine-through density limit and the minimal density for the L-H transition [11], $n_{min,LH}$. In present tokamaks with moderate magnetic field $B_0 < 3$ T this density is close to the shine-through limit in ITER, $n_{min,L\text{-}H} \sim 3\times10^{19}\ m^{-3}$.

Simulations of plasma parameters with the 1.5D transport code ASTRA [12] were carried out for hydrogen, helium and deuterium plasmas for the full and half field/current cases at maximum available input power. Heat transport is fitted using an empirical approach [13] to provide the energy confinement time equal to that predicted by the L- or H-mode global scalings. Density is scanned to reduce the L-H power threshold to the level $P_{loss} = 1.5\ P_{L\text{-}H}$, provided the density remains above the NBST limit.

For hydrogen plasma the H-mode operating space at the full heating power of 73 MW shrinks practically to a single point at B_0 = 4.7 T, I_p = 13.3 MA at the density of the NBST limit $n_{NBST} \sim 3\times10^{19}\ m^{-3}$ (figure 2). Therefore, H-mode operation at full performance looks unlikely. For the whole range of permitted densities [$n_e = (0.3 – 1.2)\times10^{20}\ m^{-3}$ limited by the relation $n_{NBST} < n_e < n_G$] at the available flux consumption for current flat-top of 30 V·s, the maximum duration of the flat-top, $\Delta t_{FT} \sim 100$ s, corresponds to the lower density limit, $n_{NBST} \sim 3\times10^{19}\ m^{-3}$. The minimum duration, $\Delta t_{FT} \sim 50$ s, corresponds to the Greenwald limit, $n = n_G = 1.2\times10^{20}\ m^{-3}$. The normalised beta value remains rather low, β_N = 0.4 – 0.6. H-mode operation becomes possible for the half field/half current case. But even for the lowest density, $n = n_{NBST} \sim 3\times10^{19}\ m^{-3}$ for 53 MW of the input power, the power loss through separatrix remains close to the threshold, $P_{loss}/P_{L\text{-}H} \sim 1.1$. If the Type-I ELMy H-mode is possible in this case, the normalised beta, $\beta_N \sim 1.6$, can be closer to the value expected in the DT phase, and the pulse duration can reach 500 s even for 30 V·s available during the current flat-top expected for the reference 15-MA scenario. For the I_p = 7.5 MA case the duration of the H-mode can be even longer and beta could be sufficient for NTM suppression studies.

As shown in Fig. 2 (bottom) ELMy Type-I H-mode operation is possible in helium plasmas with full field/full current/full power, although the operating space is narrow. Power loss remains slightly below the desirable level, $P_{loss}/P_{L\text{-}H} \sim 1.5$, even at the lowest permitted density with additional shine-through armor, $n = n_{NBST} \sim 2.7\times10^{19}\ m^{-3}$. After the L-H transition, the shine-through limit increases to $n = n_{NBST} \sim 2.9\times10^{19}\ m^{-3}$, and the $P_{loss}/P_{L\text{-}H}$ ratio drops to ~1.2. However, if ELMy Type-I H-mode operation at this density is possible, then the duration of the current flat-top can be large, $\Delta t_{FT} > 500$ s at $\beta_N \sim 0.8$. In the case of half field/half current/full power operation, robust Type-I ELMy H-mode operation ($P_{loss}/P_{L-H} > 1.5$) will become possible at a density far above the NBST limit: $n < 5.3\text{x}10^{19}\ m^{-3}$. At this density the normalised beta can reach $\beta_N \sim 1.6$ with a duration of the current flat-top, $\Delta t_{FT} >$ 400 s, that is comparable to that expected in the DT reference inductive discharge.

For deuterium plasmas robust Type-I ELMy H-mode operation looks possible well above the NBST limit for both full and half performance cases. For B_0/I_p = 5.3 T/15 MA (full field) in L-mode $P_{loss}/P_{L-H} > 1.5$ can be reached for $n < 4.2\times10^{19}\ m^{-3}$ and after the L-H transition for $n < 3.3\times10^{19}\ m^{-3}$ with $\beta_N \sim 1.15$ and $\Delta t_{FT} > 700$ s. For B_0/I_p = 2.65 T/7.5 MA (half field) the ratio P_{loss}/P_{L-H} exceeds 1.8 in the whole range of the density scan from $n = n_{NBST}{\sim}3\times10^{19}\ m^{-3}$ to $n= n_G = 6\times10^{19}\ m^{-3}$, with β_N~2.3–2.4 and $\Delta t_{FT} >$ 1000 - 500 s for the corresponding extremal densities. Therefore, such conditions can be used for studies of the hybrid long pulse scenarios foreseen during DT operation.

As it follows from the analyses above, some of the desirable operating modes become possible only at low density, $n_{min,L\text{-}H} \sim 3\ 10^{19} m^{-3}$ close to the L-H minimum density, $n_{min,L\text{-}H}$ with moderate magnetic field, $B_0 < 3$ T [11]. Near this minimum the L-H power threshold is very uncertain [14]. Moreover, for higher magnetic field the minimum density required for the transition can be higher [15]. Therefore, during ITER construction operation near the minimum threshold density must be studied more carefully to clarify which scenarios are possible for the non-active phase.

Notice that according to B2-Eirene predictions [5, 6] the low density operation $n\sim 3\text{x}10^{19}\ m^{-3}$ can be provided by the gas puffing only, at least for hydrogen case. Therefore, ELM mitigation by resonance magnetic perturbations (RMPs) and low field side (LFS) pellet injection may be studied independently of fuelling by pellet injection.

4. Assessment of Tritium and Neutron Yield for Deuterium Phase

Characteristics of deuterium plasmas are similar to those of DT plasmas except for the amount of alpha particle heating. Therefore, the reference operation scenario (for example, with P_{fus} = 400 MW and $Q = 10$) can be simulated in this phase. In a deuterium plasma some tritium nuclei will be produced in the D-D reactions. Therefore, the addition of a small amount of tritium from an external source will not significantly change the activation level of the machine. By using limited amounts of tritium in deuterium plasmas, the integrated commissioning of cooling and tritium recycle systems is possible.

For self-consistent evaluation of the production and accumulation of tritium in the D plasma we used the ASTRA code with transport coefficients normalised to the ITERH-98(y2) confinement scaling with the normalization factor $H_{98,y2} = 1$ [13]. Plasma core contamination with a single external impurity (Be) at the level of $n_{Be}/n_e = 2$ % was assumed. The thermal particle transport equations are solved for n_e, n_{He} and n_T only. The fuel density is calculated assuming plasma quasi-neutrality, i.e., $n_D = ne - 2n_{He} - n_T - 4\ n_{Be} - \Sigma_i Z_i n_{Zi}$, where $\Sigma_i Z_i n_{Zi}$ is the sum of the suprathermal fusion product density. Possible accumulation of thermal hydrogen and ^{3}He and possible core contamination with carbon are not taken into account. Pedestal transport was adjusted to maintain the pressure gradient below the ballooning limit.

To estimate the maximum tritium and neutron yields, the tritium diffusivity was assumed to be equal to the minimal value, $D_T/\chi_{eff} = 0.3$, from the range of $D_T/\chi_{eff} = 0.3$-1.5 obtained in the DT JET experiments [16]. The duration of the current flat-top can be estimated from the volt-second consumption available in the reference inductive scenario with $I_p = 15$ MA, i.e. $\Delta\psi = 30$ V·s. Thus, the current flat top duration $\Delta t = \Delta\psi/U_{loop}$, where U_{loop} is a loop voltage, and the maximum neutron fluence is $F = \int S_{neutron}\, dt$. In the simulations, the plasma density was varied to provide safe operation in the H-mode regime with $P_{loss}/P_{LH} > 1$. The tritium consumption and 14.1 MeV neutron and alpha production due to secondary fusion reaction are taken into account: D + D= T(1.01 MeV) + p(3.03 MeV) $\rightarrow$ T(1.01 MeV) + D = He (3.52 MeV) + n(14.1 MeV). The probability of secondary fusion reactions during slowing down of 1 MeV tritium ions is calculated in the finite electron temperature, cold ion approximation [17].

Two cases, $D_T = 0$ and $D_T/\chi_{eff} = 0.3$, were considered at the plasma densities corresponding to the L-H limit. In both cases, the plasma loop voltage, U_{loop}, and total neutron yield of 2.45 MeV DD neutrons $N_{n2.45}$, are similar: $U_{loop} \approx 60$ mV and $N_{n2.45} \approx 9.1\times10^{20}$. On the other hand, the total tritium production during the current flat-top, N_T, and the total neutron yield of 14.1 MeV D-T neutrons, $N_{n14.1}$, are different: $N_T = 7.6\times10^{20}$ and $N_{n14.1} = 1.1\times10^{20}$ in the case $D_T/\chi_{eff} = 0.3$, and $N_T = 1.9\times10^{20}$ and $N_{n14.1} = 6.8\times10^{20}$ in the ideal case $D_T = 0$.

The maximum low energy neutron yield, $N_{n2.45}$, produced in the reaction D + D = n (2.45 MeV)+ ^{3}He(0.82 MeV) increases with decreasing plasma density (which may be used to prolong the pulse). In the ideal case ($D_T = 0$) $N_{n2.45}$ approaches ~ 10^{21}. The sum of the tritium yield and the fast neutron yield is close to the low energy neutron yield, $N_T + N_{n14.1} \approx N_{n2.45}$. The ratio $N_T/N_{n14.1}$ depends on details of the tritium transport. At $D_T = 0$ the tritium yield should saturate at the level $N_T \sim N_D\, S_{DD1}/S_{DT}$, where S_{DD1} and S_{DT} are the rates of the reactions D + D = T(1.01 MeV) + p(3.03 MeV) $\rightarrow$ T(1.01 MeV) + D = He (3.52 MeV) + n (14.1 MeV), and N_D is the total deuterium content in the plasma.

6. Summary and conclusions

The H-mode operating space in ITER is assumed to be restricted by the Greenwald density, $n/n_G < 1$, by power loss across the separatrix exceeding the L-H power threshold, $P_{loss}/P_{L\text{-}H} > 1$, and by the NBI shine through (NBST) limit, $P_{NB,shine} < 4$ MW/m^2. With these assumptions, hydrogen operation at low density ($n_e \sim 3\times10^{19}$m^{-3}) required for H-mode access is close to the NBST limit even in the presence of impurities. Type-I ELMy H-mode operation therefore looks unlikely during the hydrogen phase for 73 MW of input power due to the unfavourable mass dependence of L-H power threshold.

Helium operation looks more attractive for H-mode access because of the reduction of the NBST and possibly reduced threshold power at a given density. But the possibility of high density operation in helium plasmas without core fuelling requires the presence of an anomalous particle pinch. Further supporting research is required to provide a more solid basis for such operation. The possibility of extrapolating of experience obtained with helium H-mode plasmas to D and DT operation should be studied during the construction phase. Uncertainties in the stopping cross-sections for high beam energies could either expand or reduce the operating space for all low activation scenarios examined here. Experimental verification of theoretical predictions would reduce this uncertainty.

During hydrogen operation, transition to robust Type-I ELMy operation appears unlikely. Good H-mode confinement expected with $P_{NB} + P_{EC} + P_{IC} > 1.5\ P_{L-H}$ is more likely for helium and deuterium cases. In He and D plasmas the normalised beta and current flat-top duration can simulate conditions expected in the reference DT inductive and long pulse scenarios.

Some of the desirable operating modes become possible only at low density, $n_{min,L-H} \sim 3\ 10^{19} m^{-3}$, which is close to the density where the threshold power is expected to be minimum. Near this minimum the L-H power threshold is very uncertain. Therefore, during ITER construction supporting research on operation at densities near the minimum power threshold should be studied to clarify which scenarios are best during non-active phase for plasma system commissioning and ITER licensing.

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