Assessment of Plasma Parameters for Low Activation Phase of ITER Operation

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Abstract
Assessment of plasma parameters is carried out for low activation phase 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 foreseen for ITER with hydrogen and deuterium neutral beam injection (NBI). Here we assess the possible operating domain for providing safe operation and possible schemes 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 the 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 the H-mode operation is marginal for hydrogen operation. A good H-mode confinement expected at $P_{NB} + P_{EC} + P_{IC} > 1.5\ P_{L-H}$ is more likely for helium and deuterium cases. It is obtained, that plasma parameters for full power/half field/half current operation can be similar to those required for DT long pulse operation. Preliminary assessment of the upper limit of tritium and neutron yield for deuterium phase of ITER operation is carried out.

1. Introduction

The operation sequence of 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 operation 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. Under H-mode conditions the divertor performance can only be examined 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 in H or He plasmas. 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 the full DT operation. 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. At 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 by assessment of plasma parameters foreseen for ITER operation at non-active and deuterium operation phases.

2. Operational Space for Low Activation Phase of ITER

Following a period of Integrated Commissioning, demonstrating the readiness of major tokamak subsystems, the demonstration of plasma operation will define the completion of the Construction Phase. First Plasma demonstrated in 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 which allows plasma breakdown to be achieved.

During the following non-active phase, the 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, most 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 operational 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 helium plasmas to exclude the device activation. Full energy (870keV) H^0 beams have higher penetration then 1MeV D^0 beams because of their higher velocity. The shine-through loss is reduced by impurities.

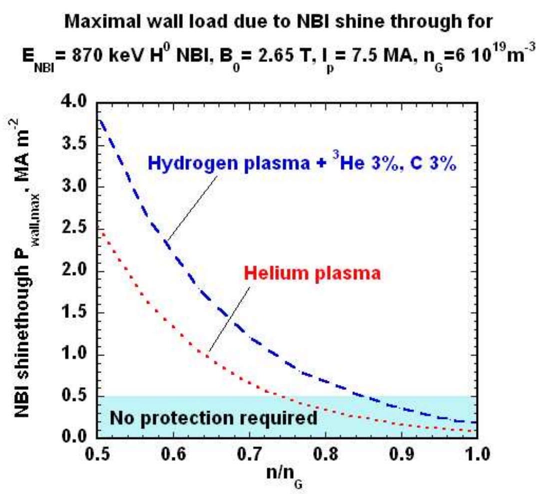

FIG. 1. NBI shine through loss for helium (dotted line) and hydrogen (dashed line) plasmas for 870 keV hydrogen NBI as a function of density for half field/half current operation B_0/I_p = 2.65T/7.5 MA.with n_G=6x10^{19}m^{-3}.

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 (Fugure 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 anomalous particle pinch one can expect a flat density 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 $n_e \sim 3x10^{19}m^{-3}$ is close to the NBST limit even in presence of impurities. Formally helium operation looks more attractive for reduction of the NBST. Unfortunately, in the ITER plasmas in absence of the core helium fuelling there is no way to control helium density especially if the edge density saturates at the low level $n_e \sim 3x10^{19}m^{-3}$ as it is predicted by B2-Eirene simulations for DT plasmas.

Possible high density operation in the helium plasmas can be expected in presence of the 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]. In these simulations we do not take into account the auxiliary currents produced by ECCD and ICCD. Nevertheless, taking into account the possible q values at the plasma centre and edge, $q(0) \sim 1$, $q_{95} \sim 3$, we can estimate the range of the central densities as: $n(0)/n(1) = 1.7 - 3$, $n(0) = (5.2 - 9)x10^{19}m^{-3}$. Thus, for assessment of the target plasma parameters with an anomalous pinch we use a simple approximation: $n_{He}(\rho) = (n(0)-n(1))\ x\ (1-\rho^2)^{0.5} + n(1)$, with boundary conditions $n(1) = 3x10^{19}m^{-3}$ and central density in the range $n(0) = (5.2 - 9)\times 10^{19}m^{-3}$. Possibility of high density operation with helium plasmas will require further experimental and theoretical studies at 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 higher then for deuterium plasma [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 operation 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. It 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 defined by chosen parameters of the RF heating: 20MW ICRH with frequencies 40-55 MHz, 20MW ECRH with 170GHz gyrotrons. The ECRH 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 by the ranges B_0 = 3.7 – 5.3 T and B_0 = 2.5 - 2.8 T (in D and He plasmas). In other areas only two of the heating methods can be applied (NBI+ECRH or NBI+ICRH). This means that the total available input power in these ranges reduces to 53 MW. Appropriate ranges of the magnetic field are shown in the Figure 2 for the ECRH by blue and for ICRH by orange stripes.

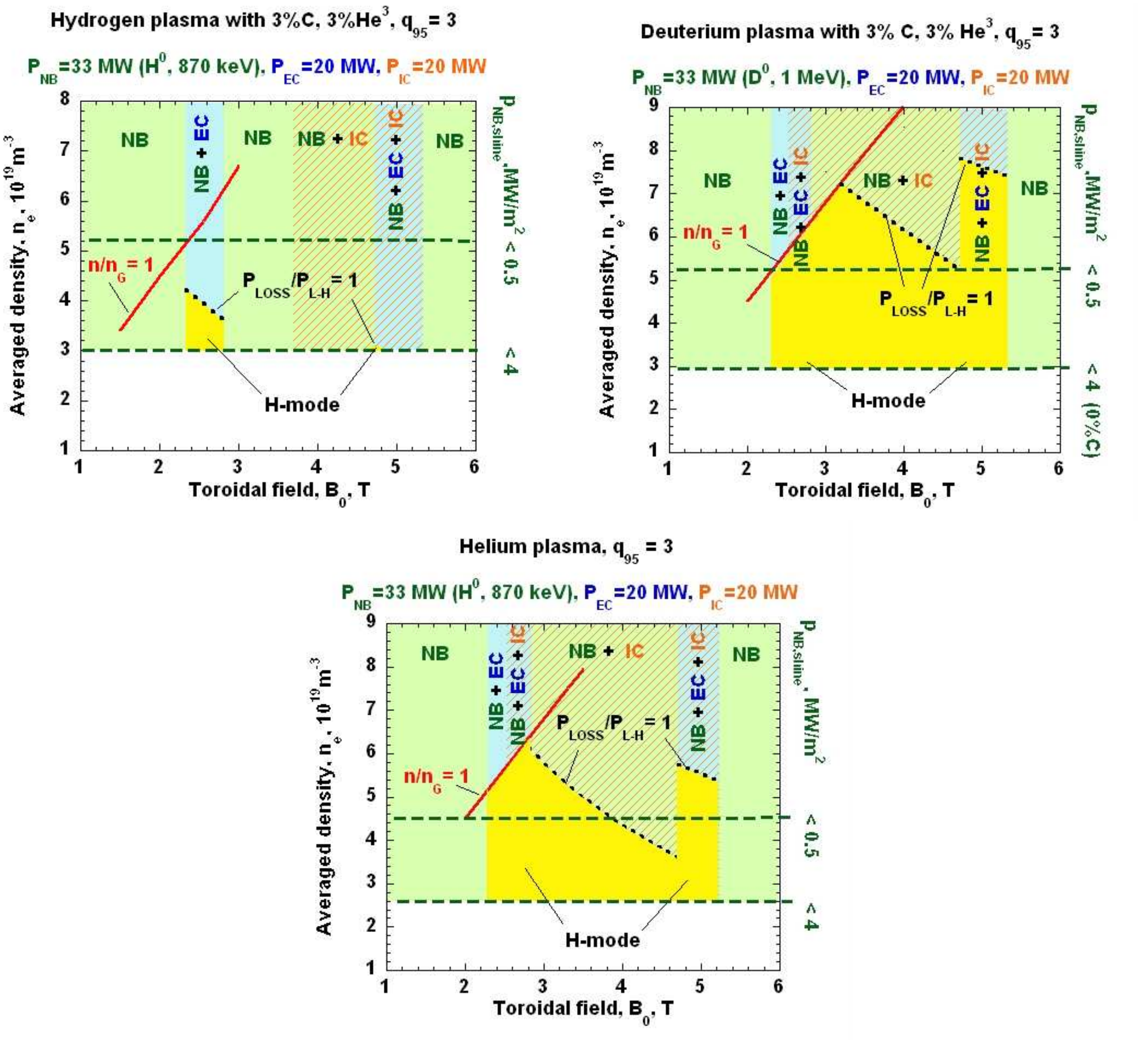

FIG. 2. Operation space in hydrogen (top left), deuterium (top right) and helium plasmas (bottom). The vertical divisions indicate the toroidal field ranges where each of the additional heating systems can operate. H–mode operation is indicated by the yellow shaded areas. The horizontal dashed lines 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 the H plasmas would possibly have very narrow operating domains 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, etc. can be extrapolated 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 ELM-type-I H-mode requires the input power higher than the L-H power threshold, $P_{aux} = \alpha P_{L-H}$, with α = 1-1.3 in ASDEX-U and $\alpha > 1.4$ in JET experiments. The H-mode operation with type-III ELMs, which can be reached at $P_{loss} = P_{L-H}$ has 15-20% lower confinement and different pedestal characteristics. Therefore, it is possible to expect that robust ELM-type-I H-mode operation will be possible at $P_{loss} = 1.5\ P_{L-H}$. In general, it is possible to increase α by density reduction. However, the density reduction is limited by shine-through density limit and the minimal density for 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, $n_{min,L-H} \sim 3\ 10^{19} m^{-3}$.

Simulations of plasma parameters with 1.5D transport code generated by 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 scalings. Density is scanned to reduce the L-H power threshold to the level $P_{loss} = 1.5\ P_{L-H}$, provided the density remains above the NBST limit.

For hydrogen plasma the H-mode operation space at 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 NBST limit $n_{NBST} \sim 3\times10^{19}\ m^{-3}$ (figure 2). Therefore, the 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 Vs, the maximum duration of the flat-top, $\Delta t_{FT} \sim 100$ s, corresponds to the lower density limit, $n_{NBST} \sim 3x10^{19}\ m^{-3}$, and the minimum duration, $\Delta t_{FT} \sim 50$ s corresponds to the Greenwald limit, $n = n_G = 1.2\ 10^{20}\ m^{-3}$. The normalised beta value remains rather low, $\beta_N = 0.4 - 0.6$. 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, $P_{loss}/P_{L-H} \sim 1.1$. If the ELM type-I H-mode will be possible in this case, the normalised beta, $\beta_N \sim 1.6$, can be closer to the value, expected in the DT phase and pulse duration can reach 500 s even for 30 Vs available for the current flat-top expected for the reference 15-MA scenario. For I_p = 7.5 MA case the duration of the H-mode can be even longer and beta could be sufficient for the NTMs suppression studies.

As shown in Fig.2 (bottom) the ELM-type-I H-mode operation in the helium plasma at the full field/full current/full power case although the operation space is narrow. Power loss remains slightly below the desirable level, $P_{loss}/P_{L-H} \sim 1.5$, even at the lowest permitted density, $n = n_{NBST} \sim 2.7\times10^{19}\,m^{-3}$. 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-H} ratio drops to ~1.2. Anyway if the ELM-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 the half field/half current/full power operation the robust ELM-type-I H-mode operation ($P_{loss}/P_{L-H} > 1.5$) will become possible at density far above the NBST limit: $n < 5.3\text{x}10^{19}\,m^{-3}$. At the maximum density, $n = 5.3\times10^{19}\,m^{-3}$, the normalised beta can reach $\beta_N \sim 1.6$ with similar duration of the current flat-top, $\Delta t_{FT} > 400$ s expected in the DT reference inductive discharge.

For deuterium plasmas the robust ELM-type-I 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 in the L-mode $P_{loss}/P_{L-H} > 1.5$ can be reached for $n < 4.2\ 10^{19}\,m^{-3}$ and after the L-H transition for $n < 3.3\ 10^{19}\,m^{-3}$ with $\beta_N \sim 1.15$ and $\Delta t_{FT} > 700$ s. For B_0/I_p = 2.65 T/7.5 MA the ratio P_{loss}/P_{L-H} exceeds 1.8 in the whole range of the density scan from $n = n_{NBST} \sim 3\ 10^{19}\,m^{-3}$ to $n = n_G = 6\ 10^{19}\,m^{-3}$, with $\beta_N \sim 2.3 – 2.4$ and $\Delta t_{FT} > 1000 - 500$ s for the correspondent extremal densities. Therefore, such scenarios can be used for studies of the hybrid long pulse scenarios foreseen for DT operation.

As it follows from the analyses above some of the desirable operational modes become possible only in the low density area, $n_{min,L-H} \sim 3\ 10^{19} m^{-3}$ close to the L-H minimum density, $n_{min,L-H}$ 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 transition can be higher [15]. Therefore, at the construction operation near the minimal threshold density must be studied more carefully to clarify which scenarios can be possible for non-active phase for plasma system commissioning and ITER licensing.

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 with the gas puffing only, at least for hydrogen case. Therefore, ELM mitigation by resonance magnetic perturbations (RMPs) and low field side (LFS) pellet injection can be studied independently on the fuelling 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 amount of the alpha particle heating. Therefore, the reference operation scenario (for example, with P_{fus} =400MW 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, an 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]. A 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. Fuel density is calculated assuming plasma quasineutrality, 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 fitted to provide 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 $D_T/\chi_{eff} = 0.3\text{-}1.5$ obtained in the DT JET experiments [16]. Duration of the current flat top can be estimated from the volt-second consumption available for the current flat at the reference inductive scenario with $I_p = 15$ MA, i.e. $\Delta\psi = 30$ Vs. Thus, current flat top duration $\Delta t = \Delta\psi/U_{loop}$, where U_{loop} is a loop voltage, and the maximum neutron fluence is $F = \int S_{neutron}\, dt$. Plasma density was varied to provide safe operation in the H-mode regime with $P_{loss}/P_{LH} > 1$. Tritium sink 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). A probability of the secondary fusion reaction during slowing down of a 1 MeV tritium ion is calculated in 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 $Nn_{2.45}$,, are similar: $U_{loop} \approx 60$ mV and $Nn_{2.45} \approx 9.1\times10^{20}$. Meanwhile, the total tritium production during the current flat-top, N_T, and the total neutron yield of 14.1 MeV D-T neutrons, $Nn_{14.1}$, are different: $N_T = 7.6\times10^{20}$ and $Nn_{14.1} = 1.1\times10^{20}$ in the case $D_T/\chi_{eff} = 0.3$, and $N_T = 1.9\times10^{20}$ and $Nn_{14} = 6.8\times10^{20}$ in the ideal case $D_T = 0$.

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 decrease of plasma density due to possibility of pulse prolongation. 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, $N_T + N_{n14} \approx N_{n2.45}$. The ratio N_T/N_{n14} 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 volume.

6. Summary and conclusions

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, $P_{NB,shine} < 4$ MW/m^2.

Hydrogen operation at low density $n_e \sim 3\text{x}10^{19}\text{m}^{-3}$, required for the H-mode access is close to the NBST limit even in the presence of impurities.

The H-mode operation with ELM-type-I looks unlikely at the hydrogen phase for 73 MW of the input power due to unfavourable mass dependence of L-H power threshold.

Helium operation looks more attractive for reduction of the NBST. But the possibility of high density operation in helium plasmas without core fuelling suggests the presence of anomalous particle pinch. Further experimental and computational studies are required to provide the solid basis for such suggestions. The possibility of extrapolation of experience obtained with helium H-mode plasmas to D and DT operation has to be studied during the construction phase.

Uncertainties in the stopping cross-sections for high beam energies ~1 MeV, could either expand or reduce the operating space. Experimental verification of theoretical predictions is required to reduce this uncertainty.

During hydrogen operation transition to the robust ELM-type-I operation looks difficult. A 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. 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.

The ELM mitigation by resonance magnetic perturbations and low field side pellet injection can be studied independently of the fuelling pellet injection for low densities with plasma fuelling by gas puffing.

Some of the desirable operational modes become possible only in the low density area, $n_{min,L\text{-}H} \sim 3\ 10^{19} m^{-3}$ close to the L-H minimal density, $n_{min,L\text{-}H}$. Near this minimum the L-H power threshold is very uncertain. Therefore, at the construction operation near the minimal threshold density must be studied more carefully to clarify which scenarios can be possible for non-active phase for plasma system commissioning and ITER licensing.

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