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Modelling of plasma performance and transient density behaviour in the H-mode access for ITER gas fuelled scenarios

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Abstract

ITER operations require effective fuelling of the core plasma for conditions in which neutral dynamics through the scrape-off layer is expected to significantly affect the efficiency of gas penetration. On the basis of the previous analysis for stationary conditions, pellets are foreseen to provide core fuelling of high-Q DT scenarios. In this paper we present a modelling study of the gas fuelling efficiency in ITER providing an estimate of the maximum plasma density achievable with gas fuelling only in various DT reference scenarios. Dynamical integrated core-edge plasma simulations for various phases of ITER DT H-mode discharges have been carried out with the JINTRAC suite of codes. Simulations of the L-mode phase show that divertor detachment sets the maximum density achievable at the separatrix by deuterium-tritium gas fuelling. The maximum volume-average density is achieved for 15 MA/5.3 T and it is close to the requirement for stationary application of neutral beam injection heating at full power (16.5 MW per injector) and ion energy (1 MeV) compatible with acceptable shine-through loads on the first wall. The achievable density in gas fuelled H-modes is typically a factor of 2–3 larger than in L-modes. The fusion performance of gas fuelled H-modes at 15 MA is typically found to be moderately high ($Q \sim 6$ –8). Sensitivity of the above predictions to modelling assumptions and validation of the physics models are discussed.

Keywords: ITER, fuelling, integrated modelling, scenario, transport, SOL, detachment

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

1. Introduction

In order to assess the fuelling requirements in stationary L-mode and H-mode operations as well as the plasma behaviour in transients, integrated core-SOL plasma modelling has been carried out for conditions expected in ITER DT scenarios

with plasma current 7.5 MA and toroidal field 2.65 T and plasma currents 15 MA, 10 MA, 7.5 MA, 5 MA and toroidal field 5.3 T. The latter correspond to time slices during the current ramp-up/down and flat-top operation in the ITER high Q_{DT} baseline scenario while the former is representative of low toroidal field H-mode operation in ITER. The modelling

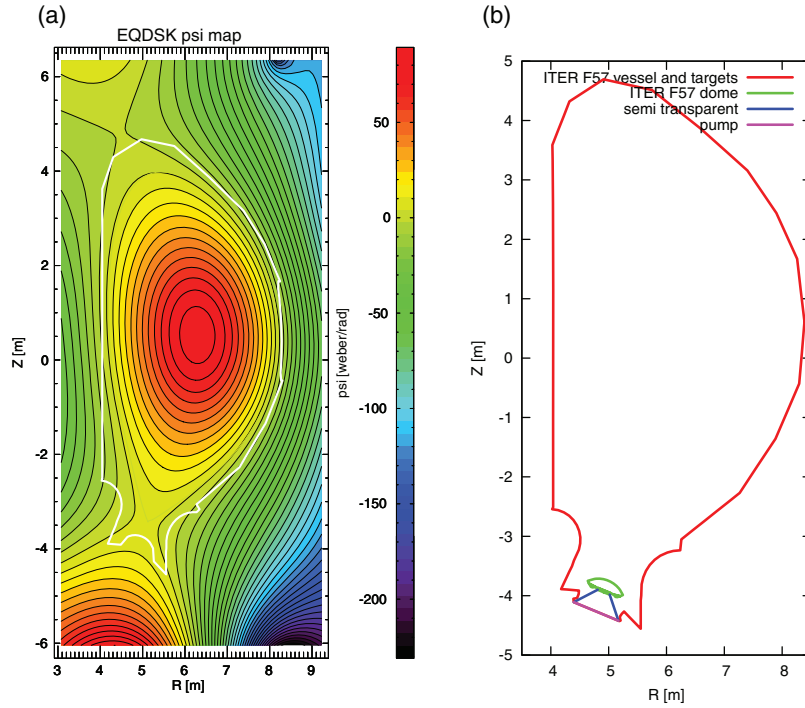


Figure 1. (a) Ψ map of the equilibrium for the 15 MA / 5.3 T scenario. (b) Vessel and target structure with dome, semi-transparent surfaces and pump (dashed pink line).

includes: core-particle sources from both gas fuelled neutrals and those recycled at the main-chamber wall and divertor-targets along with their ionization in the SOL, divertor and plasma core and their pumping by the divertor cryo-pumps. The simulations consider all the phases of the H-mode scenarios from the L-mode phase, L-H transition, initial ELM-free H-mode and the stationary H-mode phase with controlled ELMs. The JINTRAC [1] suite of codes has been used for this analysis. As a first step the edge plasma has been modelled separately for a range of conditions with the EDGE2D/EIRENE code (see references in [1] for each JINTRAC module) and previous results obtained with SOLPS have been confirmed [2, 3]; namely that, due to the size of the machine, the magnitude of the core plasma neutral source in ITER is typically smaller than 1–2% of the total recycling flux even for conditions far away from the semi-detached divertor operation and for plasma temperatures at the divertor in excess of 10 eV [4]. Full plasma simulations have been carried out with JINTRAC in the integrated mode with both the Bohm-gyro-Bohm [5] core-transport model and with GLF23 [6] including two impurity species (Be and Ne). The latter species has been introduced in order to evaluate the extrinsic impurity seeding requirements to maintain the peak divertor power flux under 10 MW m^{-2} , which is the ITER divertor design criterion for stationary power fluxes.

2. Simulation setup

For the EDGE2D grid generation the equilibrium map of the 15 MA/5.3 T ITER reference case has been adopted, with a resolution of 65×129 (see figure 1(a)). The poloidal

magnetic field is obtained from the derivatives of the Ψ -map. The toroidal magnetic field is represented by $B_{\text{tor}}(R) = B_0 R_0 / R$ where $B_0 = 5.3 \text{ T}$ and $R_0 = 6.2 \text{ m}$. A closed structure to confine the neutral particles has been defined, consisting of the contour of the vacuum vessel and the plasma facing components (red line, figure 1(b)). The divertor dome (green line, figure 1(b)) is also included and the structural support of the dome is represented as semi-transparent surfaces (blue lines, figure 1(b)) with a transparency of 0.5 from both sides. The pump is represented by a pumping surface (dashed pink line, figure 1(b)) between the dome support structures with an albedo of 0.9928. The value of Ψ has been rescaled for the equilibrium at lower currents keeping the shape of the flux surfaces in the SOL. The material for all wall surfaces is assumed to be beryllium and different surface temperatures are assumed for different wall regions. Thus, the sputtering of tungsten from the divertor is neglected as the tungsten transport is not modelled. A grid with a resolution of either $106 \text{ rows} \times 34 \text{ rings}$ (for high current H-mode simulations) or $84 \text{ rows} \times 24 \text{ rings}$ has been adopted for the simulations with a low level of compression towards the targets, as a compromise between execution time and grid resolution, shown in figure 2. The EIRENE code (coupled to EDGE2D) is used to determine the neutral distribution of all particles, main plasma and impurities, in the SOL. Two impurity species have been included in the simulations: Be as the non-recycling intrinsic impurity and Ne as recycling puffed species. In these simulations the main fuel is puffed from the top of the machine consistent with the ITER gas injection geometry with a variable puff rate. Plasma rotation is set to zero at the separatrix and the momentum equation in the core is not solved. The reference simulations are carried out with transport coefficients in the

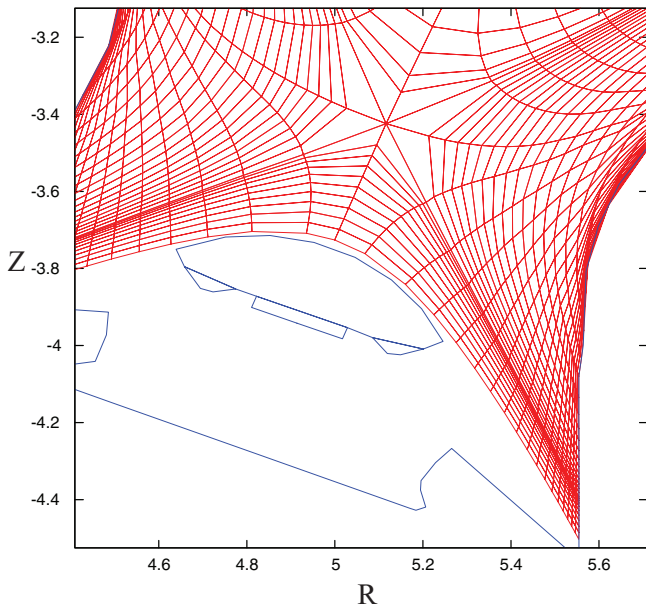


Figure 2. Details of the divertor region grid used in the simulations.

SOL determined by extrapolation from core values to fixed boundary values (at the last SOL surface) which are setup to $1 \text{ m}^2 \text{ s}^{-1}$ for the heat conductivities and $0.3 \text{ m}^2 \text{ s}^{-1}$ for the diffusivities of main ions and impurities. No SOL particle pinch velocity is included and SOL drifts are not included either. The maximum time step in EDGE2D is 10^{-5} s . For the neutral dynamics, 20000 Monte Carlo particles have been used in EIRENE. A regularly spaced radial grid of 300 points and maximum time step of 10^{-5} s (same as for EDGE2D) have been adopted in JETTO for all simulations in this report. The L-mode simulations have variable ECRH heating while the H-mode simulations are run with variable NBI (neutral beam power deposition profiles have been calculated with the PENCIL code) and ICRH power. The ECRH, ICRH power deposition profiles have been modelled by a Gaussian peaked at the location of the resonance. For the prediction of anomalous heat, main ion and impurity particle transport in the core, we have adopted the Bohm/gyroBohm (D_{BgB} , χ_{BgB}) model in addition to NCLASS for the neoclassical transport. In the case of DT, the transport is calculated separately for each species. However, the difference in transport is expected to be minimal as there is no mass dependence for Bohm diffusion and only a very weak ($\sim m^{1/2}$) dependence for gyroBohm. Two different sets of coefficients have been adopted for the BgB transport model in the L-mode and in H-mode simulations. In particular, the BgB model used in the H-mode simulations has been determined from the benchmark with GLF23 for the same plasma conditions. The model includes an inwards pinch $v(\rho) = 0.5 \cdot D_{\text{BgB}}(\rho) \cdot S(\rho)^2 / (dV/d\rho \cdot V(\rho_{\text{sep}}))$, where S is the flux surface area and V is the volume enclosed. In the H-mode simulations a fixed-width transport barrier model has been chosen. The barrier width has been fixed to $\Delta\rho_{\text{norm, ETB}} \sim 0.05\text{--}6$, corresponding to $\sim 6\text{--}7 \text{ cm}$ at the outer mid-plane. Transport in the barrier region is approximately reduced to the ion neoclassical level for all transport channels as a lower limit. ELMs have been modelled with the ‘Continuous ELM

model’: when the normalised pressure gradient in the ETB exceeds the imposed alpha critical value, the neoclassical transport is increased in the barrier region gradually in proportion to the difference between the gradient and the critical value [7]. In this manner the normalised ETB pressure gradient is clamped to the value determined by the alpha critical, in contrast to the discrete ELM model in which the ETB pressure gradient slowly builds up and then collapses at the onset of an ELM [8]. Radiation is calculated self-consistently from the distribution of impurity ionization states. It should be noted that the radiation due to W is not taken into account, as the transport of W impurities has not been included in these simulations. Nuclear fusion reactions are included as a heat source in all simulations along with sawteeth using either the ‘Continuous Sawtooth model’ (analogous to the continuous ELM model, see above) for the reconnection in the gas flux scans or the Kadomtsev sawtooth reconnection model with user defined minimum sawtooth crash intervals. The transport of He ash is not included.

3. Fuelling in the L-mode phase

Using the JINTRAC settings described in section 2, simulations for the 15 MA, 10 MA and 5 MA L-mode plasmas at 5.3 T have been run adopting various levels of ECRH heating and gas fuelling rates. In this paper we concentrate on the results of the 15 MA/5.3 T plasmas with 20 MW ECRH and varying the rate of D injection from $3.0 \times 10^{21} \text{ s}^{-1}$ up until divertor detachment is found. Detachment is assumed to be reached when the divertor target temperature gets close to 1 eV on an extended region of more than 20 cm from the strike point. The maximum gas rate sustainable (i.e. before detachment) is found to be $1.6 \times 10^{22} \text{ s}^{-1}$. Figure 3 shows the profiles of the neutral source, electron density and electron temperature after simulating 8 s of plasma dynamics for three levels of fuelling: $1.0 \times 10^{22} \text{ s}^{-1}$, $1.3 \times 10^{22} \text{ s}^{-1}$ and $1.6 \times 10^{22} \text{ s}^{-1}$. The density profile saturates at a peak density of around $2.9 \times 10^{19} \text{ m}^{-3}$ (volume average density of $\sim 2.2 \cdot 10^{19} \text{ m}^{-3}$) when the gas puff increases above $1.3 \times 10^{22} \text{ s}^{-1}$ together with the onset of complete divertor detachment. Hence, higher puff rates do not result in higher core densities than already obtained at this level, but rather only push the plasma closer towards or even into detachment with very small change in density being achieved, which is in line with previous results for the separatrix density in ITER [2, 3]. The profiles of the particle source in the SOL-midplane near the separatrix for the three different puff rates is plotted in figure 4 showing the saturation of the neutral source in the SOL for gas fuelling rates larger than $1.3 \cdot 10^{22} \text{ s}^{-1}$. Figure 5 shows the impact of the increased puff-rate on the divertor temperature and power flux and the onset of plasma detachment at the divertor. For the reference choice of the values and radial profiles of the transport coefficient in the SOL for the 15 MA scenario, the maximum volume averaged density that can be achieved by gas fuelling is thus below the value compatible with acceptable shine-through loads for stationary application of NBI heating at full power (16.5 MW per injector) and ion energy (1 MeV) in ITER ($\sim 2.5\text{--}3.0 \cdot 10^{19}$

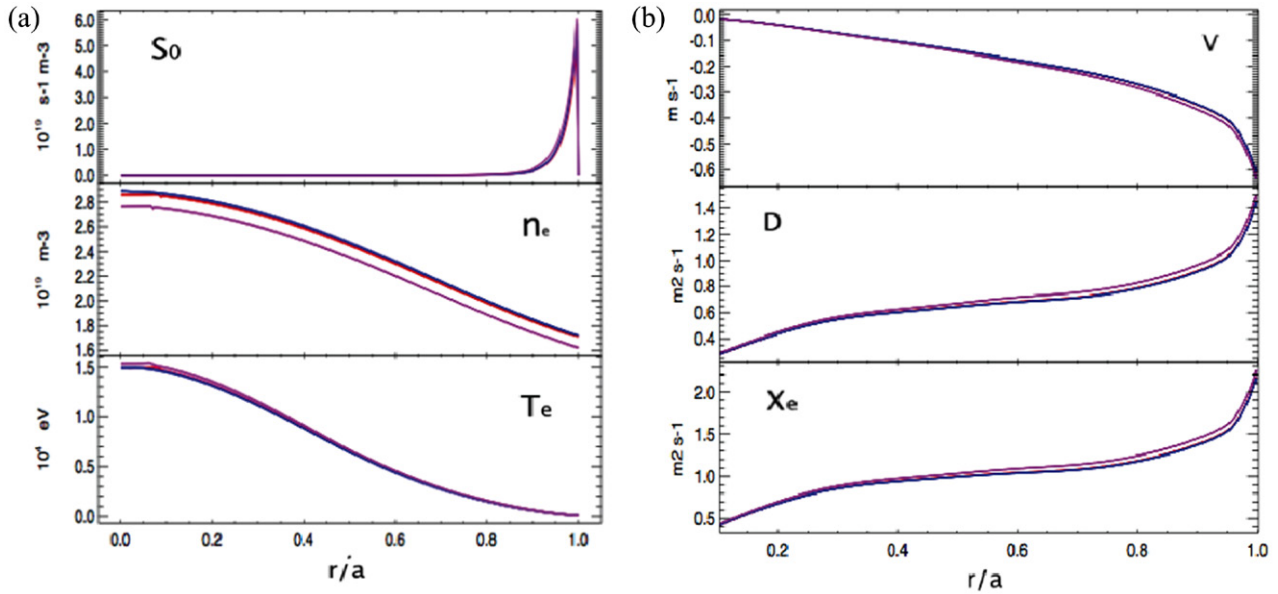


Figure 3. (a) From top to bottom, neutral source, electron density and temperature profiles versus normalised radius at 8 s for 3 different levels of fuelling: 1.0 (magenta), 1.3 (red), $1.6 \times 10^{22} \text{ s}^{-1}$ (blue) for 15 MA/5.3 T ITER L-mode. Note how the central density is saturating at $2.9 \times 10^{19} \text{ m}^{-3}$ for higher puff rates. (b) From top to bottom, pinch velocity, diffusion coefficient, and electron heat conductivity profiles.

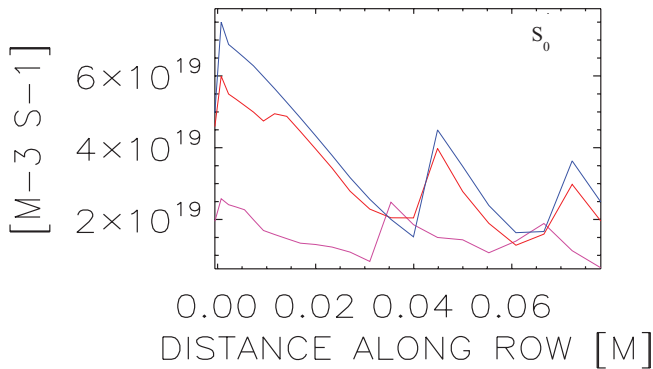


Figure 4. Radial profiles of the electron source in the SOL region near the separatrix at midplane for three different levels of fuelling: 1.0 (magenta), 1.3 (red), $1.6 \times 10^{22} \text{ s}^{-1}$ (blue) for 15 MA/5.3 T ITER L-mode. The distance along the row is the distance from the separatrix (located at row 0.0) on a computational line perpendicular to the magnetic field (in the 'radial' direction), see the EDGE2D grid in figure 2.

m^{-3}) [9]. For lower values of the plasma current such as 10 MA and 5 MA the maximum achievable density with gas-fuelling only and the reference choice for SOL transport coefficient profiles is typically $1.1 \cdot 10^{19} \text{ m}^{-3}$ for 5 MA and $1.9 \cdot 10^{19} \text{ m}^{-3}$ for 10 MA plasmas with 20 MW of ECRH.

In order to assess the impact of the modelling assumptions on the achievable density for gas fuelling L-modes in ITER we have tested different levels of SOL transport. Results of the simulations performed for 15 MA plasmas (see figures 6 and 7) show that the achievable density by gas fuelling depends on the value of transport assumed in the SOL. Thus for values of the particle diffusivity in the SOL under $0.3 \text{ m}^2 \text{ s}^{-1}$ and of the thermal conductivity under $1 \text{ m}^2 \text{ s}^{-1}$ (similar to levels of effective transport coefficients reported in Alcator C-mod L-mode experiments [10]), it is possible to achieve volume averaged densities above $2.5\text{--}3.0 \times 10^{19} \text{ m}^{-3}$ compatible with

the stationary application of NBI heating at full power and ion energy. This density level taken is thus taken as the initial condition for the L-H transition study presented in the next section.

The studies carried out so far indicate that fuelling of L-mode plasmas in ITER to average densities much higher than $\sim 3.0 \cdot 10^{19} \text{ m}^{-3}$ with gas fuelling only might not be possible, particularly for lower plasma currents in ITER. Therefore studies are being carried out to study the fuelling of such L-mode plasmas with pellets which is expected to circumvent this limit. In addition, studies on the optimization of the NBI power waveform to access the H-mode in lower current plasmas, so that the plasma density evolution after the H-mode transition (see section 4) ensures acceptable shine-through loads on the first wall panels during this phase, are also in progress.

Obviously, the precise value of the maximum achievable density obtained by gas fuelling in ITER L-modes in the studies presented here depends on modelling assumptions. In this respect a detailed quantitative validation of the integrated modelling approach applied to ITER with experimental results from gas fuelled L-mode density scans in present experiments is required before final conclusions for ITER can be drawn.

4. L-H transition

Modelling of the L-H transition for 15 MA/5.3 T DT plasmas has been carried out. In the simulations presented in this paper additional heating to a level of 53 MW is applied at 85 s and the H-mode transition is modelled to occur when the loss power exceeds the H-mode power threshold (P_{L-H}) threshold (here the scaling in [11] for the H-mode threshold has been assumed). Following this the transport coefficients in the barrier region are decreased as described in section 3. Figure 8

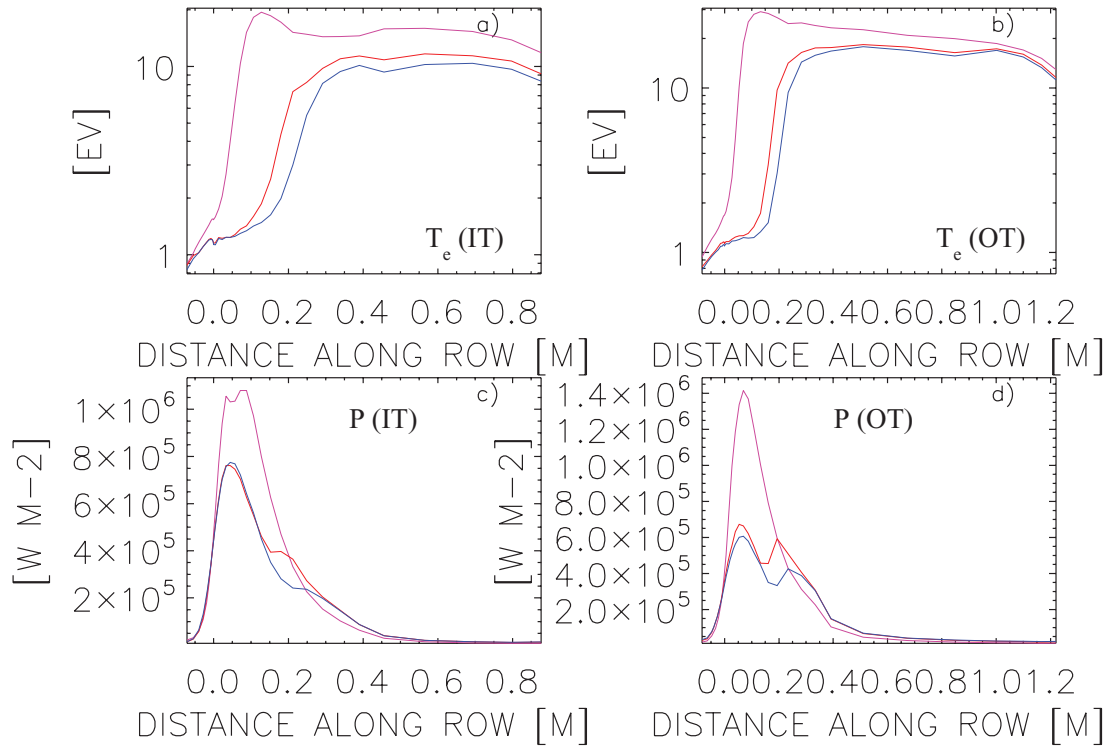


Figure 5. Profiles of the ion temperature (top) and power flux (bottom) at the inner (left) and outer divertor (right) plates respectively for three different levels of fuelling: 1.0 (magenta), 1.3 (red), $1.6 \times 10^{22} \text{ s}^{-1}$ (blue) for 15 MA/5.3 T ITER L-mode. The distance along the row is the distance from the separatrix (located at row 0.0) on a computational line parallel to the divertor plates in the ‘radial’ direction, see the EDGE2D grid in figure 2.

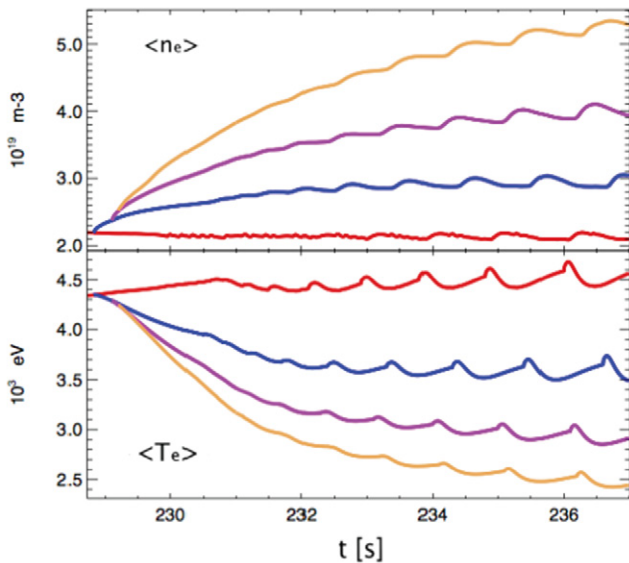


Figure 6. Time evolution of the volume averaged density (top) and temperature (bottom) for decreasing values of SOL diffusivities and constant puff rates. The radial profiles of the ion/electron thermal conductivities and the particle diffusivities are plotted in figure 7 with the same color code. The magenta trace has been obtained with SOL diffusivity $0.15 \text{ m}^2 \text{ s}^{-1}$ and thermal conductivity $0.5 \text{ m}^2 \text{ s}^{-1}$.

shows the time evolution of a set of selected plasma parameters for a L-H transition in a 15 MA/5.3 T DT plasma fuelled by gas-fuelling only. Initially, the plasma density increases on a very fast timescale ($<100 \text{ ms}$) due to the change of edge transport after access to the H-mode. The change of edge transport

in this phase leads to a reduction of the edge particle/power outflux and to an increased transparency of the edge plasma to neutrals during this phase which causes the fast density rise. However, after this initial phase the plasma density builds up at a much slower rate due to the neutral source and the low value of particle diffusion in the pedestal (close to neoclassical) and finally saturates at values of $\langle n_e \rangle \sim 6.5 \times 10^{19} \text{ m}^{-3}$ (with the onset of ELMs) even for the largest gas-fuelling rates achievable in ITER. Following the L-H transition, the plasma temperature also rises until its value at the pedestal is sufficiently large to exceed the MHD edge stability limit leading to the triggering of ELMs (at the end of the simulations shown in figure 8).

5. Gas fuelling in stationary H-mode phases

Simulations of the gas fuelled stationary H-mode for a range of currents (5–15 MA) at 5.3 T and for 7.5 MA/2.65 T (representative of low field operation) DT plasmas in ITER have been carried out. The modelled plasmas are heated by both ICRH (20 MW) and NBI (33 MW) in the simulations of the stationary DT H-mode plasmas along with the intrinsic fusion power. Figures 9 and 10 show the density and electron temperature profiles obtained for 5, 7.5, 10, 15 MA / 5.3 T and 7.5 MA / 2.65 T H-modes by varying the gas puff rate between a minimum level of $0.5\text{--}1.0 \cdot 10^{22} \text{ s}^{-1}$ and the applicable maximum without reaching complete divertor detachment. In all cases, core fuelling by neutrals was found to be low (only $\sim 0.5\text{--}1.5 \cdot 10^{21} \text{ s}^{-1}$ DT neutral particles penetrate beyond the separatrix). Due to

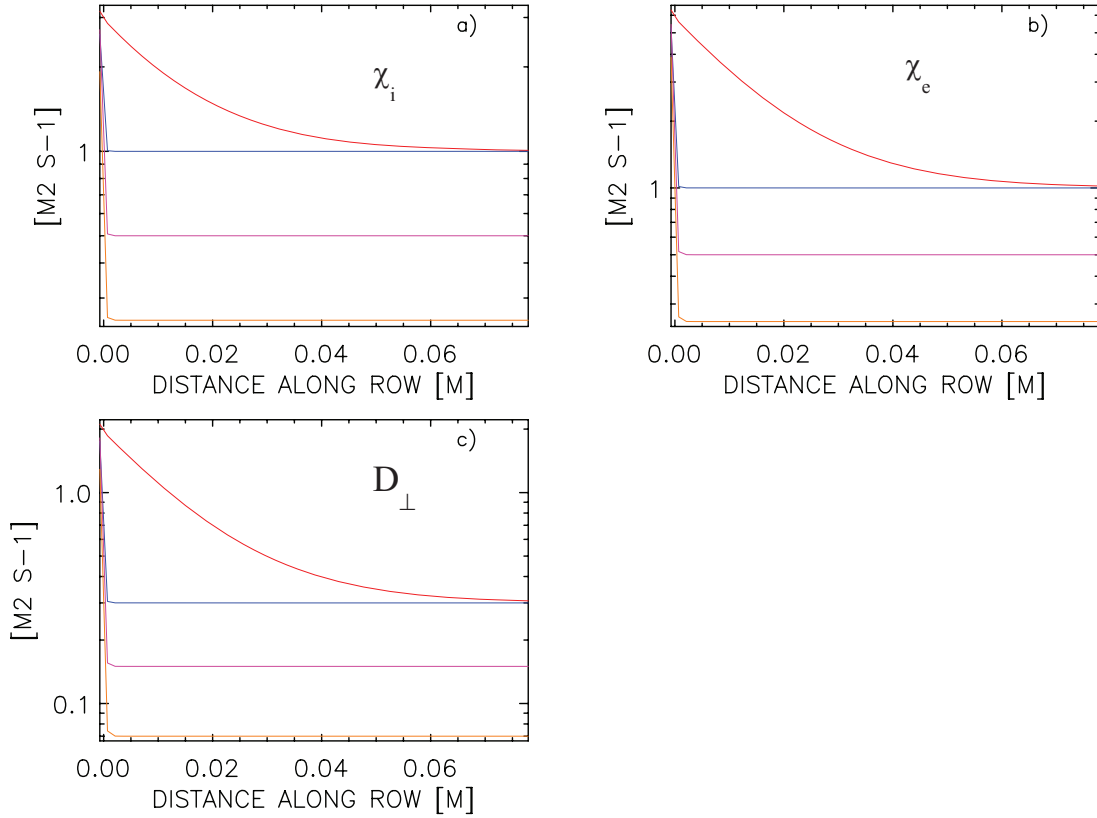


Figure 7. Radial profiles of the electron (a) ion (b) conductivities and particle diffusivity (d) used in the simulations reported in figure 6. The distance along the row is the distance from the separatrix (located at row 0.0) on a computational line perpendicular to the magnetic field (in the ‘radial’ direction), see the EDGE2D grid in figure 2.

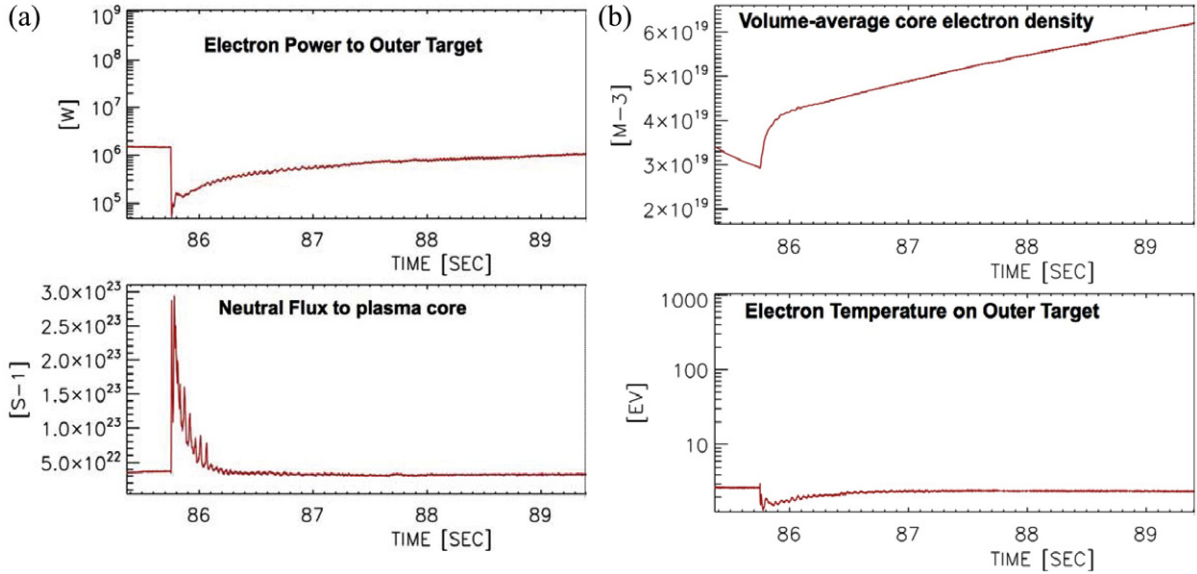


Figure 8. Time evolution of plasma parameters from the L-mode into the H=1 H-mode for a 15 MA/5.3 T plasma: (a) electron power to outer target (top), core neutral source inside the separatrix (bottom), (b) volume average core electron density (top) and separatrix ion temperature at divertor outer target (bottom).

increased collisionality at the edge, the core plasma inwards particle pinch is only present in the very central plasma region for 5 MA plasmas. The achievable separatrix density by gas fuelling together with peaking of the plasma density due to the core pinch expected in low collisionality plasmas maintain the average density above the acceptable limit for shine-through

loads on the first wall ($\sim 2.5\text{--}3.0 \cdot 10^{19} \text{ m}^{-3}$), which ensures that the full power stationary NBI power of 33 MW and ion energy of 1 MeV used in the simulations can be applied in ITER for these plasma conditions.

At low plasma currents with negligible fusion power, the density tends to saturate as the divertor detachment is

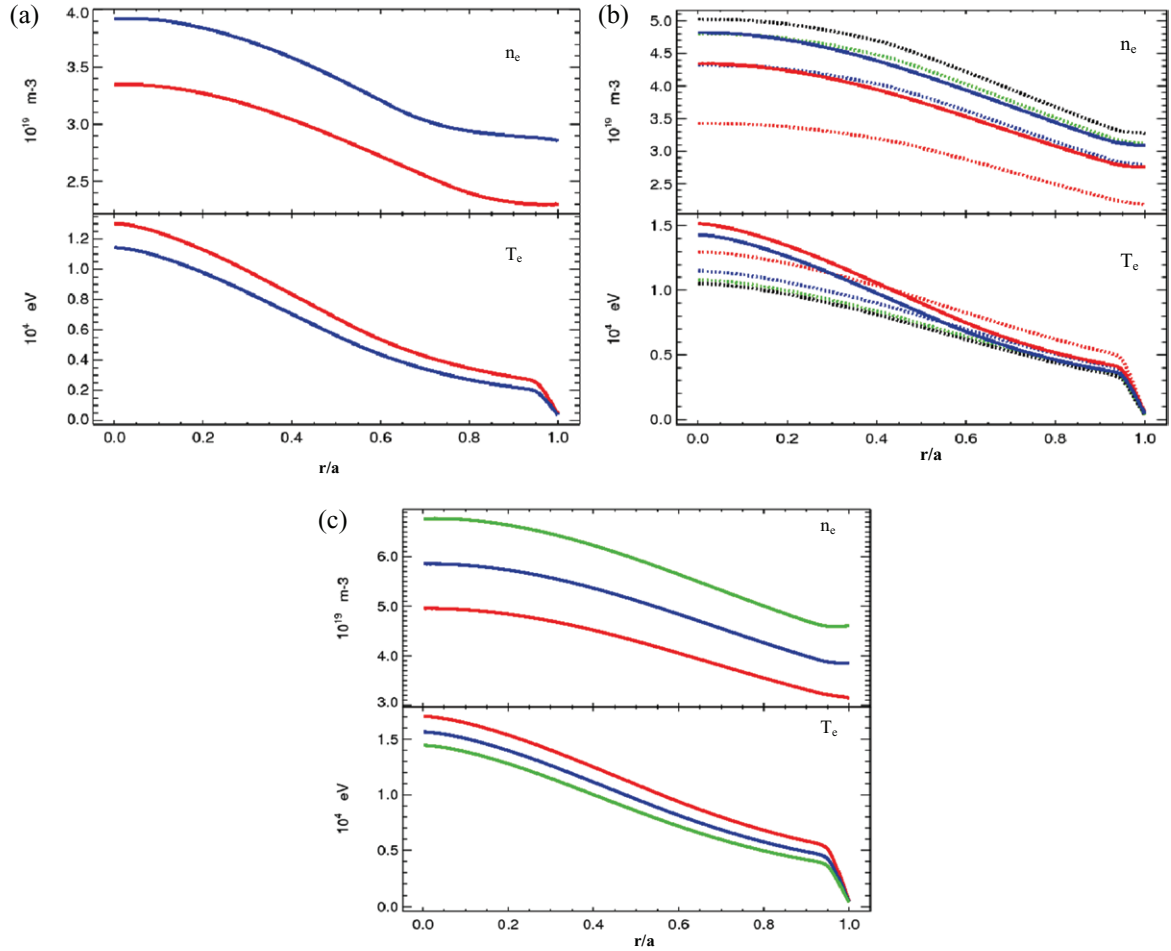


Figure 9. Electron density (top) and temperature (bottom) in the stationary H-mode, (a) $\Gamma_{DT} = 0.5 \cdot 10^{22} \text{ s}^{-1}$ (red) and $1.0 \cdot 10^{22} \text{ s}^{-1}$ (blue) at 5 MA / 5.3 T, (b) $\Gamma_{DT} = 0.5 \cdot 10^{22} \text{ s}^{-1}$ (red), $1.0 \cdot 10^{22} \text{ s}^{-1}$ (blue), $1.5 \cdot 10^{22} \text{ s}^{-1}$ (green) and $2.0 \cdot 10^{22} \text{ s}^{-1}$ (black) at 7.5 MA / 5.3 T (solid) and 7.5 MA / 2.65 T (dashed), (c) $\Gamma_{DT} = 0.75 \cdot 10^{22} \text{ s}^{-1}$ (red), $1.25 \cdot 10^{22} \text{ s}^{-1}$ (blue) and $2.0 \cdot 10^{22} \text{ s}^{-1}$ (green) at 10 MA / 5.3 T.

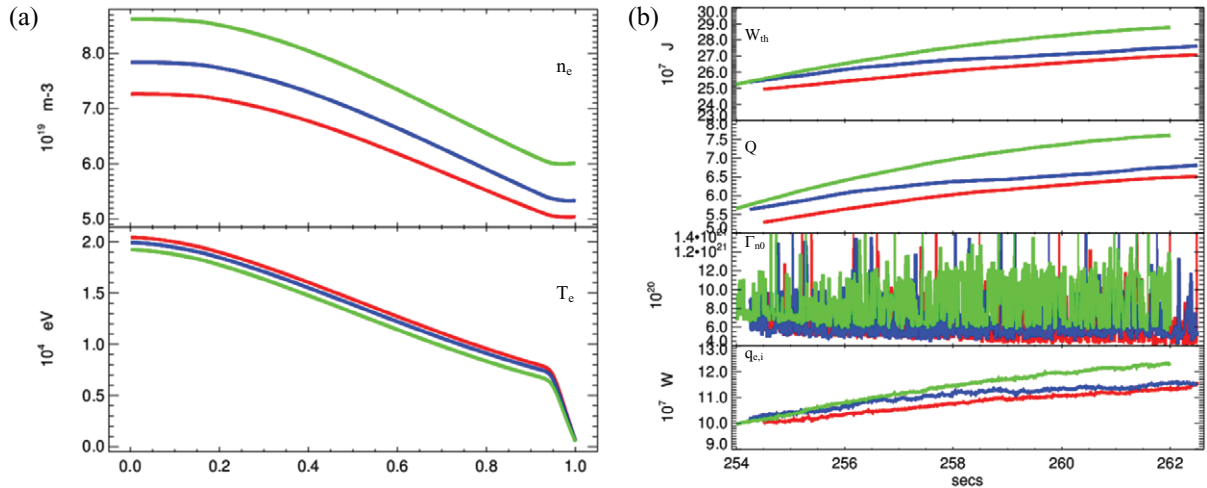


Figure 10. (a) Electron density (top) and temperature (bottom) profiles (b) from top to bottom: thermal energy, Q , D + T neutral influx at the separatrix, electron-ion heat flux at the separatrix, for a gas puff rate of $\Gamma_{DT} = 1.0 \cdot 10^{22} \text{ s}^{-1}$ (red), $2.0 \cdot 10^{22} \text{ s}^{-1}$ (blue) and $3.0 \cdot 10^{22} \text{ s}^{-1}$ (green) at 15 MA / 5.3 T.

approached in a similar fashion to the L-mode simulations. This occurs at larger values of the density than in L-mode simulations because of the increased power flux and lower values of the separatrix transport coefficients in H-mode (see

gas flux scan at $I_p = 7.5 \text{ MA}$ / $B_0 = 2.65 \text{ T}$ in figure 9(b)). Comparing cases at 7.5 MA, full and half field, the temperature gradient in the core is increased at $B_0 = 5.3 \text{ T}$ due to reduced sawtooth activity, and the density is higher due to a

higher separatrix density for fuelling levels up to $1.0 \cdot 10^{22} \text{ s}^{-1}$. However, because the SOL heat flux width is larger for 5.3 T than for 2.65 T (as a consequence of the longer connection length and of the dependence on the safety factor of the transport in the barrier / SOL region), this leads to lower SOL power densities and earlier onset of detachment at lower fuelling rates for 5.3 T than for 2.65 T. As a consequence, higher plasma densities are reached at 2.65 T at the highest fuelling rates. For plasma current levels of 10 MA and above the picture of the density behaviour above changes because the fusion power starts to play a role. Already at 10 MA the heat flux through the separatrix is increased by $\sim 20\text{--}25 \text{ MW}$ due to alpha heating. This allows the achievement of a higher density at the separatrix by gas fuelling before the onset of detachment, which in turn allows for higher plasma densities in the core to be achievable by gas fuelling. For 15 MA, the plasma density does not saturate at higher gas puff rates, as the fusion power is noticeably increased, amplifying the heat flux crossing the separatrix and thus maintaining the plasma away from detachment. This, together with the presence of an anomalous inward pinch in the core due to the low plasma collisionality plasmas (predicted by GLF23 and correctly reproduced by the retuned BgB model applied here) leads to peaked core density profiles allowing a volume average electron density of $\sim 6\text{--}7 \cdot 10^{19} \text{ m}^{-3}$ to be reached by gas fuelling only in ITER DT plasmas at 15 MA/5.3 T (the density at the separatrix is predicted to range between $5\text{--}6 \cdot 10^{19} \text{ m}^{-3}$ in such conditions). It should be noted that in these simulations the pressure on top of the pedestal is $\sim 120\text{--}125 \text{ kPa}$ which is in good agreement with estimates from edge MHD stability modelling for ITER [12, 13]. Interestingly, for this gas fuelled plasmas the pressure increase in the pedestal zone is due almost exclusively to the large temperature gradients in the ETB, as the density profiles in the ETB are almost flat. This is the consequence of negligible neutral penetration across the separatrix into the core plasma and the lack of an anomalous pinch in the barrier region (assumed in this modelling). The short ionisation path of neutrals in ITER SOL leads to neutral-fuel getting ionised near the separatrix. The fuel ions (and electrons) build up at the separatrix and diffuse in the pedestal region where they reside due to the low level of transport and large particle confinement time of the ITER barrier region. The above density dynamics is different to the one observed in today's tokamaks where a large fraction of neutrals penetrate inside the separatrix before getting ionised. Note that no anomalous inward particle pinch has been used in these simulations. Modelling results for ITER show that for a gas fuelling level of $3.0 \cdot 10^{22} \text{ s}^{-1}$, $Q \sim 7.5$ can be achieved, even if the core fuelling by neutrals remains at the low level of $\sim 10^{21} \text{ D} + \text{T}$ particles (see figure 10(b)). This high fusion performance without core fuelling by pellets is obtained in the simulations because it is assumed that the maximum achievable separatrix density by gas fuelling is only determined by the detachment limit in ITER H-modes and that increasing gas fuelling has no influence on the maximum pressure that can be achieved in the H-mode pedestal. Present experiments in H-mode plasmas show that increasing plasma density by gas fuelling tends to decrease the pedestal pressure and overall energy confinement

in H-mode, which can be due to the increased plasma collisionality affecting the edge current magnitude and thus edge MHD stability, to the increased neutral penetration increasing transport in the pedestal barrier, etc. It is presently unknown if such behaviour will be found in ITER or not, as some of the physics mechanisms invoked to explain present experimental results (i.e. decreased edge current density due to high collisionality or large core plasma neutral source) are not expected to play a significant role in ITER. In addition, it is possible that the small outflux from the core plasma for these gas fuelled H-modes may not produce the required helium exhaust to maintain the helium density at a level of $\sim 5\%$ compatible with stationary high Q_{DT} plasmas in ITER. For an alpha heating power of 100 MW the stationary helium core plasma production and stationary helium exhaust required is $1.8 \cdot 10^{20} \text{ s}^{-1}$, which corresponds to a minimum DT outflux of $3.6 \cdot 10^{21} \text{ s}^{-1}$ for 5% helium concentration assuming similar particle transport for DT and He. This value is already larger than the maximum core neutral source obtained in these simulations and would further increase due to the expected helium de-enrichment at the divertor which increases the pumped DT throughput and associated core plasma outflux to achieve a given net helium ash removal rate from the core plasma. Further modelling studies including helium transport and exhaust are under progress to assess the fusion performance achievable in these gas fuelled H-modes and the duration over which it can be eventually sustained in ITER. It is also important to note that these optimistic results for the core density and Q achievable in ITER 15 MA plasmas with gas puffing also rely on the presence of an anomalous pinch in the core, and, even more importantly, on assumptions of the reduction of anomalous transport in the SOL in H-mode plasmas. In these simulations it is assumed that the reduction of anomalous transport observed in the edge transport barrier slightly extends into the SOL. This is observed in present-day experiments and is the physics mechanism invoked to explain the observed scaling of the SOL power width in H-modes in present experiments [14]. Without extension of at least a few millimetres of the low transport domain of the ETB beyond the separatrix, the heat flux would spread over a wider area on the target plates, due to increased perpendicular transport, and detachment would be reached at much lower gas flux rates. The precise choice for the reduction of transport in the SOL of H-mode plasmas and the level of impurities at the divertor, which contribute to radiated power losses and to the decrease of divertor temperature affecting detachment onset, lead to a much higher level for the separatrix densities to be achievable in our simulations compared to original ITER studies [15]. These were performed with larger anomalous transport coefficients (uniform in the SOL) [15] and carbon impurities. More recent studies carried out with transport coefficients similar to those in the integrated modelling presented here show similar values for the saturation of the separatrix density in ITER to those reported here [16] (see section 6 for a more detailed discussion of this issue). The achievable separatrix density before the onset of detachment is also influenced by the presence of W (not considered in these studies) and the puffing of Ne because of the associated increase of plasma

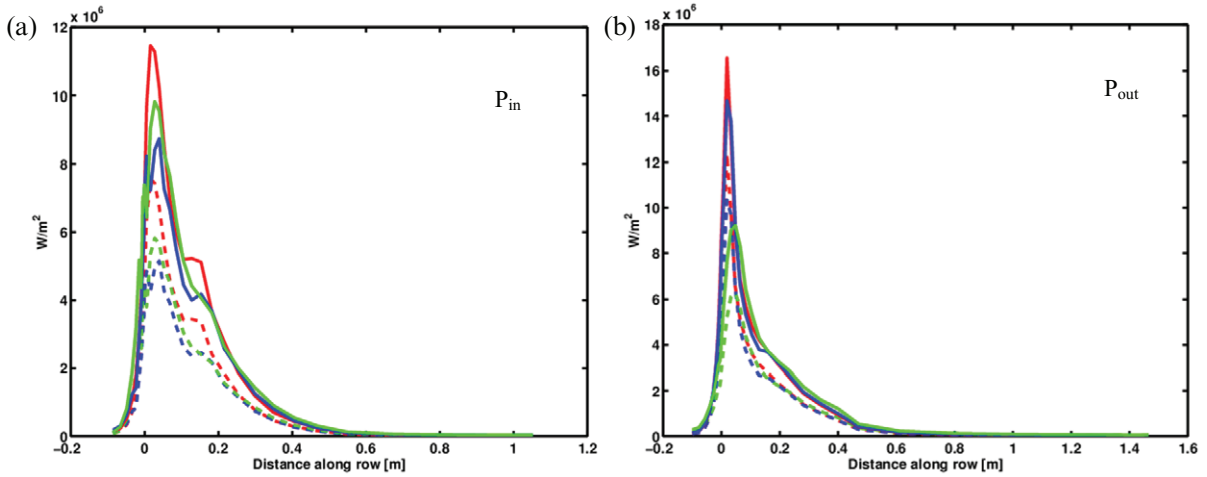


Figure 11. Upper (solid) and lower (dashed) estimate for the power density on the inner (a) and outer (b) targets, for a gas puff rate of $\Gamma_{DT} = 1.0 \cdot 10^{22} s^{-1}$ (red), $2.0 \cdot 10^{22} s^{-1}$ (blue) and $3.0 \cdot 10^{22} s^{-1}$ (green).

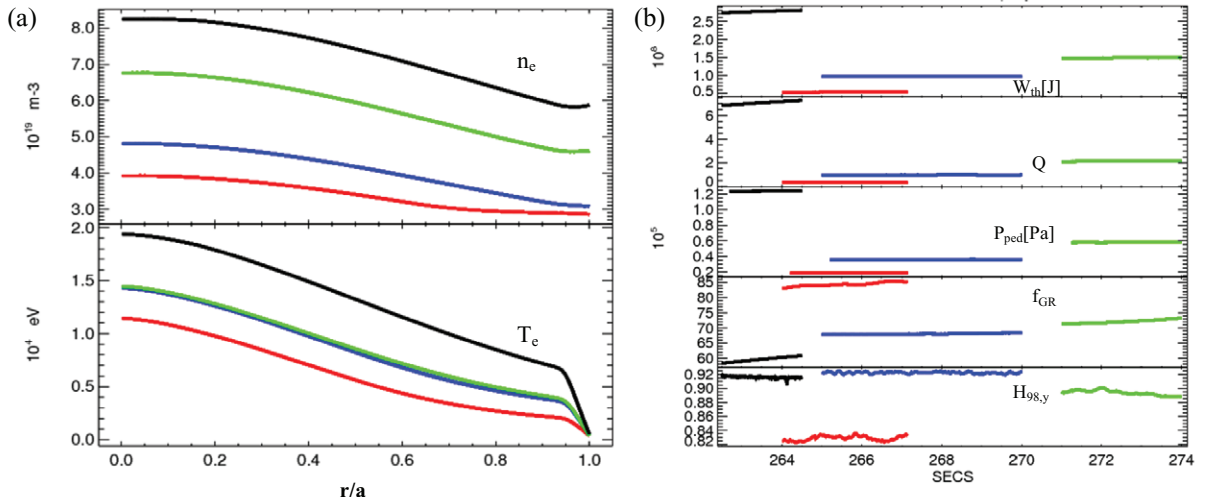


Figure 12. (a) Electron density (top) and temperature (bottom) profiles (b), from top to bottom: time evolution of thermal energy, Q , pressure on top of the pedestal, fraction of Greenwald density limit and $H_{98,y}$, for the maximum DT gas puff rates that could be applied at 5 MA (red), 7.5 MA (blue), 10 MA (green) and 15 MA (black) without reaching detachment.

radiative losses in the core and SOL divertor. It should be noted that large radiative losses are expected to be required in ITER for high Q_{DT} plasmas to maintain the peak divertor heat load below $10 MW m^{-2}$. According to estimates for the power density reaching the inner and outer target (see figure 11) to reduce the peak divertor heat load under this limit requires gas fuelling levels equal or higher than $3 \cdot 10^{22} s^{-1}$ for the Ne seeding levels explored in this study which lead to core Ne concentrations of typically only 0.05%. These gas fuelling levels correspond to a separatrix density equal or higher than $6.0 \cdot 10^{19} m^{-3}$, as shown in figure 10(a). A summary of the plasma parameters in ITER 5.3 T H-modes for currents of 5, 7.5, 10 and 15 MA at the highest density achieved by gas fuelling only with the integrated modelling tools and assumptions described above is shown in figure 12.

As in the case of L-mode simulations, a detailed quantitative validation of the integrated modelling approach applied to ITER with experimental results from gas fuelled H-mode density scans in present experiments is required before final

conclusions for ITER can be drawn. In the case of H-mode conditions this requires not only the validation of the SOL modelling assumptions setting the maximum achievable separatrix density but also of the MHD stability and transport assumptions in the ETB for H-mode plasmas with high plasma density, low pedestal collisionality and low neutral source in the pedestal region. Such validation requires dedicated experiments at the highest currents and powers achievable in the existing tokamaks to approach these ITER-like conditions.

6. Impact of ETB/SOL transport on gas fuelling of ITER stationary H-modes

Scans in ETB and SOL transport assumptions have been performed for the 15 MA / 5.3 T gas fuelled H-mode scenario at a low core impurity concentration ($Z_{eff} \sim 1.02-1.10$) (i.e. low Ne seeding) to assess the sensitivity of the predicted achievable core density, fusion performance and resulting divertor target power loads on uncertainties in edge transport conditions.

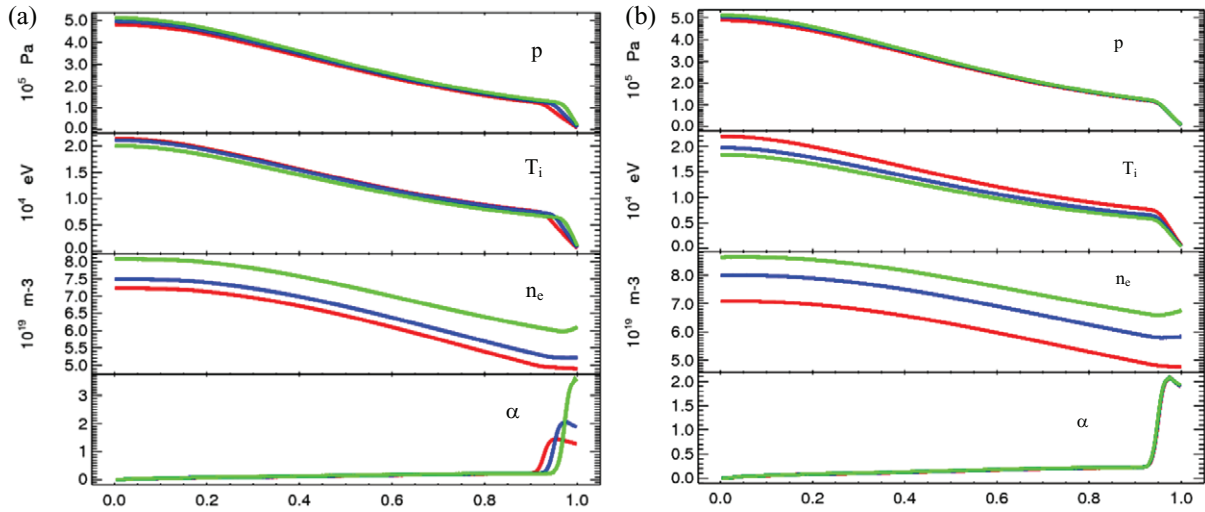


Figure 13. From top to bottom: profiles of thermal pressure, ion temperature, electron density, normalised pressure gradient. (a) Scan 1, Case 1 (red), 2 (blue) and 3 (green). (b) Scan 2, Case 4 (red), 5 (blue) and 6 (green) (see table 1).

In one scan, the width of the extension of the reduced transport zone (or ETB extension) into the SOL was kept fixed at $\Delta_{\text{ETB} \rightarrow \text{SOL}} \sim 5\text{--}6\text{ mm}$ (at the outer midplane) and the ETB width itself was varied in a range between $\Delta_{\text{ETB}, \rho_{\text{tor, norm}}} = 0.05\text{--}0.09$, corresponding to $5\text{--}10\text{ cm}$ at the outer midplane. This was done while adjusting the critical pressure gradient in the ETB so that the pressure on top of the pedestal was maintained at a level $p_{\text{ped}} \sim 120\text{--}130\text{ kPa}$ for all conditions. To ensure a fast convergence to stationary conditions a low gas flux rate of $\Gamma_{\text{DT}} = 10^{22}\text{ s}^{-1}$ was prescribed for this scan. Another scan was performed with fixed $\Delta_{\text{ETB}, \rho_{\text{tor, norm}}} = 0.07$, corresponding to $\sim 7\text{--}8\text{ cm}$ at the outer midplane, and without any extension of the reduced transport into the SOL (similar to transport assumptions in previous ITER studies [15, 17]). For this second scan the gas flux rate has been varied within $\Gamma_{\text{DT}} = 1\text{--}3 \cdot 10^{22}\text{ s}^{-1}$.

The results of these two scans are shown in figure 13 and a summary of selected plasma parameters for these simulations are summarized in table 1. The results for scan 1 demonstrate that uncertainties for the level of transport in the ETB and in its extension into the SOL or uncertainties for the width of the ETB itself lead to a comparably low variations in the predicted core density of $\Delta n_e \sim 1.0\text{--}1.5 \cdot 10^{19}\text{ m}^{-3}$ for low gas fuelling rates. A reduction in the ETB width at constant pedestal pressure and at constant heat flux from the core yields a reduction in heat conductivities in that zone, associated with the appearance of larger gradients in temperature. These steeper gradients are also present in the SOL region where the ETB transport is applied leading to a reduction of the power decay length λ_q and to a higher separatrix density for the same gas fuelling rate. This effect is expected to be more pronounced at increased gas flux rates where larger variation in separatrix densities should be obtained due to the varying detachment limit with λ_q consistent with previous edge modelling studies for ITER [16]. It should be noted that in these simulations the impurity content is extremely low (trace amounts of Ne, core Be concentration of $<0.5\%$ and W not considered), which leads to an

Table 1. Core and SOL plasma parameters for the simulation cases of two scans in ETB/SOL transport conditions.

	Scan 1			Scan 2		
	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6
$\Gamma_{\text{D, T}} [10^{22}\text{ s}^{-1}]$	1.0	1.0	1.0	1.0	2.0	3.0
$\Delta_{\text{ETB}, \rho_{\text{tor, norm}}}$	0.09	0.07	0.05	0.07	0.07	0.07
$\Delta_{\text{ETB} \rightarrow \text{SOL}} [\text{mm}]$	5–6	5–6	5–6	0	0	0
$\alpha_{\text{crit, ETB}}$	1.3	1.9	3.4	1.9	1.9	1.9
$n_{\text{D+T, sep}} [10^{19}\text{ m}^{-3}]$	4.9	5.2	6.0	4.7	5.8	6.7
$\Gamma_{\text{D, T, neut, sep}} [10^{21}\text{ s}^{-1}]$	2.0	1.8	1.4	1.4	2.4	0.5
$\chi_{\text{e, sep}} [\text{m}^2\text{ s}^{-1}]$	0.2	0.14	0.08	0.14	0.14	0.14
$T_{\text{e, sep}} [\text{eV}]$	490	600	800	450	390	350
$P_{\text{SOL}} [\text{MW}]$	115	120	125	120	120	120
$\lambda_{q, \text{out, midpl.}} [\text{mm}]$	~ 2.0	~ 1.5	~ 1.0	~ 6.0	~ 6.0	~ 5.5
$\max(q_{\text{OT}}) [\text{MW m}^{-2}]$	19	23	19	14	12	9
$W_{\text{th}} [\text{MJ}]$	265	280	290	270	275	280
$P_{\text{fus}} [\text{MW}]$	375	400	415	385	400	—
$< n_e > [10^{19}\text{ m}^{-3}]$	5.9	6.3	—	5.8	6.8	—

Note: Values for the maximum local power density on the outer target plate $\max(q_{\text{OT}})$ do not include power deposited by radiation and neutral interaction and are thus a lower-bound estimate.

improved plasma performance compared to the simulations in section 5 due to reduced dilution and radiation in the core ($Q_{\text{fus}} \sim 7\text{--}8$ is predicted already for $\Gamma_{\text{DT}} = 10^{22}\text{ s}^{-1}$). The low level of impurities together with the low gas fuelling rates leads to peak divertor power loads of $\sim 20\text{ MW m}^{-2}$, which are a factor of 2 larger than the stationary power handling limit of the ITER W divertor. In this scan, the transport in the far SOL (i.e. beyond the region of reduced anomalous transport due to the H-mode) has not been varied. However, the dependence of the predicted core density on $\chi_{\text{far SOL}}$ and $D_{\text{far SOL}}$ is expected to be smaller compared to the L-mode results, because the core plasma properties are mainly determined by transport assumptions in the ETB and the vicinity of the separatrix, which in these H-mode simulations are determined by the choice of transport in the ETB.

Scan 2 provides an estimate for the uncertainty in plasma core density and performance predictions due to the unknown size of the ETB width extension into the SOL, and it provides an opportunity to do a closer comparison with predictions for similar ITER plasma configurations that have previously been studied by combined simulations of B2-EIRENE and ASTRA [15, 17] with the same SOL transport assumptions (constant $\chi_{i,e} = 1.0 \text{ m}^2 \text{ s}^{-1}$ and $D_i = 0.3 \text{ m}^2 \text{ s}^{-1}$ throughout the SOL plasma without any reduction near the separatrix due to extension of the ETB extension in the SOL). Comparing simulation cases at low gas flux rate $\Gamma_{DT} = 10^{22} \text{ s}^{-1}$ with and without the $\sim 5\text{--}6 \text{ mm}$ extension of the ETB into the SOL (Case 2 versus Case 4), the core density and fusion power are slightly reduced ($\Delta n_e \sim -0.5 \cdot 10^{19} \text{ m}^{-3}$, $\Delta P_{fus} \sim -20 \text{ MW}$) for the case without the ETB extension. These changes are caused by an increase in λ_q due to the increase of the transport coefficients from the values typical in the ETB to the larger and constant value in the SOL for Case 4 compared to Case 2. This variation in the achieved separatrix density for a given gas flux related to the specifications of the near-separatrix SOL transport is expected to be larger at larger fuelling rates as detachment is strongly influenced by λ_q , in agreement with previous edge plasma simulation studies for ITER [16]. Comparing Cases 4–6 with similar conditions in [15] ($P_{AUX} = 53 \text{ MW}$, $\langle n_{D+T} \rangle \sim 6\text{--}8 \cdot 10^{19} \text{ m}^{-3}$), the predictions for core fusion performance and for the power crossing the separatrix are found to be similar between the two modelling approaches (e.g. $Q_{fus} \sim 7$ versus $Q_{fus} \sim 5\text{--}6$ and $P_{SOL} \sim 120 \text{ MW}$ versus $P_{SOL} \sim 110 \text{ MW}$ for Case 4 with respect to similar conditions in [15]). Indeed more recent simulations with ASTRA including the self-consistent evaluation of the pedestal pressure from the EPED1 model and the SOL relations derived from B2-Eirene in [15] find similar values of Q_{fus} for the densities modelled with the JINTRAC suite [18]. In addition, the SOL parameter relation for the maximum power density at the target plates (including power deposited by radiation and neutral interaction) given in equations (1) and (4) in [15] seems to apply quite well also to the Cases 4–6 considered here. The neutral influx at the separatrix has the same order of magnitude as that predicted by equation (5) in [15] ($\Gamma_{D,T,neut,sep} \sim 10^{21} \text{ s}^{-1}$). The relative dependency of μ (related to the degree of detachment of the divertor) with respect to the main ion separatrix density can be expressed for Cases 4–6 as $n_{D+T,sep} \propto \mu^C$, $C \sim 0.4\text{--}0.5$ and thus seems to fit well with the scaling $n_{D+T,sep} \propto \mu^{0.43}$ given in [17].

However, in absolute terms, the predicted main ion density at the separatrix is significantly higher in Cases 4–6 than that given by equation (8) in [15]. According to equation (8), $n_{D+T,sep} \sim 4 \cdot 10^{19} \text{ m}^{-3}$ for $P_{SOL} \sim 120 \text{ MW}$, whereas $n_{D+T,sep} \sim 4.5\text{--}6.5 \cdot 10^{19} \text{ m}^{-3}$ for Cases 4–6, although the normalised neutral pressure is still rather small at the applied gas flux rates $\Gamma_{DT} \leq 3 \cdot 10^{22} \text{ s}^{-1}$, $\mu < \sim 0.70$, which means that the saturation level for the separatrix density could be even higher. It should be noted that from the point of view of divertor power load control, only Case 6 is marginally compatible with the requirement of maintaining the peak divertor heat flux under 10 MW m^{-2} . Because of the discrepancy in $n_{D+T,sep}$ in absolute terms, the same fusion performance can be obtained in the

JINTRAC calculations at much lower particle source rates and μ with gas fuelling only, whereas in [15] some pellet fuelling had to be considered to achieve high core densities at limited $n_{D+T,sep}$. These differences in separatrix densities are related to the impurity species and levels considered in the two simulations (JINTRAC: Be, Ne, B2-EIRENE/ASTRA: He, Be, C) with, probably, a smaller influence of the assumptions for the material of plasma facing components (JINTRAC: Be+W, B2-EIRENE/ASTRA: C). In fact the impurity concentration in Cases 4–6 ($Z_{eff} < 1.1$) is significantly lower compared to the simulations in [15] in which significant C sputtering is present and $Z_{eff} > \sim 1.7$ at the separatrix for $P_{SOL} \sim 120 \text{ MW}$ according to equations (6), (8) and (11) in [15]. The increased impurity content leads to an increase of the radiation in the SOL by $\sim 30\text{--}40 \text{ MW}$ in the simulations in [15] compared to Cases 4–6, which leads to lower separatrix densities in the B2-Eirene/ASTRA studies. Indeed, as already mentioned in [17] and in previous ITER edge modelling studies, simulations with low impurity content of ITER high Q_{fus} scenarios would require very high separatrix densities to achieve divertor power load control. These were considered not to be compatible with high pedestal pressure/high energy confinement required for high Q_{fus} in ITER on the assumption that the plasma behaviour observed in present experiments can be extrapolated to ITER, which as mentioned above remains to be proven. Indeed, simulations with JINTRAC with Be divertor targets and low gas flux of $\Gamma_{DT} = 0.75 \cdot 10^{22} \text{ s}^{-1}$ (not shown here) have a higher $Z_{eff} \sim 1.7\text{--}1.8$, increased impurity radiation in the divertor and SOL and degraded fusion performance with $P_{SOL} \sim 95 \text{ MW}$. For such simulations, the main ion density at the separatrix dropped down to lower values $n_{D+T,sep} \sim 2.5 \cdot 10^{19} \text{ m}^{-3}$ similar to those obtained for comparable conditions in [15]. Taking into account the discrepancies in impurity modelling assumptions, good overall agreement has thus been found between independently performed JINTRAC and B2-EIRENE/ASTRA predictions for the 15 MA ITER baseline scenario core and edge plasmas, which strongly supports the validity of both of these integrated modelling approaches for stationary conditions.

7. Summary of main transport modelling assumptions

In this section we summarise the transport assumptions used in the simulations presented here:

Stationary magnetic field in the SOL during the all evolution of the discharge. Fixed separatrix boundary. Transport in the SOL simulated with EDGE2 benchmarked with SOLPS for ITER intermediate density cases. Perpendicular transport in the SOL in the coupled core / edge simulations modelled with an effective diffusivity / conductivity defined as the analytic continuation of the diffusivity / conductivity inside the separatrix extrapolated to a constant far SOL value. The dynamics of neutrals are simulated with EIRENE. The magnetic field inside the separatrix evolved accordingly to the evolution of the plasma pressure and current density. Transport in the core of the L-mode phase is Bohm / gyroBohm from [5]. Transport in the H-mode up to the top of the pedestal is

modelled with Bohm /gyroBohm calibrated against GLF23. Transport in the barrier region is neoclassical during the L-H transition and later setup to the required value to maintain the pressure gradient such that α is below α_{crit} . A description of the physics modules in JINTRAC and references to each code in the package is given in [1].

8. Conclusion

Simulations of gas fuelled plasmas have been carried out with the JINTRAC suite of codes for ITER plasmas at 15 MA, 10 MA, 7.5 MA and 5 MA / 5.3 T and 7.5 MA / 2.65 T. The simulations at 15 MA indicate that, for a range of transport assumptions compatible with observations in present tokamaks, the maximum achievable density with deuterium-tritium gas fuelling only is in the range of 2.2 and $3.5 \times 10^{19} \text{ m}^{-3}$ in the L-mode phase, limited by the onset of the divertor detachment. These density values are similar to those required for stationary application of the NBI at full power and ion injection energy in ITER with acceptable shine-through loads on the first wall, particularly taking into account the modelled density evolution after the H-mode transition. For lower plasma currents the achievable plasma densities by gas fuelling in the modelled L-mode plasmas are lower than those for 15 MA. While the accuracy of these predictions for ITER requires detailed validation of the models applied with present experimental results from L-mode gas fuelled density scans, the studies carried out so far indicate that pellet injection may have to be applied more routinely to fuelling of L-mode plasmas in ITER, particularly for lower plasma current levels, than originally foreseen. Studies are in progress to model pellet fuelling of these ITER L-mode plasmas with the same tools as in the present study. Modelling of the H-mode transition, following the increase of additional heating beyond the L-H threshold power, shows that the pedestal density can be increased to a level of $5\text{--}6 \times 10^{19} \text{ m}^{-3}$ by gas fuelling only in ITER for a plasma current of 15 MA. This density level can be maintained during the whole stationary H-mode phase with a puffing rate of $2\text{--}3 \times 10^{22} \text{ s}^{-1}$ or even further increased once a significant amount of fusion power is generated. It is worth noticing that the simulations presented in this paper do not include W as an impurity, hence neglect W radiation / contamination in the core and SOL, which might play a role in ITER steady state and transients (ELMs and L-H / H-L transition). However as ITER is a large size machine, operating at higher densities/magnetic field as compared to present Tokamaks, W penetration in the core is expected to be low (in steady state) [19] and the deleterious effects of W on discharge performances observed in e.g. JET are expected to ‘disappear’ or be strongly mitigated.

The Q obtained with gas-puff only for gas fuelled 15 MA/5.3 T plasmas in ITER is found to reach about $Q \sim 7\text{--}8$. It should be noted that the sustainment of such high Q conditions over long timescales with gas fuelling is questionable due to He ash accumulation in the core plasma and further modelling studies are in progress to assess this issue quantitatively. The present studies are thus in agreement with previous

ITER evaluations which indicated the need of pellet fuelling for the achievement of the high density required for the $Q = 10$ baseline scenario in ITER and to provide the required helium exhaust to maintain it over long timescales. The modelled fusion performance of the 15 MA gas fuelled ITER H-mode plasmas in our studies depends critically on two assumptions:

- (a) that the achievable separatrix density in high energy confinement H-modes is solely limited by divertor detachment, which allows the increase of the separatrix density by gas puffing up to the level in which strong divertor detachment sets in and divertor peak power fluxes decrease under 10 MW m^{-2} with minimum contribution of impurity radiation in the SOL/divertor
- (b) that the existence of flat density profiles in the ETB (associated with the low neutral source for these gas fuelled plasmas) will not alter the reduction of the transport in this region associated with H-mode confinement nor the edge plasma MHD stability, thus allowing pressures up to the edge MHD limit expected for the low collisionality ITER pedestal plasmas to be reached.

Both of these modelling assumptions require confirmation from dedicated experiments and comparison with the integrated modelling tools applied to model ITER in order to strengthen the basis for the results presented in this paper.

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