A FULL TRANSPORT TREATMENT OF ITER BURN CONTROL SCENARIOS

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ABSTRACT

Reference operating scenarios for the physics phase of the International Thermonuclear Experimental Reactor (ITER) rely on low temperature, high density operation for good divertor performance. Operating points in this region are usually thermally unstable and active burn control is required to maintain operation at the selected operating point. We present a series of transport simulations of ITER burn control scenarios with modulated neutral beam heating, using the 1-1/2D code WHIST. Our results indicate that control is possible, except at very high densities ($\langle n_e \rangle \geq 1.80 \times 10^{20} \text{ m}^{-3}$) where off-axis heating due to the poor beam penetration makes control difficult, especially for negative perturbations.

1. INTRODUCTION

Most of the work on ITER burn control to date, has been based on 0-D models [1-3]. Calculations based on 0-D models are invaluable, since they allow a large number of scoping and parametric studies to be performed fast and at a minimal computational cost. They do however have a major weakness, in that profile effects cannot be properly taken into account. Although most of the recent 0-D models used in the ITER Burn Control studies [1-3] incorporate profile effects by profile-averaging of the transport equations, this is done assuming profiles of a certain, rather arbitrary but computationally convenient, functional form. Moreover, these profiles are assumed fixed i.e. they do not change during time-dependent simulations.

Profile effects however are believed to be important, both for operating point selection and control, and for evaluating the feasibility of operating at the thermally unstable regime of the operating space. In particular, the spatial dependence of the auxiliary heating power and fueling deposition can be especially important for burn control simulations, since it can affect the effectiveness of several active burn control methods under consideration, such as auxiliary power feedback and modulation of the fueling rate. Moreover, changes in the pressure profile during the thermal runaway are important since the fate of the discharge (disruption or equilibrium at a thermally stable point) can depend on the degree of peaking of the pressure profile.

To take profile effects into account in a self-consistent way, space-dependent calculations are needed in which the profiles of the plasma parameters are determined self-consistently by solving the particle and energy transport equations in 2D flux surface geometry.

In this work, a full transport treatment of ITER burn control scenarios is presented using the 1-1/2D code WHIST [4]. The WHIST code was chosen for these simulations since it is reasonably fast, yet it contains most of the physics that is relevant to our

studies. Moreover, its ability to produce POPCON plots [4] makes it specially suitable for operating point selection and burn control calculations.

The main objectives of this work are to compare the 1-1/2D results with those obtained from the 0-D models and to explain the origin of any differences, to identify candidate operating points in the low- $\langle T \rangle$, high- $\langle n_e \rangle$ regime having the desired properties, to determine whether operation at these thermally-unstable operating points is feasible when spatial effects are taken into account, and finally to evaluate any technological requirements imposed by the control methods on the associated ITER hardware systems, as well as any requirements on the diagnostic capabilities for effective control.

2. THE WHIST TRANSPORT CODE

2.1. Computational model

The WHIST code is a 1-1/2D transport code that solves the flux-surface-averaged particle and energy transport equations:

$$\frac{\partial}{\partial t} (V' n_j) = - \frac{\partial}{\partial \rho} (V' \tilde{\Gamma}_j) + V' S_j \quad (1)$$

$$\begin{aligned} \frac{3}{2} \frac{\partial}{\partial t} \left[(V')^{5/3} n_e T_e \right] = & - (V')^{2/3} \frac{\partial}{\partial \rho} \left[V' \tilde{Q}_e + \frac{5}{2} V' T_e \tilde{\Gamma}_e \right] + (V')^{5/3} \left[Q_e^{OH} + Q_e^{RAD} + Q_e^N \right. \\ & \left. - \sum_j Q_{ej} + Q_e^{NB} + Q_e^{RF} + Q_e^F \right] \end{aligned} \quad (2)$$

$$\frac{3}{2} \frac{\partial}{\partial t} \sum_j^{ions} \left[(V')^{5/3} n_j T_i \right] = - \sum_j^{ions} \left\{ (V')^{2/3} \frac{\partial}{\partial \rho} \left[V' \tilde{Q}_j + \frac{5}{2} V' T_i \tilde{\Gamma}_j \right] + (V')^{5/3} \left[Q_j^F + Q_{ej} + \right. \right.$$

$$\left. Q_j^{NB} + Q_j^{RF} + Q_j^F + Q_j^N \right] \} \quad (3)$$

where $j = H, D, T, He$. There is one particle diffusion equation for each thermal ion species j , while the electron density is computed from quasineutrality. The particle net source rate S_j contains contributions from fusion, neutral beams, pellets, and ionization of neutrals. In the energy transport equations, all thermal ion species are assumed to have a common temperature T_i . The rest of the terms have their usual meaning:

$$\begin{aligned} Q_e^{OH} &= \text{ohmic heating term} \\ Q^{RAD} &= \text{bremsstrahlung, line, recombination, and synchrotron radiation losses} \\ Q_e^N, Q_j^N &= \text{energy losses due to interactions with neutrals} \\ Q_{ej} &= \text{electron-ion rethermalization term} \\ Q_e^{NB}, Q_j^{NB} &= \text{energy transfer rate from fast beam ions to electrons and ions} \\ Q_e^{RF}, Q_j^{RF} &= \text{energy transfer rate from RF heating source to electrons and ions} \\ Q_e^F, Q_j^F &= \text{energy transfer rate from fusion alphas to electrons and ions} \end{aligned}$$

In the above equations, V' is the differential volume between flux surfaces and ρ is a flux surface label which is taken here to be the half-diameter of the flux surface at the midplane. The poloidal magnetic field is calculated from the current diffusion equation.

The flux-surface-averaged particle and energy fluxes are of the form

$$\bar{\Gamma}_j = -D_j \langle |\nabla \rho|^2 \rangle \frac{\partial n_j}{\partial \rho} + n_j v_j^{\text{pinch}} \langle |\nabla \rho| \rangle, \quad j = \text{ions} \quad (4)$$

$$\bar{Q}_j = -\chi_j \langle |\nabla \rho|^2 \rangle \frac{\partial T_j}{\partial \rho}, \quad j = i, e \quad (5)$$

where an anomalous pinch term has been included in the expression for the particle flux.

MHD equilibrium is calculated from a moments solution to the Grad-Shafranov equation. Transport of neutrals from the plasma edge is calculated using the SPUDNUT model. While the code allows for finite thermalization and diffusion of fast ions (alphas and beam ions), instantaneous thermalization has been assumed in these calculations. This is justified, since the thermal runaway time is much larger than the fast ion slowing down time for typical Physics Phase ITER parameters. Neutral beam deposition is calculated with a semiempirical pencil-beam model, generalized to multiple beamlets and arbitrary injection geometry. At this stage, impurities are not transported but are assumed to be a constant fraction of the electron density.

2.2. Modifications to the code for the ITER studies

The WHIST code was modified to be consistent with the ITER physics guidelines. The most important modifications are:

- The stopping cross sections for neutral beam penetration are calculated using the recently published fits by Janev *et al.* [5], including multiple ionization effects.
- The new ITER recommendations for impurity concentrations that depend on the average electron density, $\langle n_e \rangle$, have been implemented [6].
- ITER-recommended transport coefficients have been installed, based on the global ITER confinement database. These transport models are described in detail in the next section.
- The definition of several global parameters ($\langle T \rangle$, τ_E , etc.) has been modified to be consistent with ITER usage.

3. TRANSPORT MODELS

Since no reliable, theory-based, transport models exist at present, we are using transport coefficients which are based on the ITER *L*-mode confinement database [7] and

which reproduce the global predictions of this database. The coefficients for the thermal energy transport have the general form:

$$\chi_i = C_i \frac{a^2}{\tau_{\text{eff}}} F(r) + \chi^{\text{CH}} \quad (6)$$

and

$$\chi_e = C_e \frac{a^2}{\tau_{\text{eff}}} F(r) \quad (7)$$

In Eqs. (6) and (7) χ^{CH} is the Chang-Hinton neoclassical ion conductivity, and $\tau_{\text{eff}} = \min[\tau_{\text{OH}}, \tau_{\text{AN}}]$, where τ_{OH} is the Neo-Alcator confinement time and τ_{AN} is assumed to be equal to the power-law ITER-89 scaling law [8]

$$\tau^{\text{ITERP-89}} = 0.048 M^{0.5} I^{0.85} R^{1.2} a^{0.3} \kappa^{0.5} n_{20}^{0.1} B^{0.2} P^{-0.5} \quad (8)$$

The stored energy form of τ_E is used, in which the degradation with power is expressed in terms of the average temperature $\langle T \rangle$ [9]. The function $F(r)$ in Eqs. (6) and (7) is a form factor, and the constants C_e and C_i are adjusted so that the energy confinement time as calculated by the code, $\tau_E^{\text{code}} = H \times \tau^{\text{ITERP-89}}$, where H is the desired confinement enhancement factor for simulation of H -mode conditions.

For the particle transport we take $D = 0.25\chi_e$, and $v^{\text{pinch}} = -2D\tau/a^2$ [10]. These values give reasonable, square-root-parabolic density profiles.

3.1. First transport model

Two different forms of the profile factor $F(r)$ have been used. The first (referred henceforth as the first transport model), has a simple parabolic dependence on the minor radius:

$$F(r) = 1 + c_1 \frac{r^2}{a^2} \quad (9)$$

The constant c_1 is adjusted to describe the desired confinement properties. Although the constants c_1 and C_j ($j = i, e$) in the expressions for the local χ 's are freely adjusted in order to obtain the desired global confinement time, it is useful to compare the transport coefficients as given by Eqs. (6), (7) and (9) for different selection of these constants, with the present experimental database. For this, we compared the χ_e as predicted with the first transport model with a recent parametrization of χ_e based on a series of simultaneous measurements of the electron thermal conductivity and particle diffusivity in JET [11], and with TRANSP-determined χ_e profiles from high current and power JET *H*-mode discharges [12]. The results of this comparison are shown in Figs. 1 and 2. It can be seen, that the first transport model gives χ_e profiles that are in good agreement with the experimentally determined ones from JET, for a range of values of the constants c_1 and C_j . The values of c_1 and C_j used in the ITER simulations reported in in this work, are in the same range.

3.2. Nocentini's transport model

The second form of the profile factor $F(r)$ is based on Nocentini's suggested transport model for ITER [13], which is based on a 1-D simulation of an ASDEX discharge. According to this model, the profile factor $F(r)$ is the product of two form factors, $F_q(r)$ and $F_m(r)$. The first factor has the form

$$F_q(r) = 1 + 0.5(c_1 - 1) \left\{ 1 - \tanh [c_2(q^2 - 1)] \right\} \quad (10)$$

and enhances the diffusion coefficient in the region where $q \leq 1$, to describe the average effect of sawteeth. The second term is taken to be

$$F_m(r) = 1 + K_1 q^{K_2} \quad (11)$$

In his paper, Nocentini has provided a set of suggested values for the constants C_j , c_1 , c_2 , K_1 , and K_2 . However, in his calculations particle transport is not considered and the density profile has a fixed square-root-parabolic form, while the χ 's are *effective* χ 's (i.e. including convection). In contrast, in WHIST the particle profiles evolve self-consistently while a convective term is included explicitly in the energy transport equations. For this reason when using Nocentini's model in this work, the set of constants is adjusted independently of the recommended values in [13].

A comparison of the thermal conductivity profiles as calculated by the two transport models is shown in Fig. 3, where the normalized $\hat{\chi} \equiv \chi_j / C_j a^2 / \tau_{\text{eff}}$ is plotted versus the normalized minor radius r/a . The first two curves in Fig. 3 correspond to Nocentini's model and the first transport model respectively, with parameters resulting in similar level of transport at the edge. The transport-enhancing effect of Nocentini's expression in the region where $q \leq 1$, is clearly displayed.

4. POPCON ANALYSIS

Presenting the results of plasma transport simulations in the form of Plasma Operation CONtour (POPCON) plots, is very convenient since it offers a global picture of the operating space and it allows a characterization of possible operating points in terms of their thermal stability properties. It also makes possible a direct comparison with results obtained with 0-D models.

In this section, POPCON calculations for ITER with the WHIST code are presented for the different transport models under consideration.

4.1. POPCON construction with the WHIST code [4]

Before the results of the simulations are presented, it is important to understand how WHIST constructs the POPCON plots. In contrast to codes based on global models

which usually perform a steady-state power balance at a certain number of points in the $\langle n_e \rangle - \langle T \rangle$ space, WHIST scans the prescribed range of the $\langle n_e \rangle - \langle T \rangle$ space by evolving the plasma through several temperature sweeps at constant density. If the ramp time is long enough, the plasma can be assumed to be in quasi steady-state at all times. At selected points in density and temperature space, the global parameters of interest are evaluated from the self-consistent profiles and used to generate contour plots of auxiliary heating power, fusion power, energy confinement time, etc.

The required auxiliary heating power (supplied by neutral beams or RF sources) at each point in the $\langle n_e \rangle - \langle T \rangle$ space, is calculated from

$$P_{\text{aux}}(t) = P_{\text{aux}}^{\text{eq}}(t) + \langle \dot{W} \rangle + 3 \langle n_e \rangle \frac{T^{\text{f}} - T(t)}{\tau_{\text{fb}}} V_{\text{p}} \quad (12)$$

where the first term represents the power required to maintain equilibrium at the operating point at time t , the second term is the power required to increase the plasma temperature at the desired rate, and the last term is a feedback term with T^{f} being the desired temperature at time t . The average density during each sweep is maintained constant, by feedback on the relevant fueling source (gas puffing in these calculations).

4.2. POPCON analysis for ITER

The transport models described in section 3 have been used to generate several POPCON plots for the ITER physics phase reference parameters. The different adjustable parameters of each transport model were chosen so that the resulting global confinement time in each case was roughly twice the value of the L -mode τ_{E} , as predicted by the power-law ITER-89 global scaling law.

A series of simulations were first performed in which the line and recombination contribution to the total radiation loss from the impurities was omitted, to facilitate

comparison with the predictions of the 0-D models. In addition, the thermal alphas were not transported but assumed to be a constant fraction of the electron density (10%, the current ITER assumption). Examples from these calculations are shown in Figs. 4 to 7. In Fig. 4, contours of constant neutral beam power are shown for the first transport model with $C_i=C_e=0.08$ and $c_1=3$. A line of constant fusion power of 1000 MW is also shown on the same plot. It can be seen that operating points in the low- $\langle T \rangle$, high- $\langle n_e \rangle$ regime with the desired properties (high Q , $P_{\text{fus}} \approx 1000$ MW, below the Troyon β limit) lie in the thermally unstable regime of the operating space. Similar results are shown in Fig. 5 for Nocentini's transport model with $C_i = C_e = 0.15$, $c_1 = 2.2$, $c_2=1.40$ and $K_1=0.0$, corresponding to a flat χ near the edge (see the third curve in Fig. 3). The current profile is nearly ohmic, so q is less than 1 for $r/a \leq 0.3$, resulting to enhanced, sawteeth-driven, transport in the central region. Comparing the two POPCONs, one can see that the left side of the ignition contour corresponding to the first transport model is shifted to lower temperatures compared with that resulting from Nocentini's model. This is due to the more peaked temperature profiles predicted by the first transport model and can be seen from Figs. 6 and 7 where contours of constant ion and electron central temperature, T_{j0} , are shown for the first and Nocentini's transport models respectively. The first transport model predicts a ratio $T_0/\langle T \rangle$ of about 2.4 corresponding to a parabolic-squared temperature profile ($\langle T \rangle$ in the graphs is the density-weighted average temperature), while Nocentini's model predicts a ratio of about 1.5, corresponding to an almost parabolic temperature profile. It is also of interest to note that the peakedness of the temperature profile increases at higher temperatures. The corresponding density profiles have an almost constant peak-to-average density ratio $n_0/\langle n \rangle \approx 1.5$ for both transport models.

To study the sensitivity of these global calculations to changes in the adjustable coefficients of the transport models, a series of POPCON calculations was performed for different values of these coefficients. The results of a selected number of these simulations have been condensed in Fig. 8, where the ignition contour for each case is plotted in the

POPCON space. It can be seen that contours corresponding to the first transport model have their left side shifted to lower temperatures. Reducing transport losses can be accomplished by decreasing the magnitude of χ all over the minor radius (by reducing the value of the multiplier C_j in Eqs. 6 and 7) or by reducing the value of χ at the edge i.e. making it flatter (controlled by the coefficients c_1 in the first transport model, and mainly by K_1 in Nocentini's model). Both approaches lead to larger ignition windows, however flatter χ profiles result in wider ignition contours (i.e. the "loop" closes at higher temperatures).

Part of the motivation for these calculations has been to compare the predictions of full 1-1/2D codes with those of much simpler 0-D models. Such a comparison can be seen in Fig. 9, where ignition contours from 0-D calculations are compared with those calculated with WHIST. The three 0-D contours in Fig. 9 correspond to three different values of the temperature profile exponent α_T , 1, 2 and 3.0, used in the 0-D model. It can be seen that the agreement between the contour corresponding to Nocentini's transport model with the reference 0-D contour (corresponding to a parabolic temperature profile, $\alpha_T=1$) is quite good. The contour from the first transport model would agree better with 0-D simulations with more peaked temperature profiles.

In the calculations described above, the line radiation from the impurities has not been taken into account. However, impurity radiation can be an important part of the total radiation loss and should be accounted for. To study the effect of impurity radiation, a series of POPCON simulations was performed in which line and recombination radiation from impurities were explicitly included in the energy balance. The impurities were not transported; the impurity profiles were assumed to be proportional to the electron density profile. The impurity fractions for carbon and iron, f_C and f_{Fe} respectively, were assumed to be decreasing functions of the average electron density according to the ITER guidelines:

$$f_C(\%) = 0.9 + 0.6(0.7/n_{20})^{2.6}$$

and

$$f_{Fe}(\%) = 0.05(0.7/n_{20})^{2.3} + 0.05$$

where $n_{20} \equiv \langle n_e \rangle / 1 \times 10^{20}$. Coronal equilibrium has been assumed for the calculation of the total radiative loss.

The contribution of the impurities to the total radiation loss is significant, specially at the low temperature region of the operating space as it can be seen from Figs. 10 a-b, where contours of constant radiation power loss, with and without line and recombination contributions from impurities, are plotted in the $\langle n_e \rangle - \langle T \rangle$ space.

As a result of this enhanced loss, ignition becomes more difficult. In Fig. 11. contours of constant neutral beam power are shown for Nocentini's transport model with parameters corresponding to curve 4 in Fig. 8. It can be seen that in order to operate at the same fusion power a larger amount of NB power is required, leading to a lower Q .

Improving confinement by decreasing the multiplying constant C_j in the expression for χ ($C_j = 0.07$), leads to an ignition curve more similar to the ones obtained without line radiation losses (Fig. 12). In the same Figure, one can observe that for $\langle n_e \rangle \geq 1.5 \times 10^{20} \text{ m}^{-3}$ the contours of constant NB power "bend" inwards creating an indentation in the ignition contour. This effect is due to the poor penetration of the 1.3 MeV neutral beams of ITER at higher densities. This effect is also present when impurity radiation is not included, however it is more pronounced here due to the larger amounts of NB power required in the low temperature region of the operating space. Repeating the calculation with higher energy neutral beams ($E_{\text{beam}} = 1.8 \text{ MeV}$), it can be seen that the indentation disappears (Fig. 13). The subject of beam penetration at high densities is very important for the burn control studies, and will be revisited in a later section.

4.3. Conclusions

From the results of the global calculations described above, our first conclusion is

that operating points with the desired properties in terms of fusion power, operational constraints (β -limit, divertor conditions, etc.), and high Q , exist in the thermally unstable low- $\langle T \rangle$, high- $\langle n_e \rangle$ region of the operating space for a variety of transport models.

There is, in general, good agreement between the results obtained with WHIST and results based on 0-D models. The main differences between 1-1/2D and 0-D models are due to:

- Profile effects: Depending on the form of the χ 's, the temperature profile can be more peaked shifting the left side of the ignition contour to lower temperatures.
- Beam penetration: At high densities ($\langle n_e \rangle \geq 1.5 \times 10^{20} \text{ m}^{-3}$) the 1.3 MeV ITER beams cannot penetrate to the plasma core resulting to inefficient heating profiles and higher requirements of auxiliary heating power.
- Line and recombination radiation from impurities can be an important part of the total radiation loss, specially at low temperatures.

However when these effects are accounted for, the agreement between 0-D results and 1-1/2D codes is quite good.

5. OPERATION AT A THERMALLY UNSTABLE POINT WITH NEUTRAL BEAM FEEDBACK

The global analysis in the previous sections helped us to identify several possible operating points with the desired properties. Since however all these points lie in the thermally unstable region of the operating space, we must demonstrate that operation at these points is controllable from the thermal stability point of view.

The procedure we use to determine controllability is the following: We first select a candidate operating point in the thermally unstable region, guided by the POPCON plots of the previous sections. We then try to reach this point, starting from arbitrary but close to its neighborhood initial conditions, using neutral beam feedback according to Eq. 12. Once we reach equilibrium at this point, an arbitrary positive or negative perturbation in

the electron and ion temperatures is introduced (the origin of which is left unspecified at this stage), which we then try to stabilize with NB feedback. We are using a simple proportional feedback algorithm based on total neutron flux measurements

$$P_{nb}(t) = P_{nb}^{eq} + g_p \left(1 - \frac{\phi_n}{\phi_n^{eq}}\right) \quad (13)$$

where P_{nb}^{eq} is the neutral beam power required for thermal equilibrium at the operating point, ϕ_n^{eq} is the total neutron flux corresponding to the operating point, ϕ_n is the total neutron flux at time t and g_p is the feedback gain factor.

Control is initiated when $|\Delta\phi_n/\phi_n^{eq}| > |\Delta\phi_n/\phi_n|_{diag}$ where $|\Delta\phi_n/\phi_n|_{diag}$ represents the relative accuracy of the neutron flux measurements (projected to be about 5% for ITER). In our simulations $|\Delta\phi_n/\phi_n|_{diag}$ is usually varied in order to determine the largest perturbation we can stabilize.

We also impose a limit on the maximum ramp rate of the neutral beam system, $|\dot{P}_{nb}| \leq 10 \text{ MW/s}$, in accordance with the present hardware assumptions.

Two separate cases are presented in this section: One at medium-high densities ($\langle n_e \rangle \approx 1.5 \times 10^{20} \text{ m}^{-3}$) and one at high densities ($\langle n_e \rangle \approx 1.8 \times 10^{20} \text{ m}^{-3}$) where the poor penetration of the 1.3 MeV ITER beams can affect the effectiveness of neutral beam feedback as a burn control method.

5.1 Operation at medium densities

Operation is first considered at a thermally unstable point with density $\langle n_e \rangle = 1.5 \times 10^{20} \text{ m}^{-3}$. The transport model of Nocentini is used with $C_i = C_e = 0.08$, the rest of the parameters being the same as in the POPCON of Fig. 12. The other operating parameters of this point are: $\langle T_e \rangle = 7.99 \text{ keV}$, $\langle T_i \rangle = 7.90 \text{ keV}$, $P_{fusion} = 1098 \text{ MW}$, $P_{rad} = 139 \text{ MW}$ (includes impurity radiation), and $\tau_E = 4.6 \text{ s}$ corresponding to an

enhancement factor $H \approx 2.0$ for the global scaling. The required NB power is $P_{nb} = 26.5$ MW, resulting to a $Q \approx 42$. The resulting profiles, shown in Fig. 14a–b, are very close to the ITER reference profiles for the Physics Phase: The peak-to-average ratio for the densities $n(0)/\langle n \rangle \approx 1.5$, and the separatrix-to-average ratio $n(a_s)/\langle n \rangle \approx 1/3.9$, while the peak-to-average ratio for the temperatures is $T_0/\langle T \rangle \approx 2.1$.

Next, a 1% uniform positive perturbation is introduced, i.e. $T'(r) = T(r)(1 + \delta T)$. If no external control action is taken, the temperature rises until it reaches a thermally stable point with $\langle T_i \rangle \approx 14$ keV at a fusion power of 2790 MW (Fig. 15a–b). However, the Troyon β limit is violated within 10s from the onset of the instability (Fig. 16). During the thermal excursion, the temperature profile becomes slightly broader ($T_{e0}/\langle T_e \rangle = 1.89$ versus 2.1 at the equilibrium point).

We now try to stabilize this thermal instability using proportional neutral beam feedback. The feedback coefficient, g_p , was taken to be 200 MW after some trial-and-error, a good compromise between fast system response and avoidance of oscillations. We find that we can stabilize positive perturbations with $|\Delta\phi_n/\phi_n^{eq}|$ as high as 32%, a value well within the projected capabilities of the ITER diagnostics.

However, the actual limit on the magnitude of controllable perturbations is set by negative perturbations, since their stabilization requires that the heating system (neutral beams here), have reserve power capability [1–2]. For the maximum NB power installed in ITER (≈ 75 MW), we find that we can stabilize negative perturbations with $|\Delta\phi_n/\phi_n|$ as high as 21%. In Figs. 17 and 18, the time histories of the electron temperatures and of the NB heating power are shown for different control cases.

The above calculations are based on the feedback algorithm given by Eq. (13), in which the input signal is proportional to the total neutron flux. However, information about local changes in neutron flux will be available in ITER from collimated neutron detectors. To examine whether the performance of the control system improves if we make use of spatially resolved neutron flux information, the above calculations were repeated

using a more general control algorithm in which changes in the neutron flux from two different parts of the plasma (characteristic of central and peripheral conditions) were monitored, besides the global flux:

$$P_{nb}(t) = P_{nb}^{eq} + g_p \left(1 - \frac{\phi_n}{\phi_n^{eq}}\right) + g'_p \left((1-f_d) \left(1 - \frac{\phi_{n0}}{\phi_{n0}^{eq}}\right) + f_d \left(1 - \frac{\phi_{na}}{\phi_{na}^{eq}}\right) \right)$$

In this algorithm, the subscripts 0 and a denote flux values of the central and peripheral region respectively, while f_d is an adjustable parameter. Control action was initiated when the value of $|\Delta\phi_n/\phi_n^{eq}|$ from any part of the plasma exceeded the imposed diagnostic threshold. We found that during thermal excursions, the neutron flux in the outer part of the plasma ($r/a > 0.5$) increases faster than the total neutron flux. Using this algorithm, we could stabilize negative perturbations with $|\Delta\phi_n/\phi_n|$ as high as 26%. This result underlines the importance of using spatially-resolved observables in our control scenarios.

5.2. Operation at high densities

During the course of our POPCON simulations it became apparent that at high densities, the 1.3 MeV ITER neutral beams (optimized for the Technology Phase machine where NB current drive is one of the main goals and for that reason lower densities are expected), cannot penetrate to the center of the plasma. A series of stand-alone beam-penetration calculations with the Monte Carlo neutral beam deposition code NFREYA [14] using the design parameters of the ITER beamline [15] and with MHD equilibrium generated by the NEQ code demonstrates this very clearly: In Fig. 19, the NB deposition profile, $H(\psi)$, is plotted versus the normalized poloidal flux $\tilde{\psi}$, for different densities. It can be seen that for $\langle n_e \rangle \geq 1.6 \times 10^{20} \text{ m}^{-3}$ beam penetration is poor and the resulting heating profile is peaked off-axis. This spatial mismatch between the mechanism responsible for the thermal instability (fusion power) that is usually peaked at the center,

and the power that is used for control (neutral beams) that is peaked off-axis at high densities, raises the question whether control with NB modulation is still possible at these conditions.

To check this, a series of burn control simulations at high densities ($\langle n_e \rangle \approx 1.8 \times 10^{20} \text{ m}^{-3}$) were performed. Operation at such high densities may be required (provided that it is compatible with the density limit), if energy confinement is better than expected, in which case the reference operating point at the desired fusion power would lie within the ignited region of the $\langle n_e \rangle - \langle T \rangle$ space. The equilibrium point has the following parameters: $\langle T_e \rangle = 6.47 \text{ keV}$, $\langle T_i \rangle = 6.42 \text{ keV}$, $P_{\text{fusion}} = 1094 \text{ MW}$, $P_{\text{rad}} = 219 \text{ MW}$ (including impurity radiation), and $P_{\text{nb}} = 59 \text{ MW}$ resulting to a Q of 16. The Nocentini's transport model is used with $C_i = C_e = 0.075$, resulting to a confinement time, $\tau_E = 4.6 \text{ s}$, corresponding to an H-factor of 2.65 for the global scaling. The energy balance of the plasma at the equilibrium point and the off-axis peaking of the NB heating power, can be seen in Fig. 20.

As before, positive and negative temperature perturbations are introduced into the system and stabilization is attempted with NB feedback for different values of the diagnostic uncertainty, $|\Delta \phi_n / \phi_n|_{\text{diag}}$. Our calculations indicate that although positive perturbations with $|\Delta \phi_n / \phi_n|$ up to 15% can still be controlled, stabilization of negative perturbations is more problematic. Perturbations with $|\Delta \phi_n / \phi_n| = 10\%$ cannot be stabilized, at least with the present technological constraints on the maximum ramp rate and maximum installed NB power. The temperature profile collapses after the introduction of the negative temperature perturbation. That this is in fact due to the poor penetration of the neutral beams, can be seen by repeating the calculations using higher energy beams (1.8 MeV) that can penetrate to the center of the plasma: Then, it is found that control of this negative perturbation is possible.

6. CONCLUSIONS

Our 1-1/2D transport simulations of ITER burn control scenarios appear to confirm earlier 0-D results that operation of ITER at a sub-ignited, thermally unstable point is feasible with feedback on the neutral beam heating source based on total neutron flux measurements. The present limitations on the maximum available neutral beam power (≈ 75 MW) and the maximum ramp rate of the heating system (10 MW/s) do not appear to impede the control action, at least for the reference operating point with $\langle n_e \rangle = 1.5 \times 10^{20} \text{ m}^{-3}$. The response of the control system can be improved if use of spatially-resolved neutron flux measurements is made.

The poor penetration of the 1.3 MeV ITER beams at high densities makes control at these conditions difficult. Our calculations indicate that it is not possible to control negative perturbations for $\langle n_e \rangle \geq 1.75 \times 10^{20} \text{ m}^{-3}$. Therefore, alternative control methods should be studied for such points if operation at such high densities is desired.

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χ COMPARISON WITH JET RESULTS

FIG. 1. Comparison of χ_e from a fit to JET data with the first transport model. The error bar corresponds to the 20% error on the measurement of χ_e .

COMPARISON WITH JET H-MODE

FIG. 2. Comparison of TRANSP-determined χ_e profile from a JET *H*-mode pulse, with the first transport model.

FIG. 3. Characteristic normalized χ_e profiles for different parameters of the two transport models.

FIG. 4. Contours of constant NB power for the first transport model with $C_i=C_e=0.08$ and $c_i=3$. Also shown is the 1000 MW fusion power contour.

FIG. 5. Contours of constant NB power for Nocentini's transport model with $C_i=C_e=0.15$, $c_1=2.2$, $c_2=1.4$ and $K_1=0.0$.

FIG. 6. Contours of constant electron central temperatures for the first transport model.

FIG. 7. Contours of constant electron central temperatures for Nocentini's transport model.

FIG. 8. Ignition contours for different transport models and parameters:

- 1: 1st model, $C_j=0.12$, $c_1=0$
- 2: 2nd model, $C_j=0.15$, $c_1=2.20$, $c_2=1.40$, $K_1=0.0$
- 3: 1st model, $C_j=0.08$, $c_1=2.0$
- 4: 2nd model, $C_j=0.08$, $c_1=2.20$, $c_2=1.40$, $K_1=0.3$, $K_2=2$
- 5: 1st model, $C_j=0.08$, $c_1=3.0$

FIG. 9. Comparison of ignition contours from 0-D calculations with ignition contours from WHIST.

FIG. 10. Contours of constant radiation power. (a): Line and recombination radiation from impurities is not included. (b): Line and recombination radiation from impurities is included.

CONTOURS OF CONSTANT NB POWER (MW)

Nocentini's Model, $C_j = 0.08$

FIG. 11. Contours of constant NB power for Nocentini's transport model with $C_j=0.08$, $c_1=2.20$, $c_2=1.40$, $K_1=0.3$, $K_2=2$. Line and recombination radiation from impurities is included.

Nocentini's model, $E_{\text{beam}} = 1.3 \text{ MeV}$

FIG. 12. Contours of constant NB power using Nocentini's transport model with the same parameters as before (Fig. 11), but $C_j=0.07$. Line and recombination radiation from impurities is included.

FIG. 13. Contours of constant NB power using Nocentini's model with the same parameters as Fig. 12, but $E_{\text{beam}} = 1.8 \text{ MeV}$.

FIG. 14. Steady-state plasma profiles at a medium density operating point. (a): electron temperature, (b): ion density.

Temperature evolution

Fusion Power

FIG. 15. Uncontrolled time evolution of (a) the electron temperature profile and (b) the fusion power, following a positive temperature perturbation.

FIG. 16. Time evolution of the plasma β following a positive temperature perturbation.

FIG. 17. Time evolution of the central and average electron temperatures with and without burn control, following a positive temperature perturbation.

FIG. 18. Evolution of the applied NB heating power, in response to positive and negative perturbations of different magnitudes. The NB power ramp rate is assumed to be ± 10 MW/s.

NEUTRAL BEAM DEPOSITION PROFILES

FIG. 19. Neutral beam deposition profiles for different densities, for $E_{\text{beam}} = 1.3$ MeV.

FIG. 20. Electron energy balance at the high density equilibrium point.

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