CORSICA Modelling of ITER Hybrid Operation Scenarios

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Abstract The hybrid operating mode observed in several tokamaks is characterized by further enhancement over the high plasma confinement (H-mode) associated with reduced MHD instabilities linked to a stationary flat safety factor \(q\) profile in the core region. The proposed ITER hybrid operation is currently aiming at operating for a long burn duration (>1000s) with a moderate fusion power multiplication factor, \(Q\), of at least 5. This paper presents candidate ITER hybrid operation scenarios developed using a free-boundary transport modelling code, CORSICA, taking relevant physics and engineering constraints into account. First, we have developed a 12.5MA ITER hybrid operation scenario by tailoring the 15MA ITER inductive H-mode scenario. Second, we have studied accessible operation conditions and achievable range of plasma parameters. ITER operation capability for avoiding the poloidal field (PF) coil current, field and force limits are examined by applying different current ramp rates, flat-top plasma currents and densities, and pre-magnetization of the PF coils. Various combinations of heating and current drive (H&CD) schemes have been applied to investigate several physics issues, such as the plasma current density profile tailoring, enhancement of the plasma energy confinement and fusion power generation. A parameterized edge pedestal model based on EPED1 was recently added to the CORSICA code and applied to hybrid scenarios. Finally, fully self-consistent free-boundary transport simulations have been performed to provide information on the PF coil voltage demands and to study the controllability with the ITER controllers.

1. Introduction

ITER will explore advanced tokamak operations, such as the hybrid and steady-state modes, to demonstrate the feasibility of fusion energy production at a reactor scale with deuterium and tritium fuels. Preparing feasible advanced operation scenarios requires an integrated full tokamak discharge simulator, such as CORSICA \([1-2]\), DINA-CH/CRONOS \([3-4]\), TSC/PTTRANSP \([5-6]\), JINTRAC/CREATE-NL \([7]\) and ETS \([8]\). Among these integrated tokamak modelling activities, the CORSICA code can provide self-consistent evolution of the free-boundary plasma equilibrium and transport using its fully implicit code coupling scheme, as well as high computation performance which is essential for simulating entire operation phases over 1000 seconds. However, the CORSICA code used only a few realistic source modules which were not sufficient for studying ITER advanced operation scenarios. In this work, several realistic source modules for heating and current drive (H&CD), such as the neutral beam (NB) injection, electron and ion cyclotron (EC&IC), and lower hybrid (LH), are either upgraded or newly added to the CORSICA code using the latest source configurations of ITER. We have then developed several ITER hybrid operation scenarios, including relevant physics and engineering constraints on the poloidal field (PF) coil system and controllers. The ITER hybrid operation scenarios have been studied focusing on achieving physics goals, such as the fusion power multiplication factor, \(Q\), and plasma burn duration. The integrated tokamak discharge modelling capability of the CORSICA code has been continuously improved during this ITER hybrid operation scenario study.

In section 2, we present a 12.5MA ITER hybrid operation scenario, which is used as a reference case for comparison, by tailoring the 15MA ITER inductive H-mode scenario. In section 3, we have studied accessible ITER hybrid operation conditions and achievable range of plasma parameters. Operational capabilities of avoiding ITER PF coil current, field and force limits are examined by applying different operation conditions and techniques. Several physics issues, such as the plasma current density profile tailoring, enhancement of the
plasma energy confinement and fusion power generation have been investigated by applying various combinations of external H&CD. A parameterized edge pedestal model based on EPED1 [9] was recently added to the CORSICA code and applied to hybrid scenarios. In section 4, fully self-consistent free-boundary transport simulations have been performed to provide information on the PF coil voltage demands and to study the controllability with the ITER controllers. A summary and discussion are presented in section 5.

2. Reference simulation of ITER hybrid scenario

Hybrid operating mode observed in several tokamaks [10-12] is characterized by further confinement enhancement over H-mode plasma operation. Although the physics understanding of the self-regulating mechanism [13] is incomplete, it appears to be associated with low-level MHD instabilities for a stationary flat $q$ profile > 1.0 in the core region. In ITER, this advanced operational capability will be demonstrated at a lower flat-top plasma current of about 11~13MA with a reduced flat-top plasma density compared with 15MA inductive H-mode operation [2]. The ITER hybrid operation is currently aiming at operating the plasma for a long burn duration (>1000s) with a moderate Q of at least 5. The key to hybrid operation is to tailor the plasma current density profile without triggering the sawtooth crashes before the start of flat-top phase. However, it is a challenging issue due to the limited number of available auxiliary H&CD systems and their limited accessibility.

We have developed several ITER hybrid operation scenarios by tailoring the 15MA ITER inductive H-mode scenario [2] and expanding the flat-top burn duration up to 1300s as shown in FIG. 1. A reference 12.5MA hybrid scenario used for comparison with its variants is presented in this section. The electron density profile evolution prescribed with a parabolic shape is almost flat in the core region. The flat-top electron density at the center is assumed to be 85% of the Greenwald density limit and the density at the boundary is assumed to be 35% of the central density. The deuterium and tritium ion density ratio is assumed to be 50:50, taking the neutral beam injected deuterium ions into account. These main ion densities become a little hollow due to the quasi-neutrality constraints, as fusion-born alpha particles are produced. The alpha particle density profile becomes peaked at the centre. Argon (Ar) and Beryllium (Be) are used as impurity species and self-consistently calculated with the effective charge number, $Z_{\text{eff}}$, satisfying the quasi-neutrality conditions. The evolution of the effective charge number is given by the Lukash’s formulary [14]. The neutral beam driven fast ions are also computed by a neutral beam injection code imbedded in CORSICA, NFREYA [15]. The losses of fast ions and beam driven currents are computed from a first orbit model coupled with NFREYA.

The plasma current is ramped up in 60 seconds and an L-H confinement mode transition is assumed at about 2/3 of the current ramp-up. The plasma is assumed to start with a larger bore limited on the inboard wall, and then it is allowed to grow along with the plasma current and experience a shape transition from the limited to diverted configuration when the

![FIG. 1. Time traces of the plasma current, driven currents, bootstrap current, volume averaged electron density and effective charge number. 12.5MA ITER hybrid operation scenario (reference case)](image-url)
plasma current is about 4.5MA at about 15 seconds (see FIG. 2). The equatorial EC launchers are switched on after the shape transition, and then a neutral beam is switched on at t=40s and assumed to trigger an L-H confinement mode transition. 33MW of NB and 20MW of EC power are applied starting from the start of the flat-top phase and the plasma is self-heated by fusion-born alpha particles. The electron density is further increased for about 30 seconds after the start of flat-top, finally approaching the central electron density of 8.5e19 m^{-3}. After the flat-top phase, the plasma current is ramped down in 210 seconds, assuming an H-L confinement mode transition at about 1/3 of the current ramp-down. 6.67MW of EC power has been maintained even after the H-L confinement mode transition to make a smooth transition with a reduced plasma beta drop.

The heat transport is computed using the Coppi-Tang transport model [16]. The pedestal temperatures are determined by scaling down the heat conductivities outside $\rho_{\text{tor}} = 0.95$. The electron temperature at the pedestal top is usually about 3~4 keV and this is slightly less than value obtained in the 15MA inductive H-mode scenarios [2]. Triggering of sawtooth crashes is modelled by changing the heat conductivities and plasma resistivity inside the inversion radius, when the minimum safety factor, $q_{\text{min}}$, becomes less than 0.97. Two off-axis neutral beams (poloidal angle = -3.331 degree) are assumed with recent design of the ITER neutral beam injection system. The plasma current density profile is optimized using 20MW of EC H&CD applied using three launchers, upper and lower equatorial launchers, and an upper launcher with lower steering mirror. Each launcher delivers about 6.67MW and the equatorial launchers provide EC driven currents in co-current direction at about $\rho_{\text{tor}} = 0.2~0.4$ and the upper launcher provides far-off-axis current at about $\rho_{\text{tor}} = 0.4~0.6$, but with reduced CD efficiency. The poloidal and toroidal EC mirror angles are determined by trial and error, based on an automatic scan of the electron heat deposition location and EC driven current, using a ray-tracing code, TORAY-GA [17].

In this hybrid scenario, the fusion power multiplication factor, Q, was above 9.0 and the alpha particle self-heating power was about 100MW during the plasma burn. This higher Q value is obtained thanks to the plasma confinement enhancement at a relatively low auxiliary heating power of 53MW. The achieved confinement enhancement factor with respect to H-mode confinement, $H_{98}$, was 1.2~1.3. The internal inductance was 0.70~0.75 which is good for stabilization of the vertical instability. High normalized and poloidal betas of about 2.5 and 0.82, respectively, were obtained. The bootstrap current was about 3.8MA and the neutral beam and EC driven currents were about 2.5MA and 0.4MA, respectively. The safety factor profile, which was initially slightly reversed, became flat in the core region ($\rho_{\text{tor}} < 0.4$) due to sawteeth triggered when $q_{\text{min}} < 0.97$. A non-inductive current fraction of about 54% was achieved, but it was not enough to maintain the central $q$ value > 1.0.

The CS coils were well within their coil current, force and field limits. The PF6 coil current briefly violated its limit at the end of the ramp-up phase (see FIG. 3). However, this PF6 coil current limit (~18.8MA) is given with an assumption that the maximum B-field
applied at the coil ($B_{\text{max}}$) is 6.5T, which is different from the actual PF6 coil current limit (~22MA) given for $B_{\text{max}}$=6.4T [18]. This PF6 coil current limit can be further increased to about 23–24MA for $B_{\text{max}}$=6.8T if a 0.4K sub-cooling of the coil is applied [18]. Although the PF6 coil does not violate its actual current limit (>18.8MA) in this simulation, it is very close to the limit. Therefore, potential techniques useful for avoiding these limits are investigated in section 3.1. The PF2 coil violated its coil current, force and field limits during the ramp-down phase (see FIG. 3). This violation can be successfully avoided by modifying the ramp-down shape evolution (see section 3.1). The total poloidal flux consumption and imbalance current in the VS1 position control system were well within their limits.

3. Achievable range of plasma parameters in ITER hybrid operation

We have studied various operation conditions and the achievable range of plasma parameters in ITER hybrid operation. ITER’s operation capability for avoiding the coil current, field and force limits are examined by applying different current ramp rates, and flat-top plasma currents and densities. Modifications to the ramp-down shape evolution and PF coil pre-magnetization [6] were studied to further optimize the evolution of the PF2 and PF6 coil currents within their limits. Various combinations of heating and current drive schemes have been applied to investigate several physics issues, such as the plasma current density profile tailoring, enhancement of the plasma energy confinement, fusion power generation and poloidal flux consumption. A parameterized edge pedestal model based on EPED1 [9] was recently added to the CORSICA code and applied to hybrid scenarios.

3.1. Avoiding coil current limits and parameter studies

The PF6 coil current limit at the end of the ramp-up phase can be avoided either by allowing additional consumption of volt-seconds or by shifting the initial flux state applied for the tokamak discharge. Both are feasible without significant challenges on the PF coil and control systems, compared to the requirements on avoiding the CSI1 coil current limit in the 15MA inductive H-mode operation scenario [2-3], in which the consumption of volt-seconds should
be minimized while the initial flux state is already very close to its limit.

First, the volt-seconds of the PF6 coil can be additionally consumed by applying a small bore start-up, delaying the application of auxiliary heating power and/or reducing the current ramp-up rate. Simulations with different current ramp-up rates are compared in FIG. 4. The PF6 coil current limit was avoided with a lower current ramp-up rate which increases the resistive volt-second consumption. However, those methods which allow the plasma to consume additional poloidal flux can reduce the duration of the plasma burn. Redistributing the demand on volt-second consumption to the other coils, by modifying the shape evolution [3], could be an alternative solution. Secondly, advancing the pre-magnetization [6] which shifts the initial flux state and PF coil currents can be applied. This method allows the PF6 coil to avoid its limit and prevents the plasma from consuming additional poloidal flux. However, the poloidal flux available for operation is already reduced from the beginning by advancing the pre-magnetization. This method would be useful only when the available poloidal flux is large enough compared with the operational requirement. Pre-magnetization modelling capability of CORSICA has been developed by solving the initial free-boundary equilibrium with an additional constraint on the flux state. We have studied several cases in which the initial flux state is reduced by either 20Wb or 40Wb from the value used in the reference simulation, and these are compared in FIG. 5. The initial PF coil currents were shifted to produce the same plasma shape but with different initial flux states. The PF6 coil current limit was avoided and the plasma profiles and parameters remained similar with those shown in the reference simulation. The evolution of the flux state was shifted without consuming additional poloidal flux, as shown in FIG. 5.

We have also studied several cases with different flat-top currents (11.5MA and 10.5MA). In these simulations, the CS and PF coil currents were away from their limits due to the reduced demands on the volt-second consumption during the current ramp-up. However, the plasma temperatures and alpha particle self-heating power were reduced due to the dependence of the plasma confinement on the plasma current. Therefore, the fusion power multiplication factor became smaller than that achieved in the reference case (12.5MA). The difference in the ohmic heating powers was negligible. A scan on the flat-top density has also been performed. As the flat-top plasma current becomes smaller, the flat-top density should be reduced to avoid the Greenwald density limit. However, low flat-top density is not favourable for achieving high fusion power multiplication factor, due to much less frequent fusion reactions. The fusion power multiplication factor was about 5.0 when the flat-top electron density at the centre was reduced down to 6e19m⁻³.

The violations of the PF2 coil current, force and field limits have also been investigated. When there were such violations during the ramp-down phase, the plasma was in L-mode.

![FIG. 5. Time traces of the PF1 and PF6 coil currents (top) and flux state (bottom). 12.5MA ITER hybrid operation scenarios with/without advancing the pre-magnetization.](image)
with no auxiliary heating. The plasma current was about 3.5MA. Therefore, these violations appear not to impose a serious control challenge on the ITER plasma control system [19]. However, another simulation with different plasma shape evolution has been investigated (see FIG. 6) because it has been identified that these violations are caused by the forced strong shape transition during the ramp-down phase. Although it appears that there are certain degrees of freedom with the ramp-down shape evolution, the current ramp-down should be further investigated taking the thermal heat load constraints on the plasma facing components into account.

3.2. Application of various combinations of H&CD systems

Application of auxiliary H&CD plays a critical role in achieving the improved confinement in hybrid operating mode. Triggering the L-H transition, tailoring the plasma current density profile, and generating the alpha heating power rely significantly on possible combinations of auxiliary H&CD systems and their operational capabilities. Several simulations have been done to study the influence of various H&CD schemes on ITER hybrid operation. Four different auxiliary H&CD powers, 53MW (reference), 73MW, 93MW and 60MW (without NB) were studied. In three 73MW simulations, 20MW of additional power (either EC/IC/LH) is added to the reference H&CD scheme (33MW NB & 20MW EC). In the 93MW case, 20MW of LH and 20MW of IC were added to the reference H&CD scheme. In two 60MW cases, 40MW of EC was applied with 20MW of either IC or LH. Both on-axis electron and ion heat deposition are assumed for 53MHz of IC wave frequency and 1/3 of power assumed for heating electrons. No IC driven current is assumed at this frequency. LH heat deposition and driven current profiles were computed using a ray-tracing code, LSC [20], assuming only 1 launcher with \( n_0 = 2.2 \) for simplicity.

Comparing the simulation results (see FIG. 7), several general tendencies have been observed. First, when far off-axis LHCD was applied, the internal inductance was effectively reduced and the safety factor profile was maintained over 1.0 until the end of the flat-top phase. The far off-axis LH driven current (\( \rho_{\text{tor}} \approx 0.8 \)) was effective in slowing down the evolution of \( q \) values in the core region with a long time-scale (~1000s). Second, the demand on the inductively driven current was reduced with larger non-inductively driven currents. Therefore, the poloidal flux consumption was reduced (see FIG. 7). Third, at higher auxiliary heating power, larger bootstrap current and alpha particle self-heating power were obtained but at a lower fusion power multiplication factor. In the cases with 60MW auxiliary power (without NB), the achieved plasma parameters were not so different with those obtained in the reference simulation, except the poloidal flux consumption which was higher due to reduced non-inductively driven current.
3.3. Pedestal profile modelling

A pedestal model based on EPED1 parameterization is coupled to the CORSICA code for better estimate of the pedestal width and height. The parameterized EPED1 model has 9 input parameters, $I_p$, $n_{e,\text{ped}}$, $Z_{\text{eff}}$, $\beta_n$, R, a, $\kappa$, $\delta$ and $B_T$, and 4 output values, widths and critical pressures at the pedestal ($\psi_{\text{ped}}$) or pedestal top ($\psi_{\text{top}}=1-3/2*(1-\psi_{\text{ped}})$) [21]. A hyperbolic tangent function is used to describe the pedestal density profile and a parabolic function is used to describe the core density profile. The pedestal electron density is assumed to be 80% of the central electron density. In the parameterized EPED1 model, the output values are obtained by performing linear interpolations and extrapolation (up to ±50% of the available range). When the parameterized EPED1 model was applied, the feedback controlled pedestal top pressure and width were respectively higher and larger than those assumed in the reference simulation. This implies that the previously conducted ITER hybrid operation simulations underestimated the stability-based limits for these parameters and even better performance can be achieved. The total bootstrap current was close to that obtained in the reference simulation, although the bootstrap current density in the edge region was much higher. If the same central density is used, a higher density (pressure) gradient applied in the pedestal region inevitably reduces the density (pressure) gradient in the core region.

4. Full free-boundary control simulation

Self-consistent free-boundary transport simulations have been performed to provide information on the PF coil voltage demands and to study the controllability with the ITER controllers, JCT2001 and VS1. In these simulations, the coil currents obtained from the prescribed boundary transport simulations were used as the reference coil currents for the controllers. The PF coil currents were feedback controlled well around the reference coil currents and the shape evolution and achieved plasma parameters were very close to those obtained from the prescribed boundary transport simulations (see FIG. 8). The power supply voltage demands were well within their limits, because the PF coil voltage limits were used as saturation voltages for the power supply models. The plasma stability dynamics has been studied by repeating this simulation with a vertical displacement event triggered by disconnecting the feedback control loop at about $t=350s$. The vertical instability growth rate computed using the logarithmic estimation method [3] was about $3-4\,s^{-1}$. This value is smaller than that obtained from the 15MA H-mode scenarios [22], and this implies that the plasma can be vertically stabilized with sufficient control margins in ITER hybrid operation.

5. Summary and discussion
ITER hybrid scenarios have been studied using an advanced free-boundary transport simulation code, CORSICA, including relevant physics, engineering constraints, and ITER design parameters. This study shows that operating ITER in the hybrid operating mode is feasible. Optimization of the scenarios would require further investigation. A study on feasible ITER steady-state scenarios and kinetic profile control issues has been recently initiated based on this work. The improved tokamak discharge modelling capability achieved in this work will be useful for supporting the ITER Plasma Control System (PCS) and Integrated Modelling (IM) development.

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FIG 8. Time traces of the CS (top) and PF (bottom) coil currents. (dotted lines represent the reference currents used for controllers)