Plasma current ramp-up strategies for first wall heat load reduction in ITER

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1. Introduction

In the Standard ITER plasma current (I_p) ramp-up scenarios [1-5], the plasma is limited on the central column beryllium first wall panels (FWP) in the early phase, with the transition to X-point (divertor) configuration made as early as possible, satisfying the constraints of acceptable FWP heat loads and minimizing poloidal flux consumption such that burn duration is maximized. The transition is typically made when $I_p \approx 3.5$ MA (ramp-up rate ≈ 0.2 MA·s^{-1}). Recently, however, it has become clear that the near scrape-off layer heat flux channel width is likely to be much narrower than previously suspected [1, 2], posing a potential problem for wall heat loading if I_p is too high in limiter configuration and FWP alignment is not tightly controlled. Whilst efforts are now underway to improve on the original wall alignment targets, it is also important to examine different strategies for I_p ramp-up phase in the event that heat loads are still too high.

This paper presents an alternative current ramp-up scheme in which a reduction of conductive heat losses, P_{con}, is sought by reducing the value of I_p during the limiter configuration. In this alternative scheme, I_p is increased up to ≈ 2 MA in circular plasma configuration at the same rate as in the standard scenarios, but is then maintained constant for ≈ 8 s. During this time, the plasma minor radius, a, and elongation, k, are increased in preparation for the X-point transition and formation of divertor magnetic configuration. This Alternative ramp-up scheme is compared with the Standard approach for the baseline ITER 15 MA DT scenario from the point of view of maximum duration of burn.

2. Control of plasma current, position and shape in ITER

The ITER Poloidal Field (PF) system comprises the 6 module central solenoid (CS) and 6 outer PF coils (Fig. 1). With the exception of the two central CS modules (which are connected in series in the circuit CS1), all PF coils and CS modules have independent power supplies.

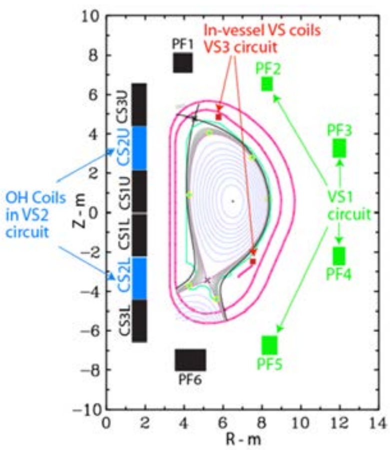

Fig. 1: PF system, vacuum vessel, first wall/divertor and magnetic configuration of 15 MA plasma.

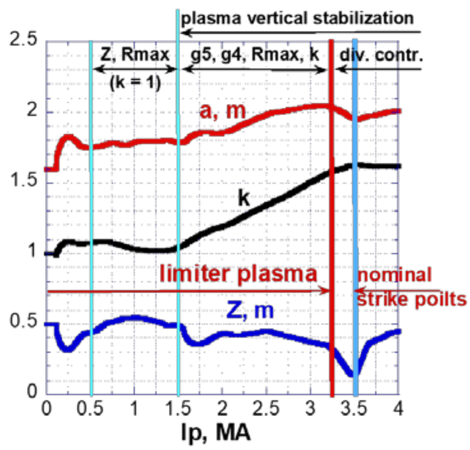

Fig. 2: Plasma feedback control at the standard I_p ramp-up.

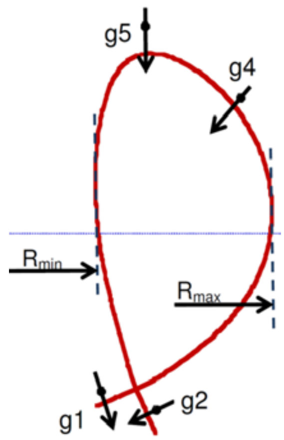

Fig. 3: Separatrix parameters controlled in divertor configuration.

At plasma start-up, the PF system uses a set of resistors in the Switching Network Units (SNU) of the 5 CS circuits and those of the two PF coils nearest the CS (PF1 and PF6), as well as pre-programmed voltages of the AC/DC converters in all 11 CS and PF coil circuits. The SNU provide the main fraction of the voltages required in these circuits for plasma start-up. All SNU, except for that in the CS1 circuit, are switched off when $I_p \approx 1.5$ MA.

The feedback control of I_p, position and shape is provided by the CS and PF coils. Plasma vertical stabilization (VS) is provided by the vertical stabilization system, varying the current in the VS in-vessel coils (VS3 circuit) shown in Fig. 1 [3]. The circuit VS1, which uses a varying differential current in (PF2+PF3) and (PF4+PF5), reduces the averaged current in the VS3 circuit over longer timescales. The plasma feedback control in the standard I_p ramp-up scenarios of [4-6] has the following phases (Figs. 2, 3).

1) $I_p < 0.5$ MA (plasma initiation): feedforward control, circular plasma cross-section ($k \approx 1$),
2) 0.5 MA < I_p < 1.5 MA (limiter controller): feedback control of I_p, Z, R_{max}, k ($k_{target} = 1$),
3) 1.5 MA < I_p < 3.3 MA (limiter controller): feedback control of I_p, $g5$, $g4$, R_{max}, k (k_{target} increases from 1 to 1.6) and plasma vertical stabilization,
4) 3.3 MA < I_p < 3.5 MA (transition from the limiter controller to divertor controller),
5) Divertor controller: feedback control of I_p and six plasma shape parameters - displacement of separatrix in the directions $g1$, $g2$, $g4$, $g5$, R_{min}, R_{max} and plasma vertical stabilization (Fig. 3).

3. DINA simulations of 15 MA DT scenarios with Standard and Alternative I_p ramp-up

Simulations of 15 MA DT scenarios (fusion power P_{fus} = 500 MW, Q = 10) with the Standard and Alternative I_p ramp-up have been performed with the DINA code. The code comprises a 2D free boundary plasma equilibrium solver and a 1D plasma transport model (0D for I_p < 1.5 MA). The simulations take into account eddy currents in the vacuum vessel (shown by purple lines in Fig. 1) and engineering limits imposed on the coils and their power supplies. Moreover, the limits imposed on position of the inner separatrix relative to the first wall and divertor (green boundary in Fig. 1), and on the minimum distance between the inner and outer separatrices are taken into account.

In the simulations, beryllium (Be), tungsten (W) and neon (Ne) impurities are considered. The impurity content, $\gamma_{imp} = n_{imp}/n_e$, is independent of the magnetic surface coordinate. During the limiter phase, only Be is considered with $\gamma_{Be} = n_{Be}/n_e$ decreasing from 0.1 to 0.07. During the I_p flattop and burn, $\gamma_{Be} = 2 \cdot 10^{-3}$, $\gamma_W = n_W/n_e = 1 \cdot 10^{-5}$ and $\gamma_{Ne} = n_{Ne}/n_e = 2 \cdot 10^{-3}$.

It should be noted that, for a given plasma transport model, an increase in the I_p ramp-up rate reduces the inductive and resistive poloidal magnetic flux consumption during the current ramp, which increases the magnetic flux swing available for burn and the burn duration. However, the ramp-up rate increase also reduces the plasma internal inductance, l_i, at the start of the I_p flattop, t_{SOF}. This increases the maximum value of the magnetic field on the conductor of coil PF6, $max(B_{PF6})$, which has an engineering limit of 6.4 T. Therefore, for a fair comparison of maximum duration of burn in scenarios with different schemes of I_p ramp-up, t_{SOF} should be adjusted in preliminary simulations to obtain the same value of $max(B_{PF6})$. In the scenario with Alternative I_p ramp-up, t_{SOF} was adjusted to obtain $max(B_{PF6}) \approx 6$ T (t_{SOF} = 80 s) as in the Standard I_p ramp-up scenario (t_{SOF} = 65 s).

The start of burn (t_{SOB}) is defined as the time when P_{fus} increases to 500 MW. The end of burn (t_{EOB}) corresponds to the time when the CS1 coil current increases to 44 kA (the engineering limit is 45 kA). In both scenarios, I_p ramp-up was simulated assuming $\langle n_e \rangle / n_G \approx 0.35$, where $\langle n_e \rangle$ and n_G are the plasma volume averaged electron density and the "Greenwald" limit.

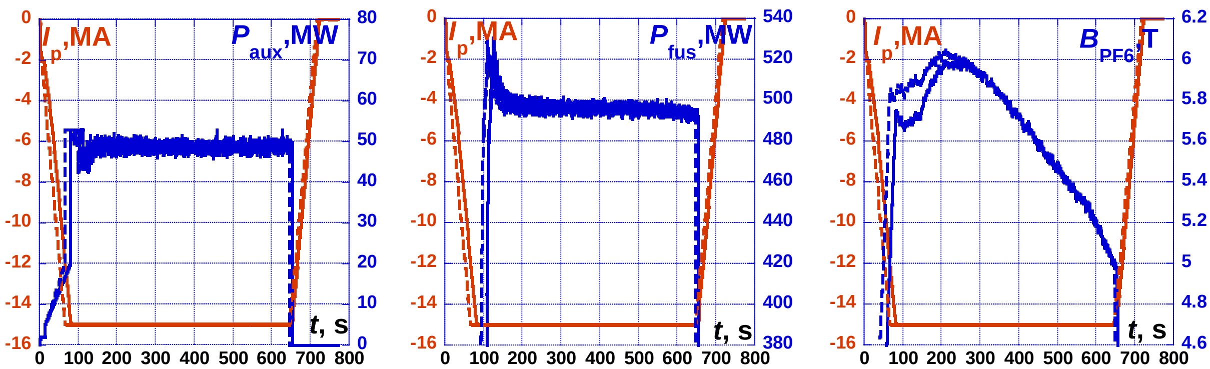

Fig. 4: The plasma current (I_p), the power of auxiliary heating (P_{aux}), the fusion power (P_{fus}) and the maximum value of magnetic field on the conductor the PF6 coil (B_{PF6}).

Fig. 4 shows I_p, P_{aux}, P_{fus} B_{PF6} in the two I_p ramp-up scenarios (Alternative (solid curves) and Standard (dashed curves)). Assuming similar "Greenwald" ratio, $<n_e>/n_G$, and adjusting t_{SOF} to obtain similar $max(B_{PF6})$, both I_p ramp-up schemes lead to about the same burn duration in 15 MA baseline DT scenarios. The Alternative I_p ramp-up reduces the maximum duration of burn by only ≈ 9 s relative to the Standard scheme (less than 2%). In both schemes a modest amount (2 MW) of additional (electron cyclotron resonance) heating has been added during the limiter phase. This provides some stability against radiation collapse of the plasma in these very early phases and also helps to increase the final burn duration. In practice, this level of P_{aux} will of course be adjustable and could be reduced if the FWP heat loads are too high.

Fig. 5 shows plasma parameters over the first 25 seconds of these scenarios. Solid and dashed curves show the scenarios with Alternative and Standard I_p ramp-ups with the last limiter configurations at 2 MA, t ≈ 10.4 s and at 3.3 MA, t ≈ 11.4 s respectively. The Alternative I_p ramp-up with 2 MA plateau allows significant reduction of the heat loads on the central column FWP (P_{con} is reduced by about 1 MW).

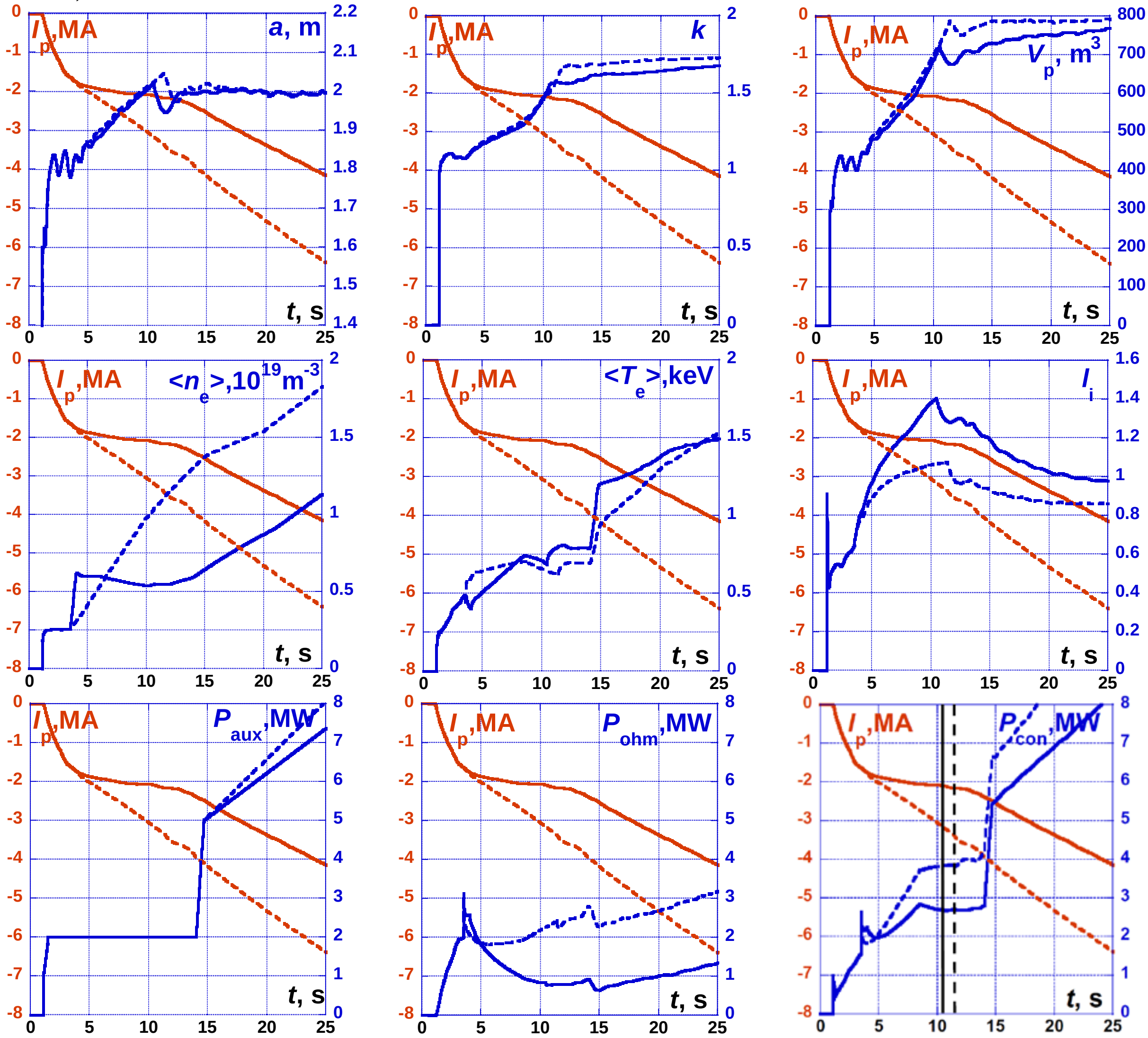

Fig. 5: The plasma current (I_p), minor radius (a), elongation (k) and volume (V_p), volume averaged electron density ($<n_e>$) and volume averaged temperature ($<T_e>$), the plasma internal inductance (l_i), the powers of auxiliary (P_{aux}) and ohmic (P_{ohm}) heating, the power of conductive heat losses through the plasma boundary (P_{con}). Vertical black lines on the last figure show the times corresponding to the last limiter configuration.

Fig. 6 shows the voltages produced by converters of the PF and CS coil circuits. Blue and red curves correspond to the Standard and Alternative ramp-up schemes. The voltage engineering limits are 3.15 kV for the circuits PF2, PF3, PF4, PF5 (three converters) and 2.1 kV for the circuits PF1, PF6 (two converters), 2.1 kV in all CS circuits (two converters) except for CS1 which has a 4.2 kV limit (four converters). The last figure shows the total power required for plasma magnetic control.

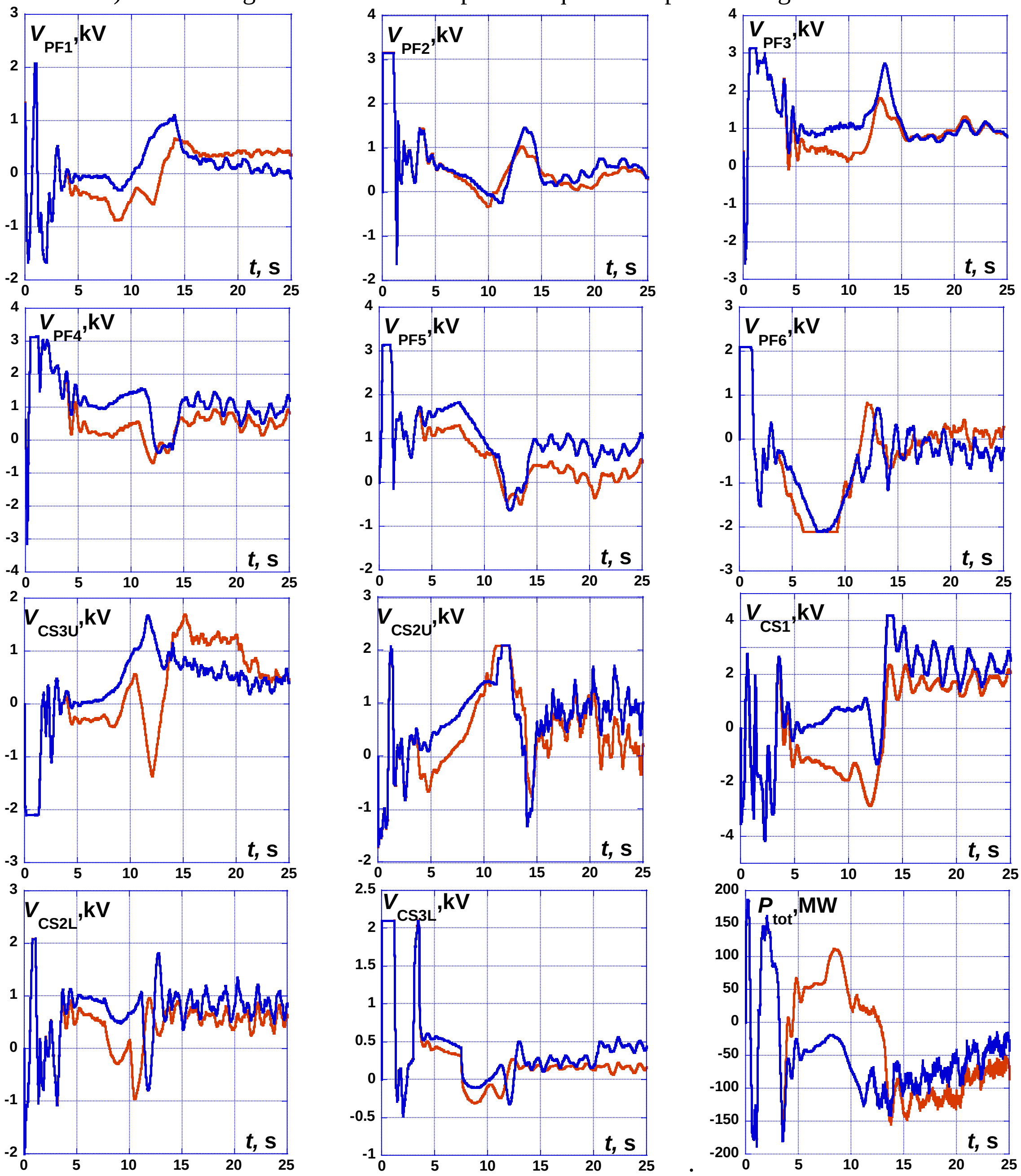

Fig. 6: Voltages produced by the coil converters and the total power required for plasma magnetic control, P_{tot}

4. Conclusion

An alternative I_p ramp-up scheme with a 2 MA, 8 s plateau reduces the limiter heat loads on the ITER central column beryllium first wall panels by about 1 MW compared with that in the standard scenario in which I_p is raised more rapidly to ~3.5 MA before the plasma is diverted. Assuming similar $<n_e>/n_G$, both I_p ramp-up schemes lead to about the same burn duration in 15 MA baseline DT scenarios.

Disclaimer: ITER is a Nuclear Facility INB-174. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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