Plasma magnetic control scenarios for the ITER PFPO-1 phase

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1. INTRODUCTION

The first ITER non-active (H or He fuel) Pre-Fusion Power Operation phase (PFPO-1) will also be the first in which the full blanket first wall and divertor will be installed and in which the ITER Research Plan foresees the achievement of diverted plasma scenarios potentially up to plasma currents of I_p = 10 MA in L-mode and 5 MA in Type I ELMing H-mode. This in turn requires that the Plasma Control System (PCS), in particular for plasma current, position and shape control, be developed with the full required functionality well in advance of the campaign. This PCS PFPO-1 design activity is now underway, with completion expected in 2025. Scenario simulations with specific focus on plasma magnetic control and taking into account the engineering limits of the ITER machine, are a key input to this PCS design.

2. PLASMA MAGNETIC CONTROL

In ITER control of plasma current, position and shape is provided by the central solenoid (CS) coils (5 independent circuits) and poloidal field (PF) coils (6 independent circuits), as shown in Fig. 1. Plasma vertical stabilization (VS) is provided by in-vessel coils (VS3 circuit) or using the PF coils (VS1 circuit) [5]. The magnetic control in a standard I_p ramp-up scenario has the following phases (Fig. 2):

1. I_p < 0.5 MA (plasma initiation): only CS&PF current control, circular plasma ($k \approx 1$);
2. 0.5 MA < I_p < 1.5 MA: feedback control of I_p, Z, R_max, elongation, k ($k_\mathrm{target} = 1$);
3. 1.5 MA < I_p < 3.3 MA: feedback control of I_p, plasma-wall gaps $g5$, $g4$, R_max, k (k_target increases from 1 to 1.6), plasma vertical stabilization starts;
4. 3.3 MA < I_p < 3.5 MA (transition from Limiter controller to Divertor controller);
5. Feedback control of I_p, and six plasma shape parameters $g1$, $g2$, $g4$, $g5$, R_min, R_max.

Magnetic control on ITER is complicated as a result of long settling times caused by the thick vacuum vessel walls, the high inductance of the poloidal field coils and the relatively low voltage limits allowed by the superconducting magnets. In addition, there are many engineering limits related to various machine systems. The scenarios are designed and simulated taking into account engineering limits imposed on:

1. CS and PF coils currents, magnetic fields B_{max} (I, T), forces and combination of forces;
2. Voltages and time constants of the coils power supplies ;
3. Value and oscillations of total power requested from electric grid;
4. Presence of white noise in the VS contour (RMS = 0.2 – 0.6 m/s in [0, 1kHz]);
5. Minimum values of plasma-wall gaps;
6. Minimum distance between inner and outer separatrices in the outboard midplane;
7. Location of the separatrix strike points (upper and lower);
8. Maximum current, voltage and temperature of the VS in-vessel coils.

Plasma density waveforms are optimized to be high enough to reduce the possibility of locked modes, but not to exceed the Greenwald limit.

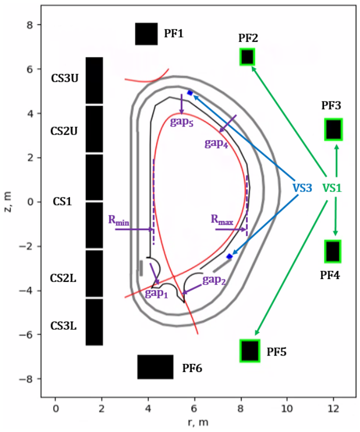

Fig. 1. PF system, vacuum vessel, first wall/divertor, baseline plasma separatrix, feedback controlled plasma geometry parameters and VS stabilisation circuits.

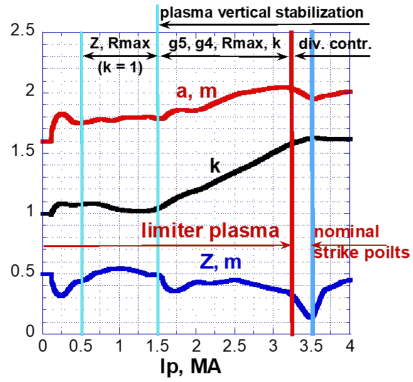

Fig. 2. Plasma feedback control during the standard I_p ramp-up.

The simulations of PFPO-1 scenarios were performed with the DINA code [1-4] comprising:

- 2D free boundary plasma equilibrium solver;
- Circuit equations for the active PF coils and eddy currents in the conducting structures;
- 1D poloidal flux diffusion in the plasma core;
- 1D plasma core transport model (0D at $|I_p| < 280$ kA) includes electron and total ion energy, ions and neutral particles, coronal model for impurity radiation (Be, W).

3. LOW CURRENT HYDROGEN SCENARIOS

The low current scenarios are developed for initial operation to test and commission the PCS, diagnostics and other systems. In these scenarios only half of the CS capacity is used to preserve lifetime. The length of these scenarios is about 100 seconds.

- Scenario 1b: 2MA/2.65T inboard limiter ohmic plasma sweeping vertically with $Z \approx 0 - 2$ m, minor radius $a \approx 1.8$ m, $k \approx 1$ (Fig. 3, feft).
- Scenario 1d: 2MA/2.65T outboard limiter ohmic plasma sweeping vertically with $Z \approx -0.6 - 1.9$ m, $a \approx 1.8$ m, $k \approx 1$ (Fig. 3, centre).
- Scenario 2: 3MA/2.65T inboard limiter ohmic plasma $Z \approx 0.5$, $a \approx 1.9$ m with k increasing during the I_p flattop from 1 to 1.5 – 1.6 (Fig. 3, right). The scenario was simulated with different RMS noise levels in the vertical stabilization contour and with/without the use of in-vessel (VS3) coils. With the VS3 circuit, $k = 1.6$ is the maximum achievable elongation if the RMS noise on $dZ/dt = 0.6$ m/s. With VS1 only, the maximum $k = 1.5$ is limited by loss of stability if RMS of $dZ/dt = 0.6$ m/s. Due to the noise in VS contour, there are large oscillations in power consumed from the electric grid with peaks up to ~ ±50 MW, even if the RMS noise on dZ/dt is reduced to 0.2 m/s.
- Scenario 3: 3.5MA/2.65T divertor plasma with X-point formation at 3MA (Fig. 4). It was found that with high plasma internal inductance $l_i(3) = 1.5$, typical for ohmic mode, the curvature of the separatrix legs differs strongly from the required reference separatrix. This requires that the strike-points be shifted upwards to avoid touching the divertor dome umbrella (Fig. 5). It is also found that a particle transport pinch as well as off-axis electron cyclotron heating (ECH) reduce $l_i(3)$ allowing the reference separatrix to be almost maintained. The scenario in Fig. 5 is simulated for 140 s, but estimations show that in the case of ohmic heating only, the scenario can be ≈ 1000 s long.
  With $P_{ECH} = 5$ MW it can be up to ≈ 4000 s.

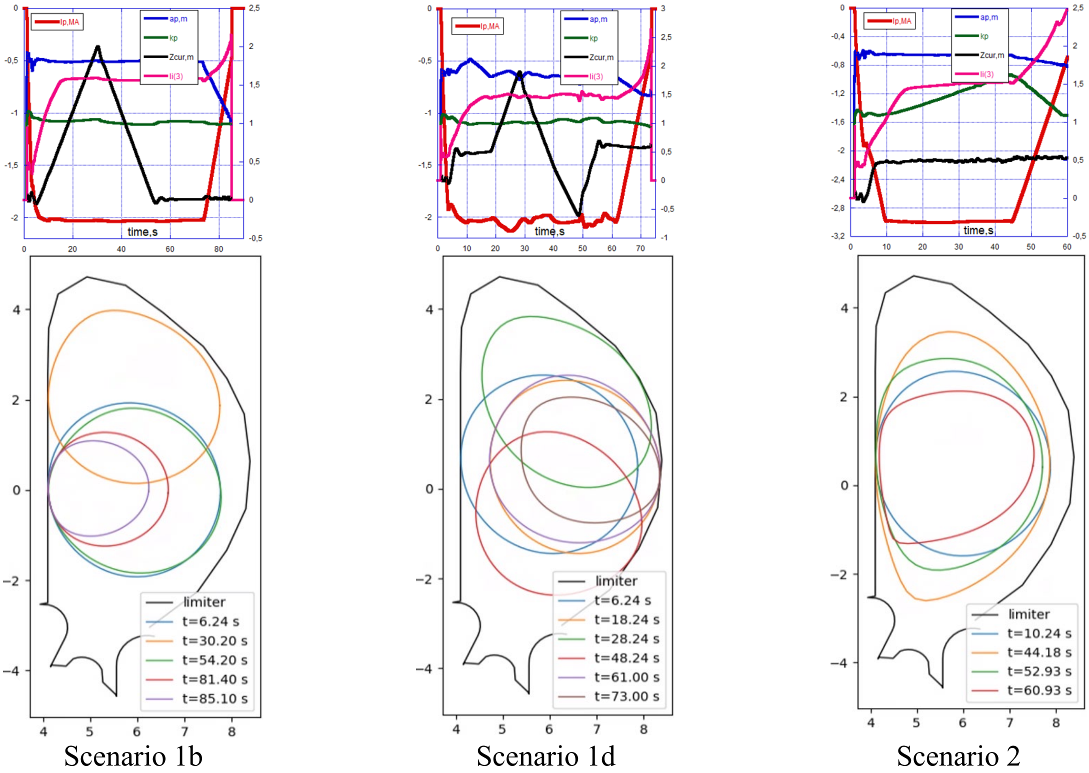

Fig.3. Upper: Waveforms of I_p (red), a (blue), k (green), l_i (pink), Z-coordinate of the current centroid (black).Lower: the plasma boundary outline during the scenario.

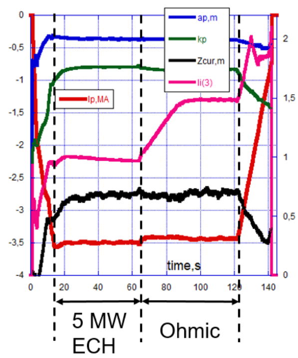

Fig. 4. Waveforms of I_p (red), a (blue), k (green), l_i (pink), Z-coordinate of the current centroid (black).

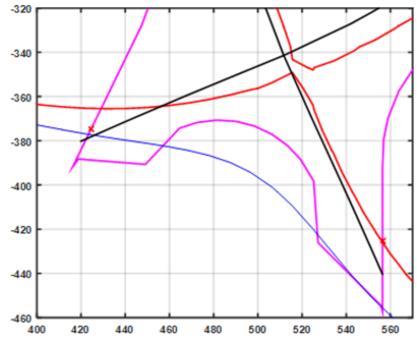

Fig. 5. The divertor area: plasma separatrix at t=121 s, $l_i(3)$ = 1.5 (red); the reference separatrix (black), the divertor wall (pink).

4. 5 MA/1.8T H-MODE HYDROGEN SCENARIO

This scenario is simulated with P_{ECH} = 30 MW for 50 seconds (this duration is a limit of the ECRH system in PFPO-1) and a second segment (80 – 105 s) which continues in ohmic mode. Half of the Central Solenoid capacity is used to preserve lifetime. The results of the simulation are shown in Fig. 6.

5. 10 MA/5.3T OHMIC HYDROGEN SCENARIO

This scenario requires the full CS capacity due to the high poloidal flux consumption. The simulations were performed (Fig. 7) for two levels of impurity content of beryllium (Be) and tungsten (W) (Be assumed only in the limiter phase, W in the diverted plasma phase), resulting in flattop duration varying in the range 70 – 140 seconds.

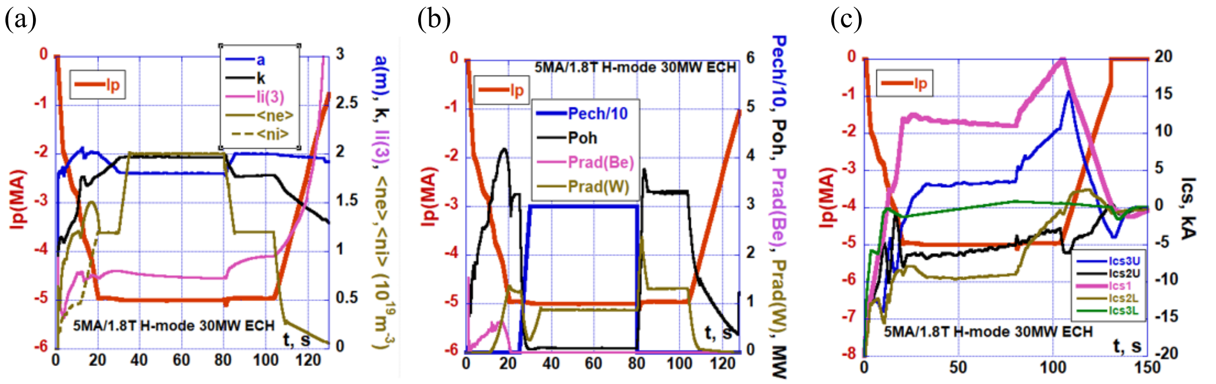

Fig. 6: 5MA/1.8 T hydrogen H-mode scenario. (a) a (blue), k (black), l_i (pink), electron and ion densities (brown). (b) P_{ECH} (blue), ohmic power, P_{oh} (black), Be radiated power (pink), W radiated power (brown). (c) CS module currents. The I_p trace (red) is common to all plots.

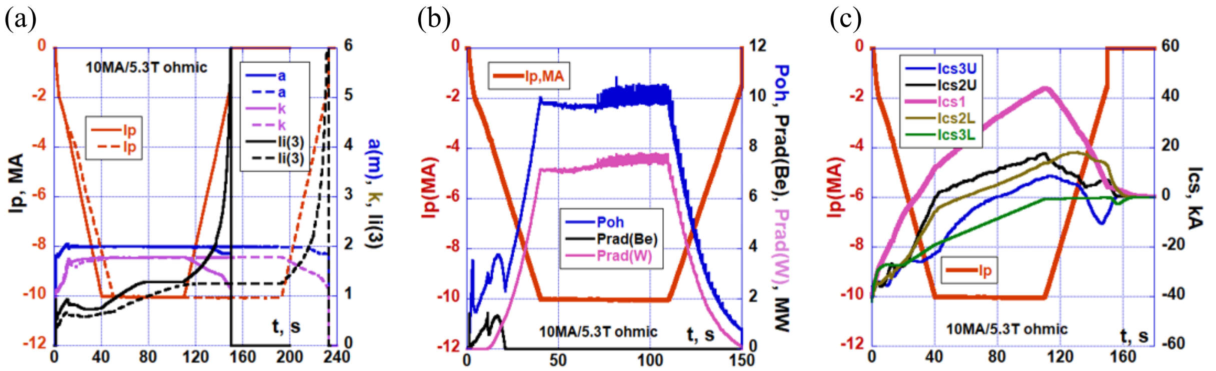

Fig. 7: 10 MA/5.3 T hydrogen ohmic scenario. (a) I_p (red), a (blue), k (pink), l_i (black). Solid lines $\gamma_{Be} = 0.4$, $\gamma_W = 3.0\times10^{-5}$, dotted lines: $\gamma_{Be} = 0.1$, $\gamma_W = 1.0\times10^{-5}$. (b) I_p (red), P_{ECH} (blue), P_{oh} (black), Be radiated power (pink), W radiated power (brown). (c) CS module currents. The I_p trace (red) is common to all plots.

6. CONCLUSIONS

A series of PFPO-1 scenarios with hydrogen plasmas are being designed for use in the design of the ITER Plasma Control System. They include magnetic controllers, satisfying machine engineering limits, but efficiently using the available margins. The foreseen pulse duration is found to be in the range of 70 - 150 s for PFPO-1 with the possibility to significantly increase it for low current scenarios by using more of the CS capacity. For ohmic divertor scenarios, giving high l_i, the divertor strike points must be raised in comparison to the reference separatrix, which may have some impact on density control at the low PFPO-1 input powers. The influence of the impurity content on the pulse duration is assessed for the 10 MA ohmic scenario, giving a factor 2 variation in pulse duration for a factor 3 variation in the assumed W content.

7. REFERENCES

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