ITER-FEAT Scenarios and Plasma Position/Shape Control

Y. Gribov 1), R. Albanese 2), G. Ambrosino 2), M. Ariola 2), R.H. Bulmer 3),
M. Cavinato 1), E. Coccorese 2), H. Fujieda 4), A. Kavin 1), R. Khayrutdinov 5),
K. Lobanov 6), V. Lukash 7), L. Makarova 6), A. Mineev 6), P.L. Mondino 1), A. Pironti 2),
A. Portone 8), E. Rumyantsev 6), I. Senda 4), T. Shoji 4), V. Vasiliev 6)

1) ITER Joint Central Team, Naka Joint Work Site, Naka-machi, Ibaraki-ken, Japan
2) Association EURATOM/ENEA/CREATE, Reggio Calabria, Italy
3) LLNL, Livermore, California, USA
4) JAERI Naka Fusion Research Establishment, Naka-machi, Ibaraki-ken, Japan
5) TRINITI, Troitsk, Moscow Region, Russia
6) Efremov Institute, St. Petersburg, Russia
7) Kurchatov Institute, Moscow, Russia
8) EFDA, CSU, Max Planck Institute for Plasma Physics, Garching, Germany

e-mail contact of main author: gribovy@itergps.naka.jaeri.go.jp

Abstract. The capability of the ITER-FEAT poloidal field system to support the four “design” scenarios and the high current “assessed” scenario have been studied. To operate with highly elongated plasma, the system has segmentation of the central solenoid and a separate fast feedback loop for plasma vertical stabilisation. Within the limits imposed on the coil currents, voltages and power, the poloidal field system provides the required plasma scenario and control capabilities. The separatrix deviation from the required position, in scenarios with minor disruptions is within less than about 100 mm.

1. Introduction

The ITER-FEAT poloidal field (PF) system was designed to support the following main scenarios. Scenario 1: 15 MA inductive scenario with additional heating during the current ramp-up (fusion power 500 MW, Q = 10). Scenario 2: 15 MA inductive scenario without additional heating during the current ramp-up (fusion power 400 MW, Q = 10). Scenario 3: 13.5 MA hybrid scenario (fusion power 400 MW, Q = 5.4). Scenario 4: 10 MA steady-state scenario with weak negative shear (fusion power 370 MW, Q = 5). In addition to the four “design” scenarios, the high current scenario should be assessed to study its possibility and operational flexibility. This is scenario 5: 17 MA inductive scenario with additional heating during the current ramp-up (fusion power 700 MW, Q = 20).

Each scenario has six key states: start of discharge (SOD); the X-point formation (XPF), the start of plasma additional heating (SOH), the start of driven burn (SOB), the end of burn (EOB) and the end of plasma cooling (EOC). Table I shows plasma currents at these key states of the scenarios. Up to XPF the plasma touches the central part of the limiter. The PF system provides the plasma cross-section expansion during the current rise keeping the edge safety factor roughly constant (4.8). At XPF, the transition from a limited to a fully developed diverted configuration is achieved. Further ramp-up of the plasma current continues in the divertor configuration. As an example, Figure 1-A shows the evolution of the plasma aperture during the current ramp-up in scenarios 1 and 2. In scenario 2 SOH is at the start of the current flat-top (SOF). In the other scenarios the SOH is before SOF and SOB is at SOF. At EOB, a controlled termination of the fusion burn is started. The plasma must be cooled prior to the current ramp-down in order to avoid large negative currents at the plasma edge, when a negative surface voltage is applied.

TAB. I: PLASMA CURRENT (MA) AT THE KEY STATES OF THE SCENARIOS.

State	XPF	SOH	SOB	EOB	EOC
Scenario 1	7.5	13	15	15	12.3
Scenario 2	7.5	15	15	15	12.3
Scenario 3	7.5	9.5	13.5	13.5	11
Scenario 4	5	6	10	10	8
Scenario 5	7.5	15	17	17	13.2

TAB. II: PLASMA CURRENT, q, l_i, β_p, β_N DURING BURN, MAGNETIC FLUX AVAILABLE FOR BURN (REFERENCE l_i) AND EXPECTED DURATION OF BURN.

Scenario	Scenario 1	Scenario 2	Scenario 3	Scenario 4	Scenario 5
I_p, MA	15	15	13.5	10	17
q_{95}	3.0	3.0	3.3	4.4	2.7
q_{axis}	1.0	1.0	1.0	2.4	1.0
q_{min}	1.0	1.0	1.0	2.0	1.0
reference l_i	0.85	0.85	0.90	0.67	0.77
maximum l_i	1.0	1.0	1.0	-	0.9
minimum l_i	0.7	0.7	0.7	-	0.7
β_p	0.70	0.65	0.80	1.8	0.7
β_N	1.88	1.75	1.94	3.2	2.1
$\Delta\Psi_{burn}$, Wb	37	30	53	≈10	20
Δt_{burn}, s	500	400	1000	infinite	220

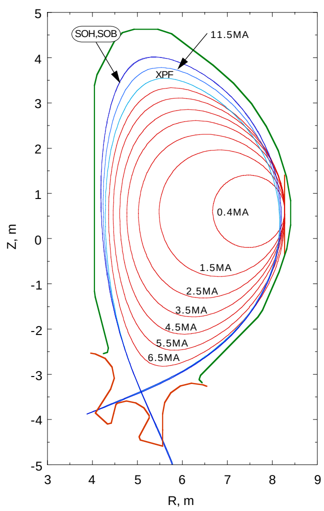

FIG. 1-A. Plasma boundary during the current ramp-up, heating and burn phases of Scenarios 1 & 2.

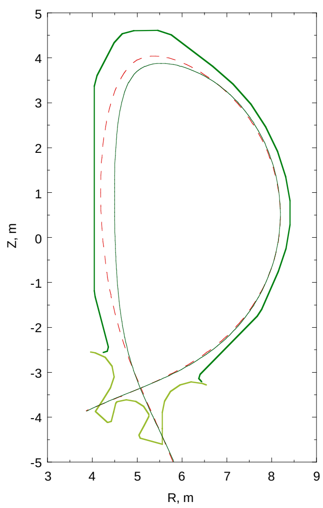

FIG. 1-B. Separatrices during the burn in Scenarios 1, 2, 3, 5 (dashed line) and in Scenario 4 (solid line).

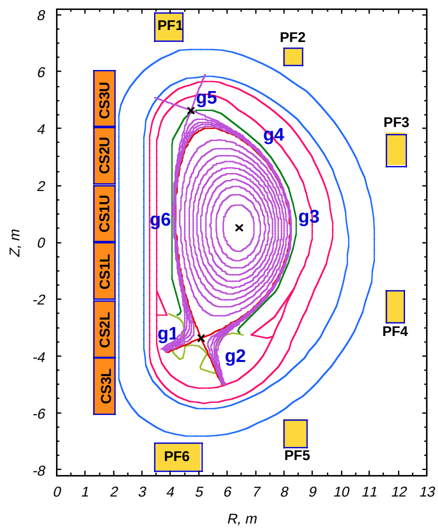

FIG. 1-C. TF, CS and PF coils, vacuum vessel and magnetic surface contours at burn in Scenarios 1, 2, 3, 5.

Plasma in scenarios 1, 2, 3 and 5 is characterised, during the current flat-top, by the following parameters of the separatrix: R = 6.2 m, a = 2 m, κ_{sep} = 1.85 ($\kappa_{95} \approx 1.70$), δ_{sep} = 0.49 ($\delta_{95} \approx$ 0.33). The sensitivity study described in [1] indicated that this plasma elongation should be controllable. Plasma of scenario 4 (at burn) is shifted outwards: R = 6.35 m, a = 1.85 m. This plasma has higher elongation κ_{sep} = 1.95 ($\kappa_{95} \approx 1.83$) and higher triangularity δ_{sep} = 0.56 (δ_{95}

≈ 0.40), than the plasma of scenarios 1, 2, 3 and 5 (see Figure 1-B). Table II shows values of q, l_i, β_p and β_N during the driven burn (the toroidal magnetic field at radius 6.2 m is 5.3 T).

The PF system requires the segmented central solenoid (CS) and six PF coils shown in Figure 1-C, to provide such high shaping capability. The CS consists of six modules. All PF coils and all CS modules, except for the two central modules, have independent power supplies, used for the plasma current, position and shape control. The two central modules of the CS are connected in series in a common power circuit.

2. Poloidal Field Scenarios

All the scenarios mentioned above were at first studied using plasma equilibrium codes with partly prescribed plasma position and shape. The codes use also prescribed values of l_i and β_p. Eddy currents induced in the conducting structures are not taken into account. The deviation of the separatrix from the reference separatrix, shown by the dashed line in Figure 1-B, was minimised within the following limits: less than ± 60 mm in the divertor region, less than ± 30 mm near the limiter. The distance between the inner and outer separatrices at the equatorial plane on the outboard side was kept greater than or equal to 40 mm, for reliable operation in a single X-point mode. Assumptions on the resistive losses of the poloidal magnetic flux are given in Table III.

TAB. III: ASSUMPTIONS ON RESISTIVE CONSUMPTION OF POLOIDAL MAGNETIC FLUX.

Parameters	Value
Flux loss at breakdown	10 Wb
Flux loss during the plasma current ramp-up until SOH	$0.45\mu_0\Delta(R_pI_p)$
Flux loss from SOH to SOB for inductive scenarios	10 Wb
Flux loss from SOH to SOB for hybrid and non-inductive scenarios	17 Wb
Flux loss during the plasma cooling (till EOC)	10 Wb

Two types of studies have been performed with these codes;

i) Study of the operational range of l_i to obtain the magnetic flux available for burn, and maximum values of the (scenario) currents in the CS and PF coils. Three values of l_i, shown in Table II, were considered for scenarios 1, 2, 3 and 5 at SOH, SOB and EOB. For scenario 4 only one magnetic configuration was analysed: a state during burn with l_i = 0.67. The magnetic flux available in these scenarios for burn (reference l_i) is given in Table II. The table also shows the expected duration of the burns. The maximum value of the total flux swing of the PF system is about 283 Wb (17 MA scenario with reference l_i). Maximum values of the coil currents and magnetic fields, arising at SOD and in the l_i-survey at SOH, SOB, EOB states, are within the coil capabilities with some margins required for plasma control actions.

ii) Design of the PF scenarios in terms of plasma equilibrium snapshots. The goal is to provide the time evolution of the plasma parameters (e.g. major and minor radii, elongation, triangularity) and waveforms of the coil currents, voltages, and power.

Both types of studies have demonstrated the capability of the PF system to support scenarios 1 - 5. The PF scenario 2 was verified and simulated with more self-consistent codes using the following two steps;

a) The first step was done with the MAXFEA code (free boundary plasma equilibrium). This code takes into account eddy currents in the vacuum vessel and a model of the power supplies (in particular resistors in the CS modules and coils PF1, PF6, used at initial phases of the current ramp-up). Moreover, the code simulates feedforward and feedback control of the plasma current, position and shape, according to the control scheme described in section 3. However the evolution of l_i, β_p and the resistive flux loss need to be prescribed for the code. Waveforms of the plasma parameters and coil currents obtained with EQUCIR were used as input data in MAXFEA simulations.

b) The second step was done with the DINA code. In addition to the MAXFEA performance, DINA calculates self-consistently the evolution of the plasma temperature and current profiles (i.e. l_i, β_p and the resistive flux loss). Evolutions of the plasma density, Z_{eff} and additional power are prescribed in the code. In particular, the linear rise of the plasma density during the current ramp-up was adjusted to come to the SOH state with the values of l_i and Ejima coefficient C_{Ejima} close to 0.85 and 0.45 respectively. Waveforms of the coil feedforward voltages and reference values of the controlled gaps between the separatrix and the first wall, used in the DINA simulations, were obtained with MAXFEA. The active power demand was less than 100 MW. The simulations confirmed the assumptions using in the plasma equilibrium snapshots studies (e.g. the range of l_i and C_{Ejima} at SOH) and demonstrated the capability of the PF system to support scenario 2.

Studies of plasma initiation and verification of the assumptions used in the studies mentioned above (i.e. the magnetic flux at breakdown) were performed by simulations, which take into account eddy currents in the conducting structures, plasma displacements, models of the power supplies, plasma ionisation and transport. The studies showed the capability of the PF system to provide outboard plasma initiation, assuming 2 MW of the EC power deposited to electrons.

3. Plasma Current, Position and Shape Control

The ITER-FEAT highly elongated plasma is unstable relative to its vertical displacements. The steel vacuum vessel, shown in Figure 1-C, provides passive stabilisation of the plasma vertical displacements. Important elements of the plasma passive stabilisation are toroidally continuous rings, attached to the blanket module triangular supports (shown in Figure 1-C above the divertor region). These stabilising rings improve the up/down symmetry of the plasma-facing conducting structures, which reduce, by about a factor of 2, the initial value of the plasma vertical displacement after a plasma disturbance. Conducting elements of the blanket modules, having an eddy current decay time of about 2 ms, have a negligible effect on plasma passive stabilisation. The plasma vertical instability growth time τ_g and stability margin m in scenarios 1, 2, 3 and 5 achieve minimum values at SOH and $l_i = 1.0$ ($\tau_g \approx 80$ ms, $m \approx 0.4$). During the burn with the reference l_i: $\tau_g \approx 150$ ms, $m \approx 0.65$ in scenarios 1, 2, 3, 5 and $\tau_g \approx 180$ ms, $m \approx 0.82$ in scenario 4.

To reduce the apparent power required for the plasma vertical stabilisation the Vertical Stabilisation (VS) scheme shown in Figure 2 is applied. The scheme uses the fact that in the scenarios considered the imbalance current of coils PF2 – PF5, flowing through the VS converter, is significantly lower than the currents flowing through the main converters of these coils. The scenario value of the imbalance current is limited by 12 kA. The maximum current in the VS converter is 22.5 kA, the on-load voltage limit is 6 kV. The VS feedback algorithm determines the voltage of the VS converter, using as input the vertical velocity of plasma current centre. The other feedback loops control the plasma current and shape.

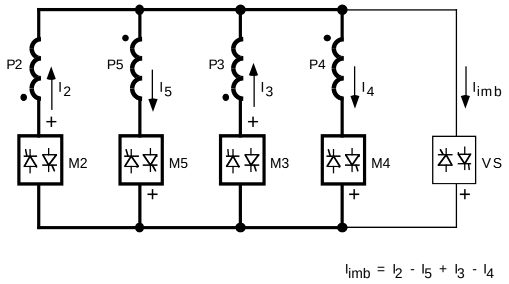

FIG. 2. The vertical stabilisation circuit.

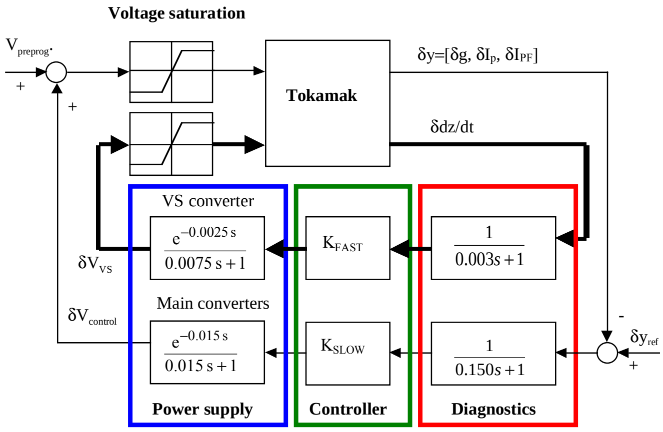

FIG. 3. Two loops control scheme adopted.

The plasma shape control in the divertor configurations is realised with the control of the six gaps between the separatrix and the plasma-facing components (the first wall and divertor). The locations of the gaps are shown in Figure 1-C. The time scale of plasma current and shape control is about a factor of 10 longer than that for plasma vertical stabilisation (0.5 – 1 s). This justifies the "separation" of the corresponding feedback loops which are shown in Figure 3. Here the long-time-scale loop provides simultaneous control of the plasma shape and current, changing the voltage on the coil main converters. The maximum current in the main converters is 45 kA, the on-load voltage limit is 1.5 kV.

Several control algorithms were proposed. They were tested with the linear models of ITER plasma in simulations with minor disruption. The minor disruption in scenarios 1, 2, 3 and 5 was defined as an instantaneous l_i drop of 0.2 (l_{i0} – 0.5), without recovery, simultaneous with β_p drop of 0.2 β_{p0}, followed by a 3 s exponential recovery. The performance of the controller "JCT/April 2000" in scenario 2 was also validated with non-linear plasma models: MAXFEA, PET (the code similar to the MAXFEA with calculation of the plasma current diffusion) and DINA. The simulations showed deviations of the gaps from the reference values less than about 100 mm and the active power demand for control less than about 100 MW.

4. Conclusion

These studies demonstrate the capability of the ITER PF system to support the main design scenarios 1 – 5, including the 17 MA inductive scenario and the 10 MA steady-state scenario with weak negative shear.

Disclaimer:

Reference:

[1] MONDINO, P.L., et al., "Studies to increase the elongation in ITER: impact on passive and active plasma vertical position stabilisation systems", (Proc. Conf. 21st SOFT 2000, Co-01).