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Plasma Ramp-up and Sustainment Scenarios for Tokamak Energy's Fusion Pilot Plant

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Abstract. Tokamak Energy is designing a Fusion Pilot Plant under the US Department of Energy's Milestone-Based Fusion Development Program. Tokamak Energy's Fusion Pilot Plant is based on low aspect ratio tokamak with high-temperature superconducting coils, exploiting the potential to access high confinement, high beta, and high bootstrap current fraction. Representative parameters used in the present modelling are: major radius $R_0 = 4.25$ m, aspect ratio A = 2.15, toroidal field $B_t = 4.0$ T, and plasma current $I_p = 13.6$ MA. Since the flux swing capability of the central solenoid (CS) is not sufficient to reach full I_p, a novel I_p ramp-up and sustainment scenario utilising RF power (EC and IC), bootstrap current, and induction by the CS and other poloidal field coils was developed. The plasma is initiated with EC power injected to the "trapped particle configuration". Following initial ECCD overdrive with CS assist at low density, plasma is densified and ion heating by IC is applied to initiate fusion burn. The increased plasma stored energy drives the bootstrap current, and additional ramp-up to full I_p is achieved with inductive assist from the increasing vertical field. Optimisation of the CS usage during I_p ramp-up is essential to achieve quick I_p ramp-up with minimal use of the CS flux.

1 Introduction

Tokamak Energy (TE) is designing a Fusion Pilot Plant (FPP) with support from the US Department of Energy's Milestone-Based Fusion Development Program [1]. TE's FPP [2] is based on low aspect ratio tokamak for integrated test and validations of technologies, systems and processes required for commercial fusion energy deployment. The FPP, which is targeting start of operations by 2035, will consist of an operationally-relevant fusion environment. By exploiting the inherent plasma physics benefits of the low aspect ratio tokamak, the FPP will demonstrate scalable net power in a fully-integrated system.

The low aspect ratio tokamak geometry enabled by Rare-Earth Barium Copper Oxide (REBCO) High Temperature Superconducting (HTS) magnets offers some key inherent advantages over alternative approaches for fusion energy. In particular, the potential to access high confinement, high beta, and high bootstrap current fractions in a stable plasma configuration is an attractive proposition for economical power plants. Representative parameters used in the modelling are: geometric major radius R_0 = 4.25 m, aspect ratio A = 2.15, toroidal field B_t = 4.0 T (at R_0), and plasma current I_p = 13.6 MA.

Because of the low aspect ratio, the flux swing capability of the central solenoid (CS) is not sufficient for I_p ramp-up to the full operating condition. Therefore, creative I_p ramp-up scenarios must be developed. In this paper, novel I_p ramp-up and sustainment scenarios utilising RF power (EC and IC), bootstrap current, and induction by the CS and other poloidal field (PF) coils are presented. The main tool used for time-dependent scenario modelling is METIS [3].

The design workflow and tools used for scenario development are described in Section 2. The I_p ramp-up and sustainment scenarios proposed for the FPP are described in Section 3. Conclusions are given and further analyses are discussed in Section 4.

2 Design workflow and analysis tools

The global race to commercial fusion and decadal timescale motivates pragmatic design choices and the need to balance risk across physics, engineering, R&D, schedule, cost, etc. Our design approach is focused on system integration and phased fidelity [4]. The design workflow is described in [5].

Initial assessment of potentially attractive design space is identified using the 0-D whole plant systems code pyTok developed by TE. It includes simplified models for all major plant systems, parametric CAD generation for cost modelling and neutronics, and free-boundary equilibrium generation using FreeGS [6]. It is used for large parameter space optimisation and sensitivity studies, with continuous validation against higher-fidelity codes.

Plasma and technology assumptions used for FPP design are listed in Table 1. Examples of pyTok output (inboard radial build and poloidal cross section) are

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shown in Fig. 1, and representative parameters are shown in Table 2 for different values of the H factor (radiation corrected, relative to ITER98 scaling [7]). It can be seen that a small (~ 10%) reduction in confinement makes a substantial difference in the required external non-inductive current drive power, and on the flat-top duration (determined by the remaining CS flux). This is because of the combination of increased I_p and reduced bootstrap current fraction f_{bs} which results in lower non-inductive current fraction f_{NI} at lower H factor. It is presently assumed that ECCD is used for external non-inductive current drive during the I_p flat-top phase. Optimisation of ECCD is reported in detail in [5]. OCC is the Overnight Capital Cost which is commonly used to compare different reactor designs. It includes equipment and materials, labour and engineering, site preparation and installation, contingency and indirect costs, but does not include financing costs, inflation, or operating and maintenance costs.

Table 1. Assumptions used for FPP design

Plasma stability	$\beta_N \leq \beta_{N,\text{no-wall}}$, $f_{GW} \leq 1.0$, $\kappa \leq 0.9 \times$ controllable limit, $q_{min} \geq 2.2$
Plasma scenario	Fully non-inductive operation and pulsed operation considered
Inductive current drive	$\geq$ 50% of inductive flux required for I_p ramp-up
Plasma confinement	$H^*_{ITER98} = 1.2 - 1.6$ (radiation corrected)
Magnets	Full HTS magnet set (TF + PF + CS). Continuously wound TF coils (no remountable joints), PFs outside TF.
TF coil lifetime	$\geq$ 5 full power years
Breeding blanket	Natural, slow flowing, liquid Li breeder with He coolant
Tritium breeding ratio	$\geq$ 1.1
Plasma facing components	Solid as baseline with liquid Li option also being developed
Heating and current drive	ECCD for flat-top and ramp-up. ICRH also being considered for ramp-up
Centre column shielding	Graded WC and B_4C with He coolant (W_2B_5 if becomes available)

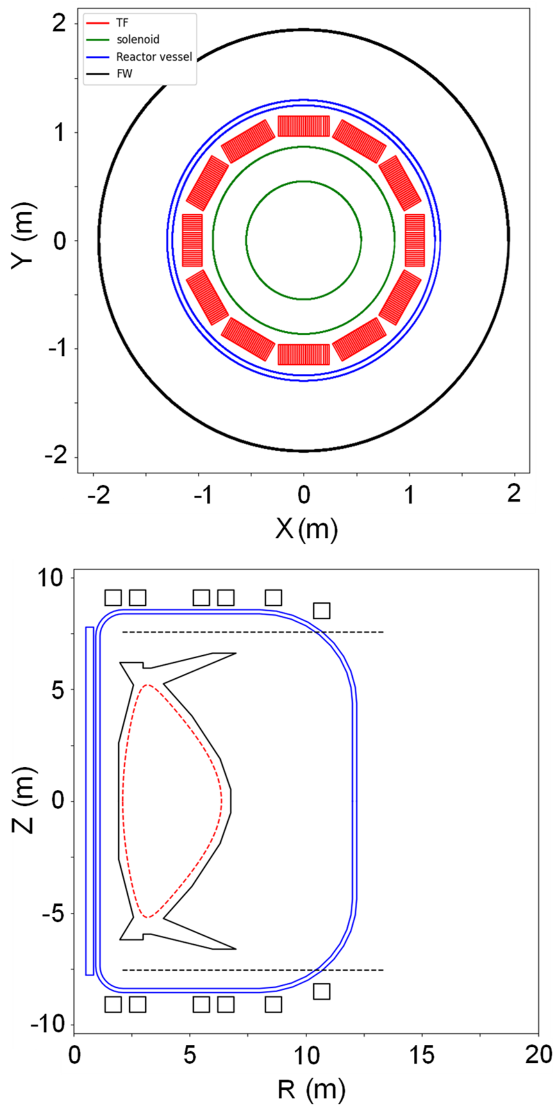

Fig. 1. Example of pyTok output: (top) radial build of the centre column showing the shield (space between the first wall and reactor vessel), TF coils, and CS; (bottom) poloidal cross section showing the plasma (red dashed), first wall (black), TF/CS coils (blue), and PF coils (black squares) for the case shown in Table 2.

Table 2. Parameters for different H factors (radiation corrected)

R_0 [m]	4.25			
$A = R_0/a$	2.0			
B_t [T]	4.5			
δ/κ	0.5/2.45			
Φ_{CS} [Vs]	45 (bipolar)			
H^*_{ITER98}	**1.2**	**1.3**	**1.4**	**1.5**
I_p [MA]	**15.9**	**14.9**	**14.2**	**13.6**
P_{fus} [MW]	930	891	842	804
$P_{elec,net}$ [MWe]	**68**	**90**	**98**	**106**
$P_{H\&CD\text{-}EC}$ [MW]	141	119	103	90
$P_{rad,core}$ [MW]	120	195	180	165
Q	6.6	7.5	8.2	9
β_N	2.81	2.93	2.93	3.09
f_{bs}	0.58	0.64	0.69	0.74
P_{sep}/R [MW/m]	21.9	22.4	19.7	18.4
OCC [$Bn]	5.75	5.54	5.38	5.24
f_{NI}	**0.8**	**0.85**	**0.9**	**0.95**
$t_{flat\text{-}top}$ [hr:mn]	**00:25**	**00:40**	**01:05**	**02:25**

Once the initial operating point is identified, ASTRA-SPIDER 1.5D transport code [8] is used as an assumption integrator to develop consistent flat-top plasma operating points by combining several low fidelity models. The 0D pyTok output is turned into 1D profiles and 2D equilibrium, which can be used for further analysis using higher fidelity codes. The temperature and density profiles assumed in pyTok analysis (shown in Fig. 2) are taken from ASTRA analysis.

3 I_p ramp-up and sustainment

In this paper, METIS 1.5D transport code [3] was used for investigation of time-dependent scenario to confirm that the operating point identified in the previous section is indeed reachable with the available CS flux and RF power. METIS combines 0D scaling-law normalised energy and particle transport with 1D current diffusion model and 2D equilibria to realize fast full tokamak plasma analyses and predictions.

METIS was used in the predictive mode, with the Bohm-gyroBohm model for thermal diffusivities χ_e and χ_i [9], with their magnitudes adjusted to reproduce the plasma stored energy corresponding to the specified confinement. The H factor relative to the ITER98y2 scaling, usually chosen in the range 1.2 to 1.4, is used. The relative magnitudes of χ_e and χ_i can be specified, but for the present work $\chi_e = \chi_i$ was used. Electron cyclotron (EC) heating and current drive profiles are specified by deposition location and width, with the EC driven current expressed using a simple ECCD efficiency scaling η_{eccd} [3]. The selectable multiplier for the ECCD efficiency is taken to be unity. The EC power level of 100 MW anticipated for FPP is much lower than the threshold for quasilinear effects to become important [10] (the threshold power density for FPP condition is 50 MW/m^3, more than an order of magnitude higher than that expected in FPP). Ion cyclotron (IC) heating and power partition between ions and electrons are calculated based on PION code results. IC heating

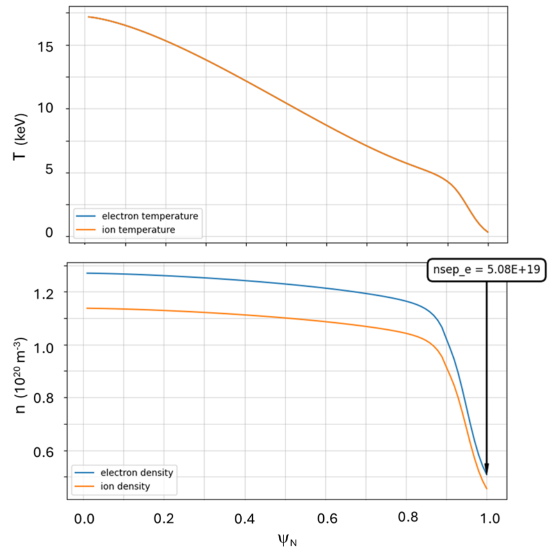

Fig. 2. Assumed profiles of (top) temperature ($T_e = T_i$ assumed) and (bottom) density used in pyTok for the case $H^*_{ITER98} = 1.4$. The horizontal axis is the normalised poloidal flux.These profiles are informed by ASTRA.

scenario is chosen to be ^{3}He minority in DT plasma, and it is assumed that IC is dedicated for ion heating (not for current drive). Particle transport is not solved, and the density profile is specified using a pedestal model and a peaking factor.

Fixed boundary equilibria represented by shape parameters were used in the METIS simulation reported here. Free boundary equilibria consistent with PF coil currents are being investigated using the tokamak simulator SOPHIA [11] coupled with ASTRA-SPIDER, in the next step of scenario development.

Emphasis was placed on I_p ramp-up combining different methods of current drive, since the inductive current drive by the CS is not sufficient to reach full I_p (because of the limited flux swing capability due to the low aspect ratio). Components of I_p other than the inductive current driven by the CS include: RF-driven current, bootstrap current (including the α-driven component), and inductive current driven by other PF coils.

3.1 Plasma start-up

Since inductive current drive is efficient even at high density, it is advantageous to minimize the use of the CS during plasma start-up and early phase of I_p ramp-up. It is possible to form a tokamak plasma with closed flux surfaces by injecting RF power to a magnetic configuration with a weak vertical field (B_v) with positive decay index (the so-called "trapped particle configuration") [12]. This method of tokamak plasma formation is used routinely in Japanese spherical tokamaks including LATE, TST-2, and QUEST. In QUEST, I_p was ramped up to over 50 kA and sustained stably for 0.9 s with 270 kW of 28 GHz second harmonic EC wave, without the use of the CS [13]. We propose to use this method of plasma start-up.

3.2 Early I_p ramp-up phase

It is well known that RF-only "non-inductive" I_p ramp-up takes a long time. This is because RF overdrive (i.e., $I_{RF} > I_p$, where I_{RF} is the RF driven current) induces a negative toroidal electric field which drives the negative inductive current to cancel the overdriven current density (j_{RF} - j_p). The current density (and the total current) can change only over the time scale of this negative inductive current to dissipate. In the regime of interest for efficient current drive with low density and high electron temperature, this time scale can be very long. For example, the typical ramp-up time to full I_p is of order 30 minutes for STEP [14].

The I_p ramp-up time can be shortened significantly by applying a positive inductive field using the CS. A typical example of ramp-up and sustainment scenario for FPP is shown in Fig. 3, and the poloidal flux balance for this case is shown in Fig. 4. Ramp-up to full I_p in 120 s is prescribed, determined by the permissible rate of current change in the HTS coils. EC power is ramped up from 2 MW to 60 MW in the first 80 s. EC driven current, bootstrap current (including the α driven component), and inductive current driven by the PF coils are calculated, and the flux that the CS must contribute is derived. It can be seen that the CS is contributing greatly to the quick I_p ramp-up during this phase of I_p ramp-up (the blue line almost coinciding with the magenta line in Fig. 4 until t ≅ 25s).

3.3 Later I_p ramp-up phase and I_p flat-top phase

During the later phase of I_p ramp-up, we propose to make use of ion heating to initiate the fusion burn. This is accomplished by densification (the line-averaged electron density increases from 0.2×10^{20} m^{-3} at 60 s to 1.1×10^{20} m^{-3} at 120 s) and injection of 40 MW of IC power (the total EC + IC power is 100 MW during this phase). This will accelerate entry to the fusion burn phase by providing direct ion heating as well as enhanced fusion reactivity by energetic ions accelerated by the IC fast wave. The fusion reaction will provide additional heating from the fusion-produced α particles, increasing the plasma stored energy and β_p. In addition to increasing the bootstrap current, B_v has to increase to maintain equilibrium, which provides additional inductive drive [15].

From Fig. 4 it is clear that during this phase, the flux contribution from the outboard PF coils (B_v coils), i.e., the difference between the red line and the blue line, is dominant. In fact, the CS can be partially recharged (where the CS flux has a negative slope) by I_p overdrive by the combination of EC-driven current, bootstrap current, and inductive drive from B_v ramp-up, and the recharged flux can be used to extend the I_p flat-top. This scenario is motivated by the JT-60U experiment which showed that access to the advanced tokamak regime with high confinement, high stability, and high bootstrap current is possible without the use of the CS [16]. Table 3 summarises the correspondence between the JT-60U experiment and the FPP scenario.

During the I_p flat-top phase, EC power is increased to 100 MW to maintain the EC driven current of about 2 MA. It is noted that the current density profile does not reach steady state over the time scale shown. It takes about 2000 s for the plasma current profile to fully relax.

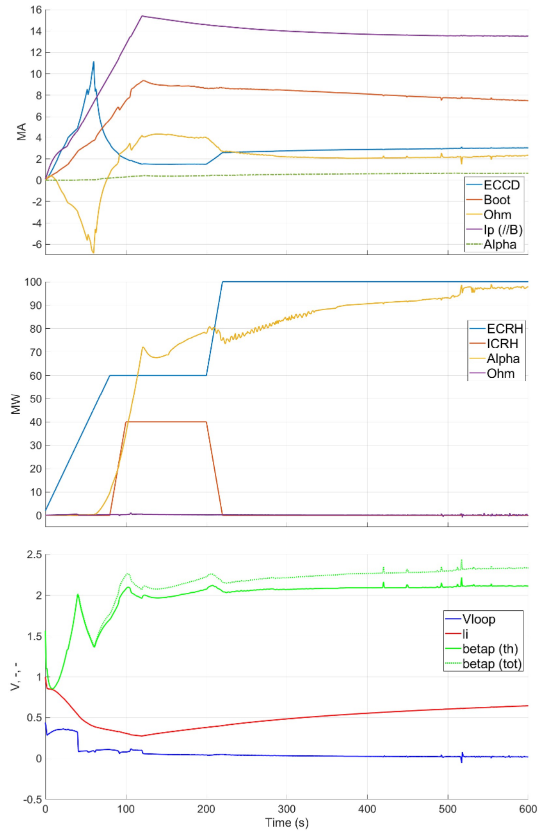

Fig. 3. I_p ramp-up scenario calculated by METIS: (top) I_p (field-aligned component) and its constituents (EC-driven, bootstrap, inductive); (middle) powers (EC, IC, α); (bottom) β_p, l_i, V_{loop}. Line-averaged electron density (not shown) is kept at 2×10^{19} m^{-3} until 60 s, and ramps up to 1.1×10^{20} m^{-3} by 120 s.

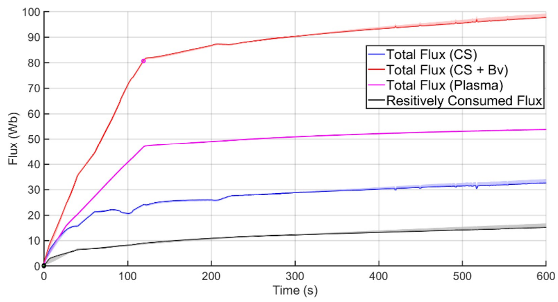

Fig. 4. Poloidal flux balance for the case shown in Fig. 3.

Table 3. Correspondence between JT-60U experiment and FPP scenario.

	JT-60U	FPP
start-up	Bv swing	EC start-up
early ramp-up	LHCD	ECCD + CS
later ramp-up	densify + NB + Bv ramp-up	densify + IC + α + Bv ramp-up

4 Conclusions and further analyses

Tokamak Energy is designing a Fusion Pilot Plant based on low A tokamak, using HTS coils. This approach offers a potential to access and sustain high confinement, high beta, and high bootstrap current fraction. Because of low A, the flux swing capability of the CS is not sufficient for full I_p ramp-up, so novel I_p ramp-up and sustainment scenarios utilising RF power (EC and IC), bootstrap current, and induction by the CS and other PF coils needed to be developed. Based on METIS analyses, the following ramp-up scenario was developed: (1) The plasma is initiated with EC power injected to the "trapped particle configuration". (2) During the early phase of I_p ramp-up, ECCD at low density is used with inductive assist by the CS. (3) During the later phase of I_p ramp-up, the plasma is densified and IC ion heating is applied to initiate fusion burn. The increased plasma stored energy (and β_p) increases the bootstrap current, and more importantly, additional I_p ramp-up is achieved with inductive drive from the increasing B_v. Partial recharging of the CS is possible during this phase, which is beneficial for extending the I_p flat-top duration. Optimisation of the CS usage is essential to achieve a quick I_p ramp-up with minimal use of the CS flux.

In the METIS analyses reported here, Bohm-gyroBohm diffusivities were used to calculate T_e and T_i profiles, for a specified density profile, and the stored energy was adjusted to match the prescribed H factor (e.g., ITER98). But in reality, T_e and T_i profile evolution for the calculated heating profiles (P_{el} and P_{ion}) depends on what happens to plasma turbulence (and therefore energy transport). The addition of IC was motivated by experimental evidence that electron heating enhances turbulence whereas ion heating (and associated fast ion generation) might suppress turbulence [17]. Addressing this question properly requires a more sophisticated modelling. The specification of the edge pedestal also has a significant effect on the total stored energy and fusion power. Additionally, compatibility with heat and particle exhaust must be considered. These are subjects of future study.

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