The FTU-D Project

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Abstract. A modification of the FTU tokamak (toroidal field $B_T = 8T$, plasma current $I_P = 1.6MA$, minor radius a=0.3m, major radius R=0.93m) is proposed in order to extend the FTU operation to strongly shaped plasmas (FTU-D: R=1m, a=0.18-0.2m, elongation =1.6, triangularity   up to =0.8). FTU has a circular vacuum vessel and was built to produce circular plasmas, however unbalancing the currents in the windings of the air core transformer a plasma shaping can be produced. Single Null (SN) and Double Null (DN) equilibria have been studied with a maximum current in the range 0.350-0.450 MA. The scientific aim of the project is the investigation of the advanced tokamak operation, characterised by the simultaneous achievement of high $_N$ (normalised beta) and high bootstrap current fraction (f_B) in regimes with high-energy confinement obtained by current and pressure profile control. The main features of FTU-D, with respect to other existing tokamaks, are the high magnetic field ($B_T = 5-2.5T$), the high density and aspect ratio value (A = R/a = 5-6) and the possibility of investigating regimes with dominant electron heating.

1.Introduction

A modification of the FTU tokamak (toroidal field $B_T = 8T$, plasma current $I_P = 1.6MA$, minor radius a=0.3m, major radius R=0.93m) is proposed in order to extend the FTU operation to strongly shaped plasmas (FTU-D: R=1m, a=0.18-0.2m, elongation =1.6, triangularity   up to =0.8). In the physics section, the scientific objectives of FTU-D are addessed**:** advanced scenarios [1] of operation on FTU**-**D are analyzed and discussed. FTU**-**D can provide a significant contribution in this field since: it can study the access to enhanced confinement regimes and advanced tokamak scenarios at high magnetic field, high density, high aspect ratio and in the presence of dominant electron heating, like that provided by -particles. Due to the high aspect ratio value, FTU-D can study plasmas with large bootstrap current fraction (60%) at moderate   N values ($_N$~2) and can investigate the neoclassical MHD stability, also experiencing active control techniques. In the third section strongly shaped plasma equilibria, both in Double Null (DN) and Single Null (SN)**,** are presented. In the last section the main engeneerieng upgrade, needed to produce shch equilibria are summarized.

2.The physics

The FTU-D magnetic field values are mainly constrained to 5T and 2.5T by the use of the 140GHz, 1.6MWX0.5sec Electron Cyclotron Heating system (ECRH)[2]. Besides

ECRH, the 8GHz Lower Hybrid (LH) system[3] will be used to control the current density profile. The LH system consists of 2 antennas. In a recent FTU campain[4], one antenna coupled routinely 1.2.MW to the FTU plasma. A second antenna will be mounted in the near future, thus the total LH power capability will be 2.4MW. Accordingly, the additional power, considered in FTU-D, is 4MW (1.6MW of ECRH+2.4MW of LH). The power threshold to enter the H mode, calculated on the basis of the ITER scaling low[5], is 1.-1.3 MW at B=2.5 T and $n=1x10^{20}m^{-3}$ and 3.4 MW at B=5T and $n=1.5 X10^{20} m^{-3}$. To enter a good quality H-mode at high density and magnetic field, a power as high as 5MW is needed. Therefore, to further increase the power level, the use of the 433MHz system [6] is also envisaged to perform Fast Wave heating and Current drive, as well as an upgrade of the ECRH power capability up to 2.6 MW.

In order to determine the FTU-D high- capability the energy confinement time was calculated on the basis of the ITER89-P scaling law [7]. In performing the analysis reported here, we make the following assumptions: 1) the confinement time is given by H^{89} times the ITER89-P scaling. (the enhancement factor can be associated either with the achievement of the H-mode or by the effect of internal transport barrier); 2) equal electron and ion temperatures are assumed (Te =Ti) 3) Zeff =1 is assumed in the calculation. The upper density value of FTU-D operations is set by the cutoff of the O-mode- $2.4 10^{20} m^{-3}$- at B=5T and by the cutoff of the X-mode -$1.2 10^{20}m^{-3}$ - at B=2.5. The Greenwald limit is beyond this values, namely $n_G=2.2-4.4x10^{20} m^{-3}$ at Ip =0.250-0.500MA. The results of this analysis are shown in Table I, where, besides the geometric data of the considered equilibrium, the main plasma parameters are reported (n_{eL} is the line averaged electron density). According to these results, high bootstrap fraction (at least 60%) is obtained in all the considered scenarios even within the conservative hypothesis on confinement, $H_{89}=2$. In high magnetic field operations (5T), $_N=2, 3$ can be achieved, in high-density plasmas, with $H^{89}=2, 3$, respectively. At low magnetic (2.5T), the target value $_N=3.5$ (corresponding to 90% of bootstrap current fraction at q=3 and A=6) is achieved already at $H_{89}=2$, whereas at H=3, a region ($_N$ ~5) potentially achievable with wall stabilisation, can be explored.

A transport simulation of the the reference equilibrium at B=5T (column 1 of tab.1) was performed by means of the JETTO code, assuming as transport model a shear dependent mixed Bohm-gyro-Bhom (BgB)[8] model calibrated on JET discharges and verified on results belonging to the ITER-database. A negative central shear discharge was simulated in the presence of 3MW of LH power and 1MW of ECRH, both deposited off axis. Under the conservative hypothesis that a complete BgB model works both for ions and electrons, $H_{89}=2$ is obtained, while using neoclassical transport for the ions a factor H=2.2 is obtained: the improvement in the confinement is mainly due to an extended region of vanishing magnetic shear. Fig.1 shows the temporal behaviour of several quantities calculated in the simulation. The bootstrap fraction is 57%, while the residual current is driven by LHCD. Fig.2 shows the current density profiles with the different contributions, bootstrap current and LHCD current.

The linear stability of the ideal n=1 external kink modes, in presence of a perfectly conducting wall, is investigated for the equilibria shown in table 1. According to this analysis, the reported values are not expected to be limited by ideal modes, because of the closeness of the wall to the plasma boundary. It must be noted that no profile optimization was performed in this MHD analysis. In one case, a scan in $_N$ was

performed of the n=1 external mode stability, in the absence of a conducting wall. Stability was found up to β_N=2.4, thus showing the stability of the Resistive Wall Mode up to β_N =2.4.

	DN A=5.73 a= .185 m		DN A=6.1 a= .175 m		SN A=5.8 a=.182m	SN A=6.3 a=.171m
I_P (KA)	351		264		361	300
B_0 (T)	5		2.5		5	2.5
k	1.56		1.65		1.45	1.65
q*	4		2.5		3.45	2.1
H_{89}-factor	2	3	2	3	2	2
τ_E(ms)	32	49	21	31.5	32	23.5
$n_{eL}(10^{20}m^{-3})$	1.4		0.91		1.5	0.78
$<T_e>$ (keV)	1.7	2.5	1.83	2.7	1.77	2.44
β_N% (mT/MA)	2	3	3.5	5.2	2.1	3.5
f_B	0.64	0.96	0.74	1	0.57	0.61

Tab 1.: *Plasma parameters of four FTU-D equilibria that could be achieved in the presence of 4MW of heating*

On the contrary high β operation in presence of a large bootstrap fraction can be limited by the onset of neoclassical tearing modes (NTM) well below the ideal limit. Stabilization of these modes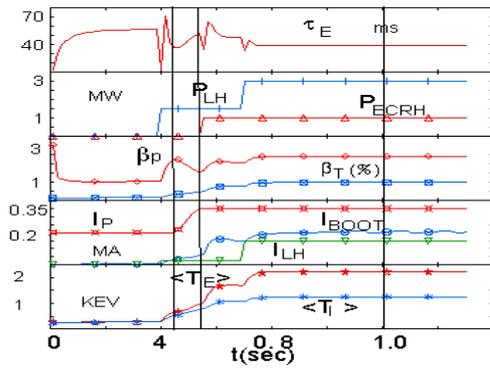 can be pursued in FTU-D by ECRH. Recent results on complete suppression of NTM in ASDEX-U [9] by ECRH, showed that 10% of the heating power was necessary to stabilize the island. Assuming a similar figure, FTU-D could need 400KW of ECRH to the same aim. The ECRH system, therefore, could be partially (1 gyrotron) used to stabilize NTM, and partially (3 girotron) to heat the plasma. The poloidal-launching angle makes it possible to drive the deposition at the desired radial position. As far as the LH power is concerned, it is possible to show, on the bases of the FTU results on CD efficiency at high density, that the LHCD capability is high enough (240kA) to drive all the FTU-D residual current. The LH antennas have circular shapes which cannot match everywhere the plasma shape in DN or SN configuration. However we evaluated that, while the reflectivity at the middle would be 7% as in circular plasmas, the reflectivity in the farest grill should not be higher that 15%. Furthermore gas puffing is planned to be extensively used for better coupling purpose in H-mode. Recent results on JET showed a

Fig.1: NCS simulated discharges:temporal behaviour of several quantities are shown.

large increase of the LH power coupling in H-mode discharges by injecting small amount of CD_4[10].

3.Plasma configurations

FTU-D has been designed in order to use almost all the existing poloidal field coils. Two classes of plasma equilibria have been defined and studied by using the equilibrium codes FIXFREE and MAXFEA, corresponding to up-down symmetric DN equilibria and SN equilibria, respectively. The reference equilibria are shown in Fig 3, 4. The highest plasma current in a DN plasma is 350 kA with the X-point position 1cm far from the new upper and lower toroidal limit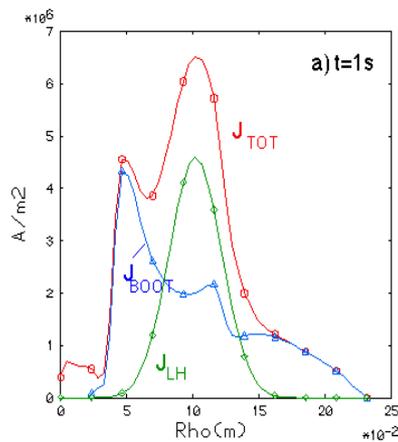ers, while the maximum plasma current in SN is 450kA with an X-point 2cm far from the protection plates. High elengation (up to 1.6) and triangularity (up to 0.8) can be achieved. The vertical stability of the plasma has been mainly studied for the equilibrium corresponding to Fig.3, using different models. The obtained growth rate is always less than =1000 sec[-1]. This result is quite robust and even in the case in which the magnetic axis is artificially located 3cm far from the magnetic axis (internally), the growth rate turns out to be =580sec-1. In order to control the elongated plasmas with the existing coils of FTU it is necessary to modify the electrical power supplies feeding the poloidal coils of the machine, therefore a new amplifier is required with a voltage rate of about 100 V/msec.

Fig.2:NCS simulated discharge at 5T and n_{Line}=1.5 10^{20}m^{-3}: Total, bootstrap and LH current prof.

4.Main Engeneering upgrade

The main engeneering upgrade consist on 1) the installation of lower and upper limiters and 2) a new central solenoid. Operations of the FTU machine with D-shaped elongated plasmas require the protection of the upper and lower parts of the vacuum vessel (V.V.) in order to avoid damage to the first wall from thermal loads. Thus two new toroidal limiters, an upper one (UTL) and a lower one (LTL) are envisaged inside the vacuum vessel. The new limiters consist of a series of TZM tiles (molybdenum alloy, used also for the ETL tiles of FTU), bolted via a single/central bolt on a support structure which is attached to the thick rigid sectors of FTU. The tiles are cooled between pulses via conduction and radiation to the V.V. which is close to liquid nitrogen (LN) temperature. In the most pessimistic assumption (**i.e.**when all the conductive/convective thermal loads are supposed to hit only one of the four toroidal tile rows) the thermal loads on the tiles result in an acceptable temperature rise of ~ 800°C and peak thermal stresses of ~250 MPa. The limiters can be cooled effectively between pulses by radiation and conduction to the V.V. Installation of the new limiters will be done with remote handling equipment via the access ports. FTU has been designed to work with circular plasmas. The transformer coils are, therefore fed in series, with currents in the same direction producing attractive force on the transformer coils.

With the D shaped plasmas, in a limiter or X-point, unbalancing currents between the different coils is required. In particular a current flowing in the opposite direction in one coil with respect to the others is needed for some plasma shaping. A new central solenoid with a new mechanical structure, able to support repulsive forces between the poloidal field coils, is needed. The new mechanical structure consists of two (upper and lower) collars linked by a central post. A double spring washer system allows the thermal expansion in the transformer while maintaining also the pre-load during operation.
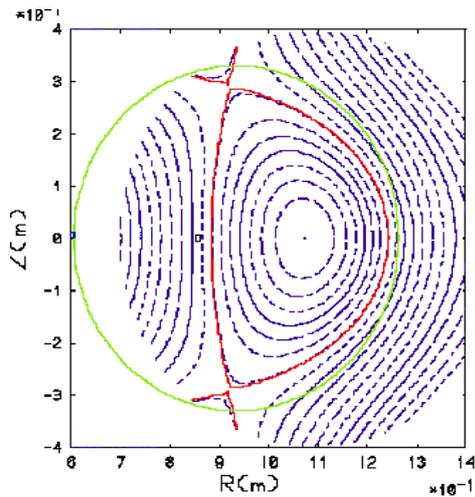

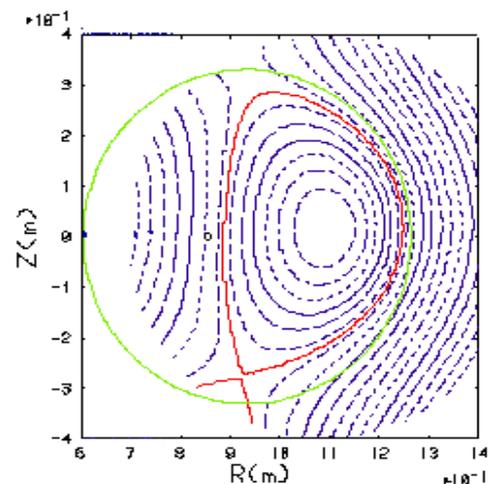fFig.3: D.N. Equilibrium configuration at Ip=350KA, =1.58, =0.8.

Fig.4: S.N. Equilibrium configuration at Ip=450KA, =1.47, =0.4.

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