Reinstallation of the COMPASS–D Tokamak in IPP ASCR

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The COMPASS–D tokamak, originally operated by UKAEA at Culham, UK, will be reinstalled at the Institute of Plasma Physics (IPP) AS CR. The COMPASS device was designed as a flexible tokamak in the 1980s mainly to explore the MHD physics. Its operation (with D–shaped vessel) began at the Culham Laboratory of the Association EURATOM/ UKAEA in 1992.

The COMPASS–D tokamak will have the following unique features after putting in operation on IPP Prague. It will be the smallest tokamak with a clear H–mode and ITER–relevant geometry. ITER–relevant plasma conditions will be achieved by installation of two neutral beam injection systems (2×300 kW), enabling co– and counter– injections. Re–deployment of the existing LH system (400 kW) is also envisaged. A comprehensive set of diagnostics focused mainly on the edge plasma will be installed.

The scientific programme proposed for the COMPASS–D tokamak installed in IPP Prague will benefit from these unique features of COMPASS–D and consist of two main scientific projects, both highly relevant to ITER – Edge plasma physics (H–mode studies) and Wave–plasma interaction studies.

The COMPASS–D tokamak will offer an important research potential as a small, flexible and low–cost facility with ITER–relevant geometry.

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1 Introduction

Recently, a possibility to continue and broaden tokamak research at IPP has appeared when the tokamak COMPASS–D (Fig. 1) was offered to IPP Prague by UKAEA. The installation of this tokamak in IPP will allow performing research on a considerably higher level and to focus on ITER–relevant physics and technology. The COMPASS–D tokamak has an ITER–like plasma shape (Fig. 2) and, after installation of an ion heating system, it will be able to access plasma parameters relevant in many aspects to ITER. The COMPASS–D tokamak will be an ideal device for detailed studies of edge plasma physics in the H–mode regime (e.g., pedestal, ELMs, turbulence, etc.)

The COMPASS (COMPact ASSembly) device was designed as a flexible tokamak in the 1980s mainly to explore MHD physics in circular and D–shaped plasmas. Its operation began at the Culham Laboratory of the Association EURATOM/UKAEA in 1989. In 1992, COMPASS was restarted with a D–shaped vessel (COMPASS–D). It is equipped with a unique fully configurable, four quadrant set of copper saddle coils to create resonant helical fields. A hydraulic vertical preload has to be used for toroidal magnetic field above 1.2 T.

During COMPASS–D operation in UKAEA, extensive experiments were performed on error–field modes (pioneering the basis and design of the error field correction coils on ITER), and ECRH experiments including current drive. H–modes were found and studied extensively; a particular feature of COMPASS–D is its capability to produce clear H–mode ohmically at low field ($\sim$1 T). However, with the growth of the spherical tokamak programme at the Culham Laboratory, it was decided to mothball COMPASS–D in 2001, because of the lack of manpower and hardware resources to run both the MAST and COMPASS–D programs simultaneously. The potential of COMPASS–D and its scientific programme were therefore not further exploited.

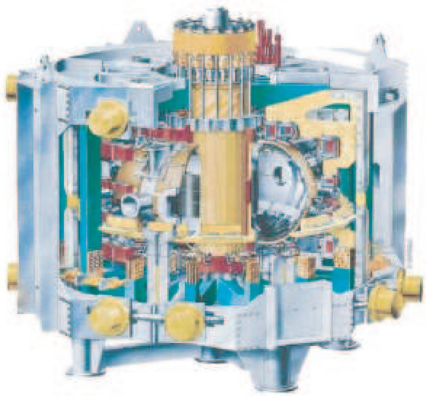

Fig. 1. Sketch of the COMPASS–D tokamak.

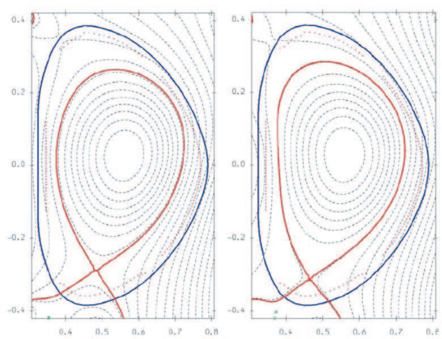

Fig. 2. Examples of ITER–like shapes of plasma in COMPASS–D.

The main parameters of the COMPASS–D tokamak envisaged in IPP Prague are summarized in the Table 1. Two successive steps of the COMPASS–D restart include:

1. transport and installation of the machine and its equipment with basic diagnostic systems, and start of operation;

2. installation of additional heating and diagnostics systems, which will allow execution of the scientific programme.

This paper is further organized as follows: Chapter 2 describes the tokamak energetics. The installation of the additional heating system is presented in Chapter 3 and Chapter 4 introduces the Plasma performance. The scientific programme and the diagnostics are presented in the Chapter 5.

Parameter	**After Step II**
R	0.56 m
a	$0.18 \div 0.23$ m
I_{p}(max)	200 kA
B_{T}(max)	1.2 T
Elongation	1.8
Shape	D, SND, ellipse
Pulse length	1 s
P_{LH}, 1.3 GHz	0.4 MW
P_{NBI}	2×0.3 MW

Table 1. Main parameters of the COMPASS–D tokamak.

2 Tokamak energetics

The COMPASS–D tokamak required electrical input power of 50 MW for pulse duration about 2–3 s during its operation in UKAEA. Such power was accessible in Culham Laboratory directly from the 33 kV grid powering the JET device.

However, only 1 MW power is available from the 22 kV grid at the campus of the Academy of Sciences in Prague, where IPP is located. Therefore, several solutions have been examined to provide the necessary input power. As a conclusion, it is proposed to accumulate of more than 40 MJ of the energy (90 MJ for full performance) from the existing electrical grid. This energy will be accumulated by a single flywheel generator (similar system is used for other tokamaks like JET, ASDEX–U, JT–60U, TCV etc.) with energy storage of approx. 45 MJ. For the future upgrade to higher toroidal magnetic field 2.1 T, the installation of an additional flywheel generator with the same parameters is envisaged.

The magnetic field system in the COMPASS–D tokamak is required to perform the following functions:

– Provide the toroidal field.

– Set up and sustain the plasma current.

– Maintain plasma equilibrium.

– Control plasma shape.

– Control plasma instabilities.

Three independent coil systems are used:

1. The toroidal field coils (TF).
2. The poloidal field coils (PF) which are a composite set of windings to provide the different requirements of magnetic flux linkage to set up and sustain the plasma current ("M" windings), the equilibrium field to correctly position the plasma over both long timescales ("E" windings) and short timescales ("F" windings), and a shaping field to provide the different plasma shapes ("S" windings).
3. Instability control windings.

The main energy consumers are TF and PF coils. The plasma is created by the start–up ohmic heating (OH) system and sustained by the magnetic field power supply (MFPS) and, eventually, by an additional heating and current drive. Currents in TF and PF coils are controlled according to predefined current waveforms. The slow ($\sim$ 100 Hz) feedback signals control the output current of AC/DC convertors. The fast (5 kHz) power amplifiers control the plasma position. Operation of all parts of the power supply system is computer controlled and monitored.

	Range of power [MW]	**Range of voltage [V]**
Toroidal field	9* $\div$ 27**	150 $\div$ 500 DC
Poloidal fields	7* $\div$ 19**	200 $\div$ 650 DC
Plasma current start–up	9	2 000 DC
Additional heating	3	3 $\times$ 400 AC
Fast feedback amplifiers (vertical and horizontal position control)	1.5	$\pm$ 50 DC
Experimental amplifiers (instability control)	0.3	$\pm$ 50 DC

*Toroidal field = 1.2 T / 1 s flat top, consumed energy $\sim$ 35 MJ, max. total power 30 MW.
**Toroidal field = 2.1 T / 0.5 s flat top, consumed energy $\sim$ 90 MJ , max. total power 60 MW.

Table 2. Power and voltage rating.

The power supply system will be put in operation in two successive stages. In the first stage, the TF current is limited by 52 kA (with the flat top phase of 1 sec). In the second stage, the TF currernt will be increased up to 92 kA with the flat top phase 0.5 sec. The system of AC/DC convertors is designed to be modular to enable upgrade to operation at higher TF current corresponding to the toroidal magnetic field $B = 2.1$ T.

The convertors are two–quadrant thyristor bridges with a freewheeling branch, over–voltage protection and a feedback system to control plasma and coil currents.

3 Additional heating

COMPASS–D will be upgraded with a new Neutral Beam Injection (NBI) system for additional heating and current drive applications (see [1]). The design is optimized for specific properties of COMPASS–D. It is a compact tokamak for which, due to short trajectory of interaction between neutrals and plasma, the NBI power, energy and geometry is chosen carefully. There are also major limitations of the tokamak structure. In particular, the toroidal field coils and the supporting structure inhibit selecting arbitrary injection directions, and the existing ports have to be modified to fit the beam injection.

NBI will provide a flexible heating and current drive system, which will consist of two injectors with particle energy 40 keV and 300 kW output power, delivering 600 kW of total power to the plasma. The basic configuration shown in Fig. 3a is optimized for plasma heating. The parameters of the COMPASS–D NBI system are summarized in Table 3. The tangential injection is also optimal for absorption due to the longest passage through the plasma achievable on COMPASS–D. Both beams are aimed in co–direction with respect to the plasma current to minimize the orbit losses. The aiming of both injectors can be shifted outside to achieve off–axis heating and current drive.

Number of injectors	2
Energy of the beam	40 keV (can be decreased)
Total ion current	2×15 A
Total power in neutrals	2×310 kW
Pulse length	300 ms
Beam diameter	$<$ 5 cm

Table 3. NBI system parameters.

For balanced injection both injectors will be located at the same port, aiming in co– and counter–current directions as shown in Fig. 3b. With proper power modulation to compensate different orbit losses for co– and counter– beams, one can obtain NBI heating scenario with minimum momentum input.

Normal injection will be also possible. This case is more suitable for diagnostic purposes, such as charge exchange radiation spectroscopy or motional Stark effect measurements.

Intensive and detailed computations have been performed to simulate NBI behavior in the COMPASS–D tokamak. Codes FAFNER (IPP Garching), NBEAMS and ACCOME were adopted for these purposes. FAFNER results for tangential injection are shown in Table 4. The shine–through for moderate densities is already

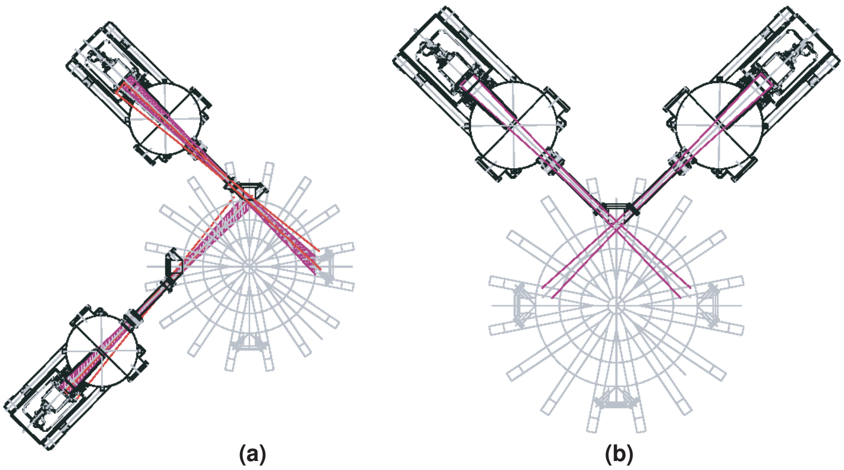

Fig. 3. Schematics of the co–injection (a) and balanced injection (b).

only 7 %. NBI in co–injection setup can drive up to 80 kA of toroidal current, which almost 50 % of the total plasma current. Unfortunately, in operation with $B_\mathrm{T} = 1.2$ T which limits the plasma current to approx. 200 kA, the orbit losses for counter–injection are around 50 %. The situation greatly improves in the case of $B_\mathrm{T} = 2$ T and $I_\mathrm{p} = 350$ kA, where the orbit losses go down to only 24 % for counter–injection and are negligible for co–injection.

4 Plasma performance

A number of combined ACCOME [2] and ASTRA [3] simulations were carried out in order to assess the performance of COMPASS with the planned neutral beam (NB and lower hybrid (LH) systems (for more details see [4]).

The simulations were carried out under consideration of the following performance targets for the operation regime $I = 0.2$ MA, $B = 1.2$ T:

- large $\beta = 2\mu_0\langle p\rangle/B_\phi^2$, but limited to $\beta_\mathrm{lim}\,[\%] \cong \frac{3I[\mathrm{MA}]}{a[\mathrm{m}]B_\phi} = 2.5\,\%$ in order to avoid ballooning and/or disruptions, etc.
- large β_poloidal for large fraction of bootstrap current $\beta_p = \frac{2\mu_0\langle p\rangle}{\langle\langle B_\Theta\rangle\rangle^2} \simeq \frac{4\pi^2 a^2\langle p\rangle(1+\kappa^2)}{\mu_0 I_p^2} \cong 0.5$
- large fraction of non–inductive driven current to assist ohmic coil $d\Psi/dt$ for longer pulse duration
- possibility of NB off–axis power deposition to support reversed shear configuration and an ITB

	scenario A		scenario B	
	Co	Counter	Co	Counter
Shine through	1 %	1 %	2 %	2 %
Orbit loss	13 %	52 %	2 %	24 %
Charge exchange loss	7 %	3 %	7 %	7 %
Power to ions	57 %	28 %	61 %	42 %
Power to electrons	22 %	16 %	28 %	25 %
Toroidal fast ion current	30 kA	19 kA	28 kA	22 kA
Current driven by NBI	20 kA	14 kA	18 kA	15 kA

Table 4. Main results obtained from the FAFNER code. Scenario A corresponds to SND equilibrium, $B_0 = 1.2$ T, $I_{\mathrm{P}} = 200$ kA,$< n >= 4 \times 10^{19}$ m^{-3} and scenario B to SND equilibrium, $B_0 = 2$ T, $I_{\mathrm{P}} = 350$ kA, $< n >= 3.5 \times 10^{19}$ m^{-3}

– high density operation, $\geq 3 \div 6 \times 10^{19}$ m^{-3}, for ELMy H–mode. This range falls well within the Greenwald limit [3].

The simulations proceed in a sequence of iterations between the two codes in order to reach a consistent state between power deposition profiles from ACCOME needed by ASTRA, and temperature profiles from ASTRA needed by ACCOME. We concentrated on the operating regime $I_{\mathrm{p}} = 0.2$ MA and $B_{\mathrm{T}} = 1.2$ T with the main results shown in Tables 5 – 6, but we were also interested in main features of operating regime $I_{\mathrm{p}} = 0.35$ MA and $B_{\mathrm{T}} = 2.1$ T , shown in Table 5.

We examine auxiliary heating and current drive operation in the two basic COMPASS–D single–null magnetic equilibrium configurations: SND (low triangularity $\Delta \approx 0.3 \div 0.4$) and SNT (high triangularity $\Delta \approx 0.5 \div 0.7$).

All of the results in Tables 5 – 6 are obtained at peak density $n_0 = 3 \times 10^{19}$ m^{-3}. The density profile was prescribed in form

$$n(r) = n_0[(1 - n_b)(1 - \rho^2)^{1.5} + n_b]\,, \tag{1}$$

where n_b is the plasma edge density, ρ is a normalized equivalent radial coordinate (square root of toroidal flux), and the exponents 2 and 1.5 were selected to approximate the density profile of previous COMPASS–D auxiliary heating experiments.

In Tables 5 – 6 we note global tendencies – the deposited NB and LH powers, the driven NB and LH currents, and the peak temperatures $T_{\mathrm{e}0}$, $T_{\mathrm{i}0}$, as function of NB injection geometry, i.e. on–axis or off–axis, balanced or co–injected beams, and as function of magnetic equilibrium, i.e. of SND or SNT.

Further, we note the substantial effect of on–axis or off–axis NB incidence. While strong ion heating is noted for on–axis incidence, very little heating occurs for off–axis incidence. On the other hand, off–axis co–NBI incidence can lead to reversed shear. Since about 20÷30 % of the absorbed NB power is deposited on the electrons, NBI is therefore an important electron heating mechanism, in fact more efficient than LH heating at $B_{\mathrm{T}} = 1.2$ T conditions. This is clear from Table 5 – 6.

equilibrium	P_{NB} [kW]	P_{LH} [kW]	I_{NB} [kA]	I_{LH} [kA]	T_{e0} [keV]	T_{i0} [keV]
SND on–axis $B_{\mathrm{T}} = 1.2$ T	505	122	80.0	45.6	1.80	4.29
SNT on–axis $B_{\mathrm{T}} = 1.2$ T	505	181	83.0	53.0	1.10	2.05
SND on–axis $B_{\mathrm{T}} = 2.1$ T	519	120	83.7	101	2.10	4.60
SNT on–axis $B_{\mathrm{T}} = 2.1$ T	514	195	82	1.39	1.88	3.66

Table 5. Comparison of $B_T = 1.2$ T and $B_T = 2.1$ T results.

equilibrium	P_{NB} [kW]	I_{NB} [kA]	T_{e0} [keV]	T_{i0} [keV]
SNT on–axis (co–injection)	506	72.6	1.24	2.44
SNT off–axis (balanced inj.)	522	$\approx$ 0	1.02	1.47

Table 6. $B_{\mathrm{T}} = 1.2$ T, $n_0 = 3{\times}10^{19}$ m^{-3}, comparison of co– and balanced NBI.

Summary of results from ACCOME–ASTRA simulations of the COMPASS–D tokamak:

– Ion and electron heating from NBI depends sensitively on the NB power deposition profile because of very high NB power density

– T_{e} and $\mathrm{T_i}$ depend on NB co– or counter–injection

– Strong ion heating observed for on–axis co–NBI ($T_{\mathrm{i0}} \approx 2$ keV)

– $\chi_{\mathrm{i}}/\chi_{\mathrm{neo}} \sim 2 \div 2.5$ in regimes with strong central ion heating

– Weak ion heating but reversed shear observed for off–axis NBI

– LH absorption is weak because of poor slow LH wave accessibility

– LH electron heating depends sensitively on $T_{\mathrm{e}}(\mathrm{r})$ and equilibrium

– The SNT equilibrium is more favorable to LHCD and heating than is SND because SNT has more poloidal asymmetry which leads to larger $n_{//}$ upshifts

5 Scientific programme

The uniqueness of the contribution of COMPASS–D will originate from the fact that this is one of the most accessible and economically operating experiments that can produce H–mode plasmas. Therefore, the emphasis will be on topics which can be studied in the proper plasma regime, yet with a combination of diagnostics which is not available or not feasible in other experiments.

The scientific and technological programme should benefit mainly from the following characteristic features of the COMPASS–D tokamak:

– ITER–like geometry with a single–null–divertor (SND) magnetic configuration

– Tokamak with a clear H–mode

– Neutral beam injection heating system

– Unique set of saddle coils for the resonant magnetic perturbation

– Lower hybrid wave system

In order to reach plasma conditions relevant to ITER, the plasma performance will be improved by installation of a Neutral Beam Injection (NBI) system for an additional ion heating. The COMPASS–D tokamak with considerably enhanced diagnostic systems focused on the plasma edge will then be an ideal device for H–mode studies. Very useful for these studies will be the data obtained in the NBI heated ELMy H–mode. The NBI system will consist of two injectors enabling both the co– and counter–injection. These unique features will allow study of the impact of the external momentum input and NBI–driven current on plasma performance. In addition, COMPASS–D will be particularly useful in benchmarking plasma modelling codes.

Therefore, the specific topics of the scientific programme for the COMPASS–D at the IPP.CR are:

1. Edge plasma physics
   (a) H–mode studies
   (b) Plasma–wall interaction
2. Wave–plasma interaction studies
   (a) Parasitic lower hybrid wave absorption in front of the antenna
   (b) Lower hybrid wave coupling

There exist still considerable uncertainty to the appropriate scaling for the pedestal width, in particular whether it is entirely determined by the plasma physics (implying that the dimensionless parameters ν^*, ρ^* and β are important) or whether such processes as neutral penetration play a role. Since experiments in COMPASS–scale devices have been typically performed using ohmic or electron heating (e. g. COMPASS–D, TCV), or may have been affected by other factors (e. g. TF ripple in JFT–2M), NBI–heated experiments in COMPASS–D should constitute an extension of the present database in ν^* and ρ^* and could therefore provide additional insight into the scaling and physics processes determining of the H–mode pedestal width. This is particularly so since a detailed edge diagnostic will be installed in the COMPASS–D tokamak. The difference in neutral penetration depth relative to larger, hotter tokamaks may also help to clarify the role of neutrals, if any, in determining the pedestal width.

Important edge problems for ITER are ELM physics and ELM control. The COMPASS–D tokamak used to be the smallest machine with good H–modes. The threshold for L–H transition was achieved and ELM–free and Type III ELMy (both OH and ECRH) discharges were attained in the COMPASS–D tokamak at Culham, UK.

It is not obvious whether the currently most desirable Type I ELM regime can be achieved in COMPASS–D even with a new neutral beam system. Based on observations from tokamaks in the AUG through JET, to have “clean” Type I ELMs one needs at least $1.3 \div 1.8{*}P_{th\_L/H}$ ($P_{th\_L/H} = 150$ kW for $B_{\mathrm{T}} = 1.2$ T) depending on plasma shaping), which is well covered by the installed NBI power of 600 kW.

However, Type I ELMS were not observed on COMPASS–D even with 1 MW of ECRH power. This could mean that at present there is no physics basis for expecting the quoted threshold to hold at the COMPASS scale. Previous results from devices such as COMPASS–D and TCV may not be relevant, since they are either ohmically heated or electron heated – and it is known that electron heated devices can have more difficulty accessing good Type I H–modes, since they typically have to work at lower density (to ensure access for ECRH), and it is known that the H–mode power threshold scaling turns up at low densities, possibly because of the decoupling of electrons from ions. This assumes, of course, that one takes the normal precautions, including vessel conditioning, which are used on larger devices. In this respect, COMPASS could represent a valuable test of our understanding of H–mode access physics. This would be particularly valuable when combined with the very good edge and pedestal diagnostics.

Furthermore, even if Type I ELMs are not achieved, the Type–III ELMs regimes can be studied since, at high collisionality, they exhibit similar ballooning (resistive modes however) features as Type I ELMs. All dynamics, space structure, turbulence during ELMs, changes of the electric field and rotation during ELMs near the separatrix etc. have a lot of common features with Type I regimes.

Since COMPASS–D will focus on this area, the planned edge and pedestal diagnostics could yield a great deal of detailed information on the phenomena associated with ELMs.

For the study of edge stability or ELMs, a major requirement is the reconstruction of the plasma equilibrium including meaningful experimental constraints on the edge pressure gradient and edge current density. Therefore, the key for new contributions to this field is diagnostics that removes measurement uncertainties, which have been the limiting factor in such studies in all previous experiments. This implies high resolution measurements of the edge pressure profile, i.e. electron and ion temperatures and densities, and attempts to measure the edge current (total or even profile) directly.

All attempts will be made to obtain the best possible radial resolution in the gradient region which in COMPASS–D is only about 1–2 centimeter wide. With such a diagnostics, there are several topics which will be studied:

– Detailed studies of the H–mode transition with a combination of profile and fluctuation measurements aiming to identify the relevant driving mechanisms for $E \times B$ spin–up and turbulence suppression. One possible question to address with a combination of the probe arrays and plasma rotation measurements is whether or not a turbulence–generated rotation spin–up (Reynolds stress) can account for the rotational shear observed before and at the transition.

– Modelling is advancing from the description of linear stability towards the non–linear evolution of the ELM instability. With sufficient time resolution and a suitable combination of diagnostics, the initial ELM growth from a more or less coherent precursor mode into the usually observed broad spectrum would be helpful to guide the development of ELM models.

– Plasma rotation and edge stability – two NBI systems, each 300 kW, will be flexible to produce co– and counter– injection to generate different regimes of toroidal plasma rotation. It will be also possible to arrange the injection of both systems in the same direction if required. This possibility is important to study the influence of the plasma rotation on the pedestal stability, ELMs and external (error) fields penetration. The rotation predicted for ITER is small, so these studies can be very ITER relevant. Also the access to ELM–free QH mode (and may be also low ν^* due to the particle losses in QH mode) will be tried.

– COMPASS–D is an ideal device for detailed studies of the open issues of plasma turbulence and transport. The main aim of such studies is to build up a systematic and as detailed as possible fluctuation database in order to improve our understanding of anomalous transport scaling, transport barrier dynamics (shear flow generation) and turbulent electric field bifurcations.

  The following key scientific problems will be addressed:

  1. Bohm versus gyro–Bohm scaling and the transition between the two regimes

2. Transport barrier formation and dynamics including the different transport channels
3. Statistics of mesoscale intermittency in transport (e.g., avalanches, Self–Organized Criticality)
4. The dynamics of transport perturbation events such as heat pulse propagation

– ELM control by external coils – An interesting ITER relevant issue is ELM control by external magnetic perturbation. The idea was pioneered on COMPASS–D. Perturbations with the toroidal numbers $n = 1$, $n = 2$ can be generated. Usually low $n = 1$ can trigger core MHD and confinement degradation. But for n > 1 one can obtain edge ergodisation in order to control edge transport and ELMs like in DIII–D experiment. The results could be used to scale and understand such control systems also for ITER.

  This area of plasma control is being developed and the capability of COMPASS to explore various mode structures for the applied field could make a significant contribution to the understanding of how external coils can be exploited to control ELMs.

A comprehensive set of diagnostics will be installed in the COMPASS–D tokamak in order to meet the goals of the planned scientific programme. Most of the diagnostics will focus on the plasma edge, namely on the pedestal region. Therefore, the spatial resolution will be in the range of $1 \div 3$ mm. The list of the diagnostics is shown in the Table 5.

6 Conclusion

The COMPASS–D tokamak with NBI heated H–mode plasma will be – together with JET and ASDEX–U – one of the few tokamaks with highly ITER relevant plasmas. COMPASS–D will allow, due to its size, a highly flexible, fast and low–cost implementation of technical modifications and programmatic adaptations if needed to optimise ITER relevant results. Therefore, the device will be particularly suited for the exploration of novel regimes or concepts (such as the investigation of ELM mitigation by using saddle coils).

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	PLASMA DIAGNOSTICS
STEP I	Magnetic diagnostics
	Microwave interferometer
	D_α and Z_{eff} measurement
	Fast camera for visible range
	Thomson scattering
	Fast bolometers
	SXR – array of diodes
	Neutral particle analyzer
	VUV and XUV spectrometers
	Langmuir probes
STEP II	Beam diagnostics
	Microwave reflectometry
	PWI ex-situ diagnostics

Table 7. Overview of the diagnostics in the COMPASS–D tokamak.

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