JET: Preparing the future in fusion

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JET (Joint European Torus) is the largest tokamak in the world and the only fusion facility able to operate with Tritium, the fusion fuel, and Beryllium, the ITER first wall material. JET also features the most complete remote handling equipment for in–vessel maintenance. As a multinational research center, JET provides logistic experience in preparing for operation of the global facility, tokamak ITER.

Experiments on JET are focused on ITER–relevant studies, in particular on detailing the operational scenarios (ELMy H–modes and advanced regimes), on enhancing the heating systems, on developing diagnostics for burning plasmas etc. Pioneering real–time control techniques have been implemented that maximize performance and minimize internal disturbances of JET plasmas. In helium plasmas, ion cyclotron heating (ICRH) created fast α–particles, mimicking their populations in future burning plasmas. The recent successful Trace Tritium campaign provided important new data on fuel transport. Current enhancements on JET include a new ITER–like ELM–resilient high power ICRH antenna (7 MW) and over twenty new diagnostics that will further extend the JET scientific capabilities and push the facility even closer to the ITER parameters.

A special mention is given to the involvement of the fusion experts from Association EURATOM–IPP.CR, who have been actively participating in the collective use of JET facility for more than three years.

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1 JET, the European center of fusion research

JET (Joint European Torus, [1], see Fig. 1) with the plasma major and minor radii $R = 2.96\,\mathrm{m}$ and $a = 1.25\,\mathrm{m}$, respectively, is the world biggest tokamak. Plasma with approx. volume $80\,\mathrm{m}^3$ is confined by magnetic field with toroidal on–axis magnetic field B_T max. 4 T, with plasma currents I_p up to 4 MA. The central plasma ion temperature can reach up to 40 keV, plasma densities have the order

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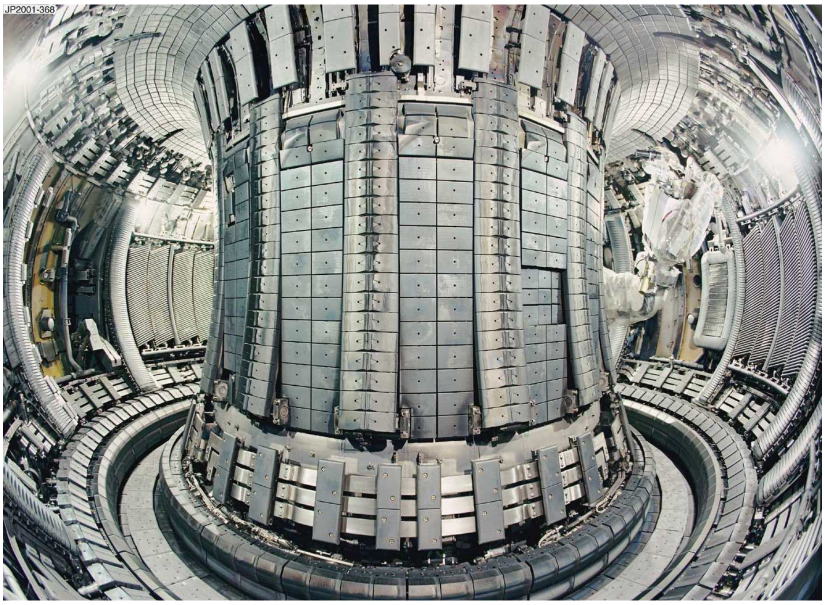

Fig. 1. Wide–angle view inside the JET torus, with Remote Handling arm in the right side. The carbon tiles coverage and the divertor region in the bottom are apparent.

of 10^{20} m^{-3} and the energy confinement time $\tau_{\mathrm{E}} \sim 1\,s$. Additional heating power is provided by Neutral Beam Injectors (NBI, 80/140 keV, installed power 23 MW) and Ion Cyclotron Resonant Heating (ICRH, $23 \div 57$ MHz, installed power 32 MW), non–inductive plasma current is driven by Lower Hybrid Current Drive microwaves (LHCD, 3.7 GHz, installed power 12 MW). JET plasma discharges last typically 20 s (up to 60 s with additional current drive) and can be run every 20 minutes.

With its size, operation capabilities and plasma shape, JET is the closest fusion facility to the next step tokamak, ITER. The worldwide project ITER will sustain burning plasmas with fusion power $500 \div 700$ MW and power amplification factor $Q \sim 10$ [2], [3]. JET is at present the only machine capable of using the fuel of the future burning plasma experiments, tritium and deuterium, and holds the record in actual fusion energy release (16 MW, $Q = 0.65$, in 1997, see e.g. [1], [4]). JET is also the only machine in the world capable of using Beryllium on the first wall (i.e. on plasma facing components), the material foreseen for the ITER first wall. All in–vessel operations are carried out by the world most complete remote handling system, from simulations in Virtual Reality to optimised action of the robotic arm.

Another significant asset of JET with regards to both ITER and European research endeavours is its multinational character, unique in scale among fusion facilities. Since 2000, JET is a user facility servicing the European Development Agreement (EFDA) [5]. Scientists from 16 European countries and associated collaborators from China, Russia and USA extensively participate in JET experimental campaigns. Remote participation in data analyses as well as in workshops, seminars and meetings has become routine. With the average age of visiting sci-

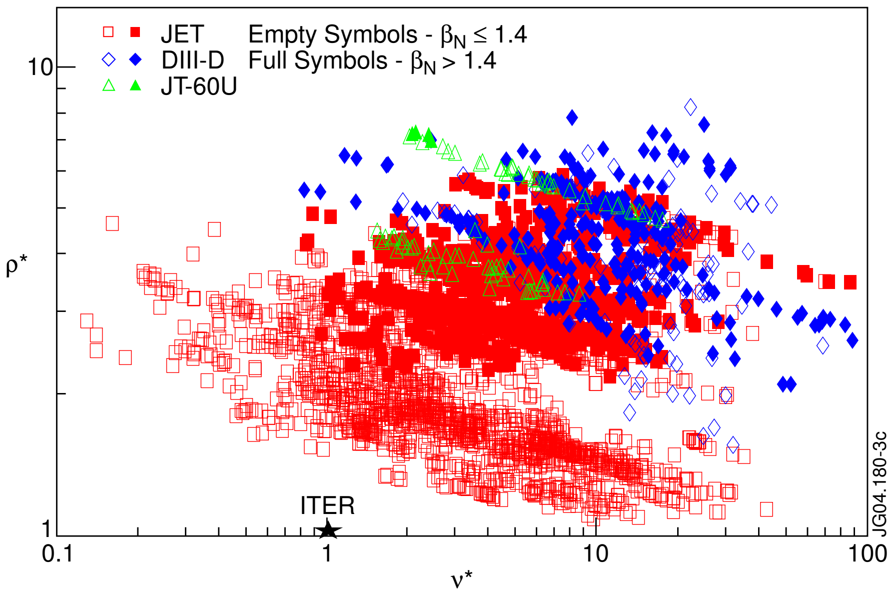

Fig. 2. Performance of the three major tokamaks in the terms of ρ^* vs. ν^*, normalized to the projected ITER value. Notice the unique position of JET data with respect to ITER.

entist around 40 years, JET plays a considerable role in training the ITER generation. EFDA JET is a member of EIROforum, the platform of seven European Intergovernmental Research Organisations[1]) who are key partners of the European Commission in building up the European Research Area [6].

2 Overview of Recent JET Results

Detailed overviews of the JET Results are regularly presented at selected international fusion conferences, e.g. at IAEA Fusion Energy Conferences, and in the corresponding editions of scientific journals, see e.g. [7]. For special terms refer to the glossary at the end of the article.

The highest priority in the present JET programme is the ITER–relevant research. The reference scenario for ITER is the inductively driven ELMy H–Mode with high confinement ($H_{98} = 1$), high density ($n_e/n_G \geq 0.85$ where n_G is the Greenwald density) and high normalized pressure ($\beta_N > 1.8$). In addition, the heat load to the divertor has to be within acceptable boundaries. Research on the ELMy H–mode now concentrates on demonstrating plasma regimes with values for key physics parameters like normalized Larmor radius (ρ^*) and normalized colli-

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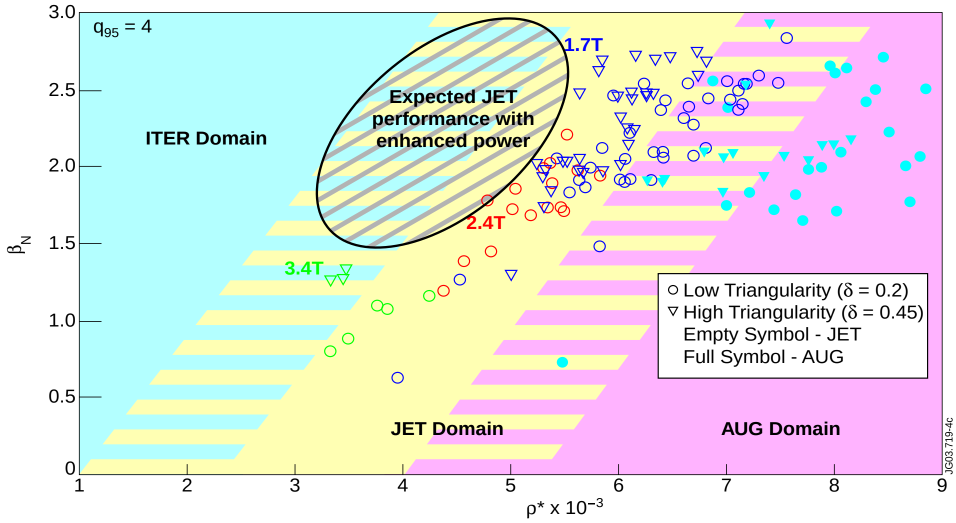

Fig. 3. Normalized pressure as a function of normalized Larmor radius and triangularity in the JET–ASDEX Upgrade identity and similarity experiments.

sionality (ν^*) as close as possible to ITER [8], see Fig. 2, consolidating the scaling of these regimes and mitigating transient phenomena (ELMs, disruptions) to extend the lifetime of plasma facing components. At JET, ELMy H–modes were achieved with simultaneously $H_{98} = 0.95$, $n_\mathrm{e}/n_\mathrm{G} \sim 1$ and $\beta_\mathrm{N} > 1.8$, i.e. above ITER reference values, at plasma currents up to $I_\mathrm{p} = 3.5$ MA. Extended ELMy H–mode scaling studies in ρ^*, ν^* and β_N confirm the predictions for ITER confinement with rather favourable margins. Divertor heat loads were reduced in ITER–like (high triangularity, $\delta \sim 0.45$) plasmas at currents up to 2.5 MA with impurity seeding using Nitrogen and Argon.

Two types of Advanced regimes are foreseen for ITER operation and are under development on JET: (i) regimes with strong reversed shear of magnetic field ($q_0 = 2-3$, $q_\mathrm{min} = 1.5-2.5$) creating an Internal Transport Barrier (ITB), which could be suitable for fully non–inductive steady state ITER operation [9], and (ii) "Hybrid" regimes with flat monotonic q profile ($q_0 \approx 1$) at reduced current ($q_{95} \approx 4$), which could be suitable for very long pulse semi–inductive operation and high neutron fluence testing on ITER [10].

Recently at JET, ITBs were sustained in ITER–like scenarios ($T_\mathrm{e} \sim T_\mathrm{i} \sim 7$ keV and low toroidal rotation) by balanced NBI and ICRH heating. Wide radius ITBs located at $r/a \sim 0.6$ at ITER–like plasma triangularities ($\delta \sim 0.45$) were achieved. ITB plasmas extended to 20 seconds with full current drive ($I_\mathrm{p} = 1.8$ MA at $B_\mathrm{T} = 3$ T) and a JET record injected energy of approx. 330 MJ, illustrating the potential of Advanced regimes for fully non–inductive steady state ITER operation. ITB operation at a core density close to the Greenwald limit ($n_\mathrm{e0}/n_\mathrm{G} \approx 1$) was also demonstrated with the pellet injection.

In parallel, significant progress in the development of Hybrid scenarios was achieved. At JET, the regime was extended to lower ρ^* values (up to $I_\mathrm{p} = 2.8$ MA, $B_\mathrm{T} = 3.2$ T), although β_N was limited by the maximum available heating power, see Fig. 3.

Substantial contributions to all ITER operational scenarios were made due to continuous development of JET Real Time Control capabilities. The implemented real time diagnostics and control techniques proved to enhance performance of the JET plasmas and minimise the internal disturbances [11]. In a recent campaign, pressure and safety factor profiles were simultaneously feedback controlled in ITB plasmas ($I_\mathrm{p} = 1.7$ MA, $B_\mathrm{T} = 3$ T) for up to 7 seconds for the first time. Dual feedback control of the confinement parameter H_{98} and radiated power fraction $\mathrm{P}_\mathrm{rad}/\mathrm{P}_\mathrm{tot}$ with deuterium and argon injection as actuators in high triangularity led to steady state ELMy H–mode plasma with $n_\mathrm{e}/n_\mathrm{G} = 1.1$ and $H_{98} \sim 1$.

Accelerating ^{4}He beam particles to several MeV by third harmonic ICRH proved to be an invaluable technique for simulating fusion–produced α–particles without the complications of a full scale deuterium–tritium (D–T) experimental campaign [12]. It is particularly important for future tokamak reactors, including ITER, to gain information on transport of α–particles and to test the α diagnostics in an early non–activating phase of operation. For example, γ–ray imaging allowed to measure changes in the fast ion distribution function during MHD instabilities such as sawteeth or Alfvén Cascades excited by ICRH accelerated ions. Slowing–down α–particles were followed by a new diagnostic method based on XUV spectroscopy, while lost α–particles were detected by measuring the activation induced in suitable samples located near the plasma edge, the method being used for the first time in a large fusion experiment.

The Trace Tritium Experiments (TTE) campaign in October 2003 was crucial for developing heating methods related to D–T operation, for fuel transport studies and for diagnostics of burning plasmas [13]. Within the TTE campaign, the fraction of tritium in the plasma was kept at $1 \div 2\,\%$ level so that tritium fusion reactions with deuterium majority fuel acted as diagnostic tracers. ICRH of tritium at its fundamental cyclotron frequency required the highest field ($B_\mathrm{T} = 4$ T) and the lowest ICRH frequency (23 MHz) possible in JET. It produced energetic tritium tails of $80 \div 120$ keV corresponding to high D–T fusion cross–section, increasing the neutron emission by three orders of magnitude. The neutron emission profile clearly shows off–axis peaking in this case, see Fig. 4, proving that a high fraction of fast tritium ions are trapped in so–called banana orbits.

Major diagnostic issues of relevance for burning plasma were addressed in recent JET work [14]. Refractometry techniques provided greatly enhanced resolution of temporal and spatial instabilities of Alfvén Eigenmodes. The total yield of 14 MeV (D–T fusion) neutrons was determined absolutely for the first time from the Magnetic Proton Recoil Spectrometer data, in agreement with the other two independent measurements provided by the profile monitor and the delayed emission. New compact neutron detectors, with diamond and organic scintillators, have been successfully tested. Major progress is being made in measuring the profile of the plasma isotopic composition by comparing the results of the Neutral Particle Analyser (NPA), more accurate at the plasma edge, with the neutron data,

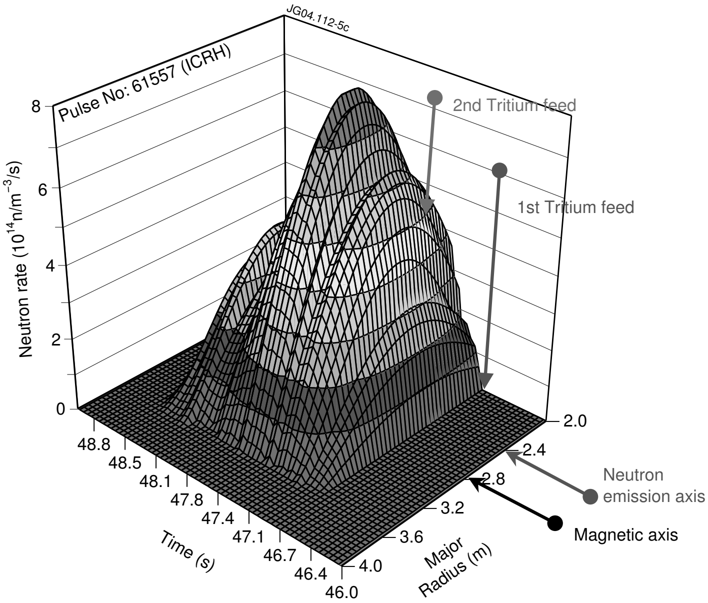

Fig. 4. Time development of 14 MeV (ie D–T fusion) neutron spatial profile in TTE discharge with ICRH of tritium. Notice that the neutron emission spectrum peaks off axis.

more reliable in the plasma core. Next, diagnostics of plasma edge and plasma–wall interactions are essential for the ITER–relevant expertise. To this end, dedicated plasma configurations were developed which allow a maximum of edge diagnostics to be used simultaneously. Lithium beam and the edge Thomson scattering were upgraded. The enhanced information on edge gradients of T_{e} and n_{e} allowed significant progress in pedestal characterization, showing different dependencies than on smaller tokamaks. Impurity seeding, tiles with tracers, slowly rotating deposit collectors and quartz microbalances are used for studies of erosion, migration and deposition of first wall materials.

3 JET enhancements in 2004

At present, JET is undergoing a major shutdown to implement new and upgraded heating components, a modified divertor and about twenty new diagnostic systems that will further extend ITER–relevant capabilities of JET [15].

The overall heating capability of JET will be increased so that up to 40 MW can be delivered to the plasma simultaneously. Apart from optimisation of neutraliser in the Neutral Beam system, a key challenge is to prepare a newly designed ITER–like ICRH antenna for installation in 2005. The novel antenna is expected to be resilient to fast varying loads due to ELMs and deliver 8 MW/m^2 for 10 s.

A new divertor configuration to be installed by the remote handling system is designed to sustain the increased power in high–triangularity ITER–like scenarios ($\delta \sim 0.56$). A new disruption mitigation system based on a very fast valve will be implemented in order to test its ability to reduce the detrimental impact of disruptions to tokamak structures. Instantaneous injection of gas during disruption is foreseen to promptly radiate the plasma thermal energy instead of dumping it to the divertor.

Present enhancements also aim at complementing the ITER–relevant diagnostic capabilities of JET. New neutron detectors using the latest advances in scintillators and data recording techniques will produce much higher count rates with boosted signal–to–noise ratio. Fine energy resolution detectors for fast α–particles with high pitch angle will be installed close to the JET plasmas. The physics of the power handling will be monitored by an ambitious infrared (IR) viewing system including a new fast, wide–angle, state–of–art IR camera, and by a new high signal–to–noise bolometric system. New halo sensors – elements that measure currents flowing partly in the plasma and partly in the conductive wall materials – will be installed to allow a detailed mapping of the halo current sinks and sources in order to estimate densities of this current as well as the corresponding forces on the in–vessel components. A high–resolution Thomson scattering diagnostic with 20 Hz repetition rate will provide temperature and density profiles with a spatial accuracy of $\sim$ 2 cm. Corrugated high transmission waveguides will be used for both reflectometry and Electron Cyclotron Emission (ECE) oblique measurements, the latter being used at JET for the first time. Fast sensitive coils for diagnostics of Alfvén waves with high toroidal mode number will enhance the scope of JET experimental studies of MHD effects.

Enhancements of the JET diagnostic systems are promoting important developments in fast acquisition electronics, novel detectors, imaging sensors and materials for neutron attenuation. Significant parts of the new diagnostic systems will also be integrated into JET Real Time Control.

4 JET and the Association EURATOM–IPP.CR

The Czech republic is the only new EU country that currently operates a tokamak. The tokamak Castor ($R = 0.4$ m, $a = 0.08$ m, [16]) is located in the Institute of Plasma Physics, Academy of Sciences of the Czech Republic (IPP AS CR) in Prague. However, Czech technological capacities relevant to fusion research are broader, including, for example, material irradiation research on nuclear research reactor LVR–15 in the Nuclear Research Institute Řež plc.

At the end of 1999, a Contract of Association between EURATOM and IPP AS CR was signed. This Contract established the Association EURATOM–IPP.CR that coordinates the Czech research programmes relevant to European fusion developments and that today consists of five research institutes[2]) and two univer-

[2]) Institute of Plasma Physics AS CR (Prague), Nuclear Physics Institute AS CR (Řež), Institute of Nuclear Research plc. (Řež), J.Heyrovsky Institute of Physical Chemistry AS CR (Prague),

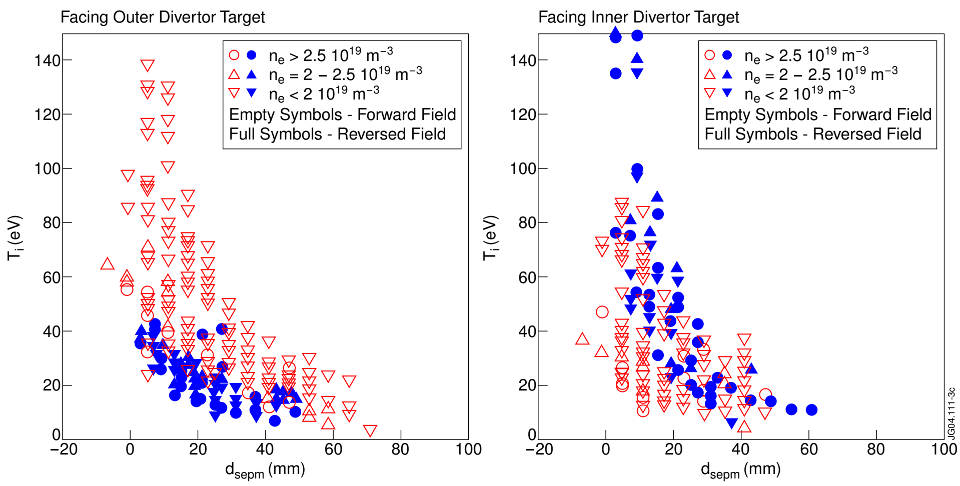

Fig. 5. Radial profiles of the ion temperature in SOL as determined using the bi–directional Retarding Field Analyser diagnostics (RFA) placed on a reciprocating probe drive. The RFA probe head faced either the outer divertor target (left) or the inner divertor target (right). The observed dependence of the T_i profiles on RFA direction and on JET magnetic field orientation is linked to the amplitude and direction of the plasma flow in the vicinity of the RFA probe head.

sities[3]). The Association is also a signatory of the European Fusion Development Agreement (EFDA) and of the JET Implementing Agreement. The latter has already allowed several Czech scientists to be directly involved in the JET scientific work, both on JET site and in remote data analyses.

In extrapolation of the Castor tokamak expertise, the Czech scientists participate on JET turbulent transport and plasma flows measurements using divertor probes and reciprocating probes in the JET Scrape of Layer (SOL) [17]. Data analyses include correlation statistics and construction of probability density function (PDF) of the signal. It is observed that PDFs are clearly non–gaussian in radial transport while they tend to be gaussian in the case of parallel flows. We also participate in data processing of other JET plasma edge diagnostic tools, including the Quartz Microbalance and the Retarding Field Analyser (RFA) [18], see Fig. 5.

Next, we take part in the JET edge plasma modelling. For example, discharges in Helium have been simulated using codes EDGE2D. Particle–in–Cell (PIC) simulations help to explain discrepancies between ion saturation currents measurement by Turbulent Transport Probe and the RFA [18]. The results indicate that the magnetic pre–sheath formed in front of the RFA is responsible for the effect.

Fast particle production in front of the Lower Hybrid antenna (grill) present another field of specific interest for our experts. Bright spots were observed that are probably caused by fast particles generated just in front of the LH grill mouth

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[19]. It was shown that these fast particles can travel several times around the torus. Angular velocity of the hot spot "trains" agrees well with variations in the safety factor q and with q ramp–down.

We also participate in tomographic reconstructions of Soft X–ray data during JET disruptions [20]. A successful implementation of the reconstruction principle, combined with Minimum Fisher Regularisation technique, was applied in the algorithm unfolding neutron spectra measured by NE213 liquid scintillator [21]. Modification of the code for the JET neutron tomography is under preparation.

Last but not least, we are currently involved in JET public information activities. The efforts in this domain are focused namely on the JET public webpage, JET Bulletin and JET Public Affairs presentations as well as on promoting JET at selected local and international events, often together with our EIROforum partners.

5 Glossary

For further details please consult [1], [22] or [23]

Divertor: a magnetic field configuration that defines last closed magnetic surface by deflecting field lines at the plasma periphery; also denotes a region, where the deflected lines divert plasma onto the tokamak first wall
Disruption: a major instability leading to a sudden termination of plasma discharge
ELM (Edge Localised mode): in the H–mode, a relaxation instability of the steep edge gradient (Edge Transport Barrier). Occurs in short periodic bursts and causes transient heat and particle loss to the divertor.
Energy confinement time: $\tau_\mathrm{E} = W_\mathrm{p}/(P_\mathrm{tot} - \mathrm{d}W_\mathrm{p}/\mathrm{d}t)$, where W_p is total plasma thermal energy and P_tot total power absorbed in the plasma
Greenwald density: $n_\mathrm{G} = 10^{20} I_\mathrm{p}/(\pi a^2)$ [m^{-3}; MA, m], empirical expression for a density limit, plasma density in most tokamaks does not exceed this value under standard conditions
H–mode: high confinement tokamak regime. Unlike the L–mode (low confinement regime), H–mode is characterised by a sharp temperature gradient near the plasma edge (the Edge Transport Barrier)
H$_{98}$ (confinement parameter): ratio of the energy confinement time in the given experiment to predictions of the energy confinement scaling law known as $\tau^{\mathrm{IPB98}(y,2)}$ (based on a multi–tokamak database of the ELMy H–mode experiments)
ITB (Internal Transport Barrier): region near the plasma core with low energy transport, characterized by sharp pressure gradient. Corresponds to magnetic shear reversal region.
Normalized collisionality: ν^* ratio of the effective collision frequency for trapped particles to their bounce frequency in the 'banana' orbits. The ratio increases with size of the tokamak, which is beneficial in terms of suppression of the enhanced transport due to trapped particles.

Normalized Larmor radius: $\rho^* = \rho/a$, Larmor radius of plasma majority ion species in the given magnetic field divided by minor radius of the tokamak
Normalized pressure: $\beta_\mathrm{N} = \beta\, a B_\mathrm{T}/I_\mathrm{p}$ [1; m, T, MA], where β is ratio of the plasma kinetic pressure and the magnetic field pressure.
Magnetic shear: $s = r/q\, \mathrm{d}q/\mathrm{d}r$, radial rate of change in the safety factor.
Power amplification factor: $Q = P_\mathrm{fus}/P_\mathrm{in}$, ratio of total fusion power to total external heating power. Under stationary conditions, $Q = 1$ and $Q \to \infty$ correspond to breakeven and ignition, respectively.
Safety factor:, $q = d\Phi/d\Psi$, rate of change of toroidal magnetic flux (corresponding to magnetic field B_T) with poloidal magnetic flux (generated by plasma current I_p), thus characterizing the helical twist of the magnetic field in the tokamak. q_0 is on–axis limit, q_{95} is the value at the magnetic surface that encloses 95% of the total poloidal magnetic flux.
SOL (Scrape–off Layer): layer of residual plasma between the last closed magnetic surface and the tokamak first wall.
Triangularity: δ, a measure of plasma cross–section triangularity. The vertical off–axis shift from a circular shape is $\sim (-r\,\delta)$. Positive plasma triangularity is beneficial for the plasma confinement.

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