Czech Technical University
Faculty of Nuclear Sciences and Physical Engineering
Department of Physics

The database for comparison of plasma parameters of the JET tokamak in various regimes of its operation

(Research project)

Martin Kubič

Supervisor: Ing. Ivan Ďuran, Ph.D
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Contents

Abstract

Presently, one of the main responsibilities of JET tokamak is to prepare the operating regimes for future fusion experimental reactor ITER, which is being built in Cadarache, France. The main aim of this report is to compare some aspects of the two ITER candidate operating scenarios, ELMy H-mode and advanced regime with internal transport barrier. For this purpose statistical approach was chosen compiling a large number of JET edge and core plasma quantities across a large discharge database to assess the level of similarity of each type of scenario. The report is focused on influence of gas impurities on plasma performance in both regimes. Mainly on temperature and density profile and radiative power fraction. In order to reduce the power load in the divertor to acceptable value of about 70%, radiation cooling by seeding of impurities is necessary. The energy balance of plasma is also presented.

List of Abbreviations

JET	Joint European Torus
ITER	International Thermonuclear Experimental Reactor
ELM	Edge Localised Mode
ITB	Internal Transport Barrier
SOL	Scrape-Off Layer
CRPP-EPFL	Centre de Recherches en Physique des Plasmas
	- Ecole Polytechnique Federale de Lausanne
Euratom	European Atomic Energy Community
UKAEA	United Kingdom Atomic Energy Authority
EFDA	European Fusion Development Agreement
MHD	MagnetoHydroDynamics
LIDAR	LIght Detection And Ranging
GIM	Gas Injection Module

Chapter 1

Thermonuclear Fusion

1.1 Introduction

For last 50 years, the scientists in fusion research aim to develop an electricity producing power plant based on the fusion reaction between the nuclei of the hydrogen isotopes: deuterium and tritium. The principle concept, how to achieve the thermonuclear fusion on Earth, is to confine a plasma (consisting of light atoms nuclei and their electrons) in a magnetic field configuration in such a way that the thermal plasma can reach conditions necessary to achieve a positive energy gain [1]. The Lorentz force makes charged particles move in helical orbits (Larmor orbits) about magnetic field lines. In a uniform magnetic field and in the absence of collisions or turbulence, the particles (better their guiding centers) remain tied to the field lines, but are free to move along them. The distance between the actual particle orbit and the magnetic field line is the Larmor radius r_L. A magnetic field is thus capable of restricting the particle motion perpendicular to the magnetic field but does not prevent particles from moving along the magnetic field. This effect serves as the basis for all magnetic confinement schemes. To avoid losses from the edges, it is necessary to close both edges together by creating a torus (Figure 1.1) [2]. The products of the D-T fusion reactions are helium nuclei (α-particles) and neutrons. The first, also bound to the magnetic field lines, are supposed to transfer their energy to the thermal plasma and thus sustain the fusion reaction. The latter, because they are not confined by the magnetic field, can leave the plasma directly and will be used to breed tritium from lithium and convert the fusion energy into heat. Many magnetic configurations for demonstration the nuclear fusion were invented through the last few decades. Nowadays, the most viable concepts are the tokamak and the stellarator, both invented in 1950s.

1.2 Tokamak

The tokamak is the most studied and most advanced fusion machine and is the most likely system to be converted into a energy producing reactor. A tokamak is a toroidal device in which the poloidal magnetic field is created by a toroidal current I_p flowing through the plasma and by poloidal field coils which are used also for plasma positioning and shaping. Figure 1.1 gives a schematic view of a tokamak. A strong toroidal magnetic field is generated by a toroidal field coil system. The toroidal current is induced by transformer effect. The plasma itself serves as a secondary winding of the transformer, while the primary is wound on an iron core. The combination of the toroidal and poloidal field gives rise to magnetic field lines which have a helical trajectory around the torus.

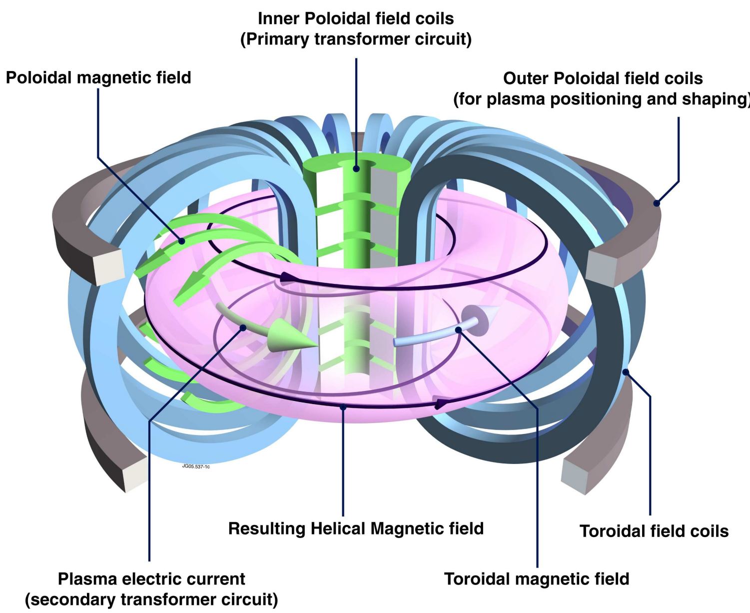


Figure 1.1: *Schematic view of a tokamak.*

There exists another magnetic configuration, called a stellarator, in which the magnetic field is provided completely by external toroidal as well as poloidal coils. The fact of not having an intense current flowing in the plasma is an advantage in the event of MHD instabilities and plasma disruption, but the drawback is the complexity of the necessary magnetic coils. This allows to build stellarator steady state. This may be seen on the Figure 1.2 of the German stellarator project W7X [3], where the coils are represented in blue and the plasma in orange colour [4].

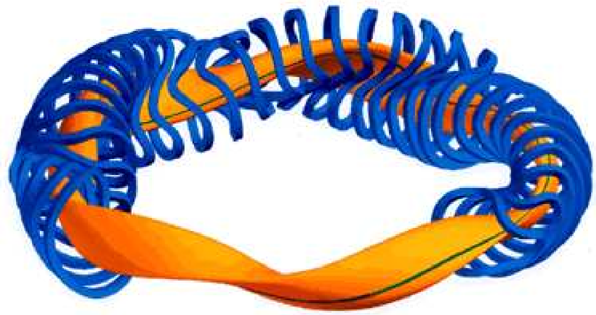


Figure 1.2: *Schematic view of strongly shaped coils system os a stellarator.*

1.3 JET tokamak

1.3.1 Introduction

The JET Joint Undertaking was established in June 1978 to construct and operate the Joint European Torus (JET). Of its time is was the largest single project within the European nuclear fusion programme. It was coordinated by Euratom (the European Atomic Energy Community), and the JET project went on to become the flagship of the Community Fusion Programme. It started operating in 1983 and was the first fusion facility in the world to achieve a significant production of controlled fusion power (nearly 2 MW) with a D-T experiment in 1991. After 1991, JET was enhanced by the installation of a divertor to handle higher levels of exhaust power. Deuterium experiments in the ITER geometry have made essential contributions to the ITER divertor design and provided key data on heating, confinement and fuel purity. This has contributed significantly to the definition of the size, heating requirements and operating conditions of ITER [5, 6]. During 1997 the JET operations included a three months campaign of highly successful experiments using a range of D-T fuel mixtures. During this campaign it was achieved three new 'world record':

- 22 MJ of fusion energy in one pulse
- 16 MW of peak fusion power
- a 65% ratio of fusion power produced to total input power

In Spring 1998 the fully remote handling installation of an ITER-specific divertor was successfully completed, demonstrating another technology vital for both ITER and a future fusion power station. Experimental work continued in 1999, in particular to characterise the new divertor configuration to control impurities and plasma density and to develop Internal Transport Barrier scenarios in preparation of ITER. The ownership of the JET

Facilities was transferred to the UK Atomic Energy Authority (UKAEA), and the overall implementation and co-ordination of further scientific exploitation is now carried out under EFDA, the European Fusion Development Agreement [7].

Nowadays, the main aim of the JET is to retain a focus on ITER priorities. It was planned to include the preparation of integrated operating scenarios for ITER and the study of phenomena important for the efficient exploitation of ITER such as the effects of toroidal field ripple, performance limiting instabilities, fast ion behaviour and transport in the core and edge. Moreover, it also included preparation for operation with an ITER-like wall, which will be installed in JET in 2009 [8].

1.3.2 Description of the JET tokamak

The toroidal component of the magnetic field on JET is generated by 32 large D-shaped coils with copper windings, which are equally spaced around the machine. The primary winding (inner poloidal field coils) of the transformer, used to induce the plasma current which generates the poloidal component, is situated at the center of the machine. Coupling between the primary winding and the toroidal plasma, acting as the single turn secondary, is provided by the massive eight limbed transformer core. Around the outside of the machine, but within the confines of the transformer limbs, is the set of six outer poloidal field coils used for positioning, shaping and stabilizing the position of the plasma inside the vessel (Figure 1.3). During operation large forces are produced due to interactions between the currents and magnetic fields. These forces are constrained by the mechanical structure, which encloses the central components of the machine. [5, 9]

The vacuum vessel is designed to hold a vacuum in which the pressure is less than one millionth of atmospheric pressure. This means that it has to carry the force of atmospheric pressure over the whole of its surface, 10 tonnes per square metre over an area of 200 square metres. To clean the plasma-facing surface from impurities, the vessel has to be baked up at 500°C.

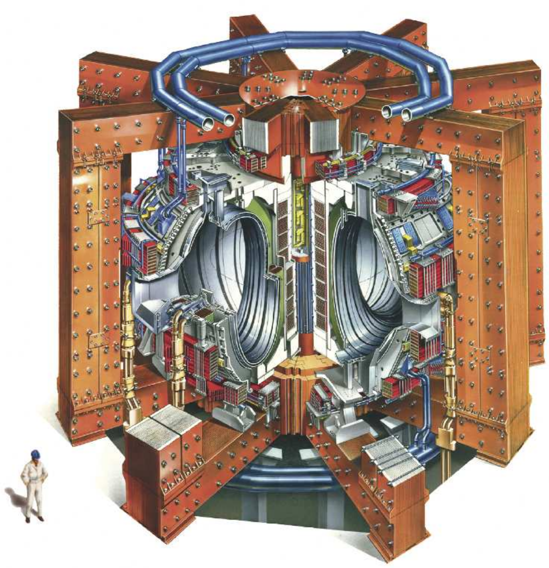
Running a JET pulse requires around 500MW of power, of which more than a half goes to the toroidal field coils. Around 100MW of power is needed to run the poloidal field system (ohmic heating and plasma shaping coils) and the rest (150MW) runs the additional heating sources (neutral beams and RF heating). The plasma itself is heated in three different ways: ohmic heating, neutral beam heating and radio-frequency heating:

- *Ohmic heating*: Currents up to 5MA are induced in the JET plasma - typically via the transformer. Only a few MW of heating power is provided in this way.

- *Neutral beam heating*: A widespread technique of the additional plasma heating is based on the injection of powerful beams of neutral atoms into ohmically preheated plasma. The beam atoms carry a large uni-directional kinetic energy. In the plasma, beam atoms loose electrons due to collisions, i.e. they get ionised and as a consequence are captured by the magnetic field of tokamak. In fusion experiments, the neutral beams are usually formed by atoms of hydrogen isotopes (hydrogen, deuterium or even tritium at JET). The energy of the beam must be sufficient to reach the plasma centre - if the beam atoms were too slow, they would get ionised immediately at the plasma edge. At JET the beam energy is 80 or 140 keV. The total power of beam heating at JET is as much as 23MW.

- *Radio-frequency heating*: Ion cyclotron resonant heating (ICRH) is routinely applied on JET. It is resonant with the second harmonic frequency of ion gyration of main plasma ions (deuterium) or with a base frequency of gyration of minority species (tritium, helium...). The available resonant frequencies at JET are in the range of 23-57 MHz. In total, the installed power of JET ICRH system is as much as 32 MW and in practice only part of this potential can be coupled for experiments.

The tokamak in principle operates in a pulse mode. Pulses can be reproduced at a maximum rate of about one every twenty minutes and each one can last for up to 60 seconds in duration, however the energy confinement time is about 1-2s. It is officially defined as the ratio of the thermal energy contained in the plasma and the power losses). The plasma is enclosed within the doughnut shaped vacuum vessel, which has a major radius of 2.96m and a D-shaped cross section of 4.2m by 2.5m. Figure 1.3 gives a drawing of the JET tokamak showing the general layout together with some basic parameters of the JET tokamak.



JET parameters

Major radius	2.96 m
Minor radius	2.10 m (vertical) 1.25 m (horizontal)
Flat-top pulse length	20s
Weigth of the iron core	2800 t
Toroidal field core power	380 MW
Toroidal magnetic field (on plasma axis)	3.45 T
Plasma current	3.2 MA (Circular plasma) 4.8 MA (D-Shape plasma)
Additional heating power	25 MW

Figure 1.3: *Basic JET parameters.*

Chapter 2

JET operating regimes

2.1 Introduction

The different confinement modes can be classified into L-mode, H-mode and internal transport barriers (ITBs). The L-mode is governed by a high level of turbulence, which is produced by auxiliary heating and enhances the radial transport perpendicular to the magnetic field lines [10]. The combination of sufficiently high heating power and a divertor configuration led to the discovery of the H-mode in the ASDEX tokamak [11]. The H-mode is characterized by a local reduction of the turbulent transport and is associated with an increase of the pressure gradient at the plasma edge. The radial pressure profiles of L- and H-mode are sketched in Figure 2.1. While in L-mode the gradients are limited over the whole plasma cross-section, the H-mode exhibits a region with large gradients at the edge, therefore also termed "edge transport barrier", but a similarly flat region in the plasma core. It is evident from the pressure profile shown in Figure 2.1 that in H-mode the product of pedestal pressure and plasma volume represents already large fractions of the plasma energy. Following the considerations concerning such an edge transport barrier, an internal transport barrier (ITB) may be regarded as a region with a steep pressure gradient inside the plasma, as illustrated in Figure 2.1. If the H-mode edge barrier is combined with an internal transport barrier, the contributions of course will add. [12]

2.2 H-mode

The "standard" H-mode is one the reference operating scenarios for ITER. Here, "standard" means that the extrapolation of the energy confinement is based on H-mode plasmas, originating from many tokamaks of different sizes, aspect ratios, plasma currents and magnetic fields (other parameters which are contained in the scaling laws are atomic mass

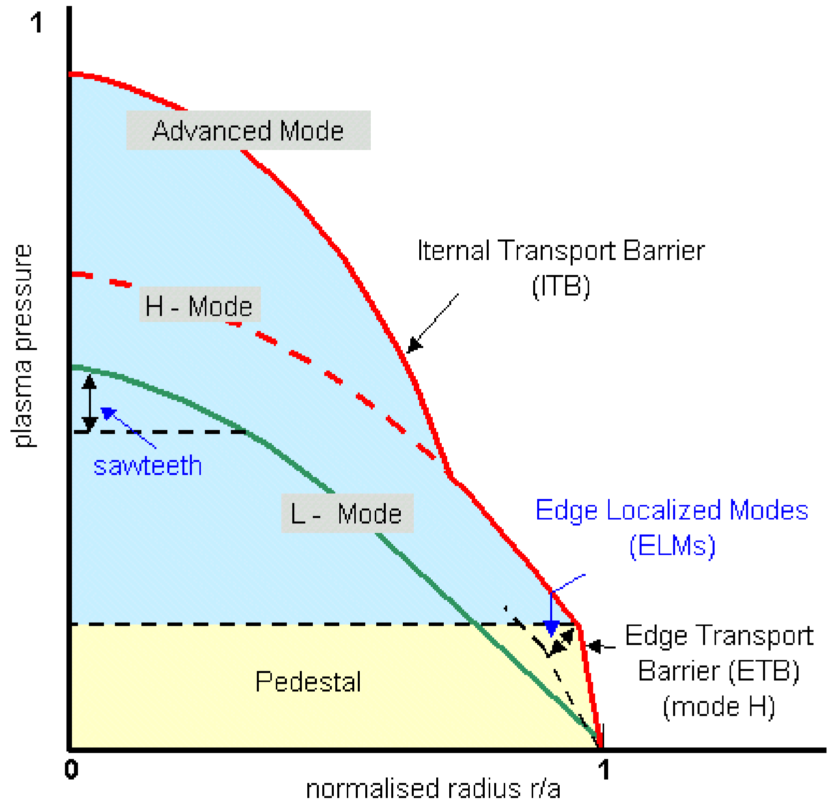


Figure 2.1: *Plasma pressure profile for L-mode, H-mode and regime with internal transport barrier.*

of the main plasma species, shaping parameters, such as elongation and triangularity, and plasma density [12]). The H-mode is one of the most robust and reactor compatible of the improved tokamak confinement regimes, combining good energy confinement with high β and in the presence of ELMs (Edge Localised Mode), with acceptable particle transport rates for the control of density, impurity and helium exhaust. There is a set of common features that are seen in all devices that obtain H-mode. The first to be identified was the formation of a transport barrier at the plasma edge, where the density and temperature gradients increased after the transition. The formation of this barrier is associated with a drop in the D_α radiation all around the plasma, indicating a significant decrease in the particle out flux.

There are several reasons why the H-mode has been chosen over the other improved confinement modes as the primary operating mode for ITER. [13]

- The H-mode with edge localized modes (ELMy H-mode) has been run for as long as 20 s on JET, with the duration limited only by power supply considerations.
- The H-mode exhibits flat density profiles in the plasma core, which are consistent

with reduced peaking of impurities and helium ash.

- The H-mode exhibits good confinement even in high density cases, where the electron and ion temperatures are in equilibrium; this is consistent with the alpha particle heating and high density operation that will be needed for ITER.

- The H-mode requires no special current profile control for long pulse operation, unlike the operating modes with negative central magnetic shear.

The H-mode is reached above a certain threshold heating power, P_{thr} , which depends on plasma conditions and machine size and it is essential to predict what value is needed for ITER. The threshold dependence on plasma configuration and parameters, studied in single devices during the past years can be summarized as follows: the threshold power is about 2 times lower for the single null (SN) configuration with the ion ∇B drift towards the X-point, than for the opposite direction or double null (DN) configuration; the threshold is about 2 times lower in deuterium than in hydrogen; reduction of neutral density and impurities by appropriate wall conditioning and good divertor retention is favorable for achieving low threshold powers. The studies in single devices also show a rather clear linear dependence of P_{thr} on $\tilde{n}_e$ and B_T.
The H-mode exhibits global energy confinement values about a factor of two better than L-mode. It is partially caused due to the formation of the edge transport barrier. Another part of this improvement is a consequence of reduction in local transport throughout the plasma after the L-H transition.

2.3 Internal transport barrier

Besides the ELMy H-mode, several other improved confinement modes have already been realized. One of the most promising ones is the negative (or reversed) shear configuration, which involves an internal transport barrier (ITB). Regime with ITB belongs to the group called "Advanced Tokamak" scenarios, which are desirable for high performance operation because they exhibit higher confinement with respect to the reference ELMy H-mode. This would allow, for the same performance level, a reduction of the operating plasma current, thereby opening the way towards steady-state operation with full non-inductive current drive [13].
Internal transport barriers arise from an influence of the magnetic shear $s = (r/q)dq/dr$ on the growth of micro-instabilities (where r and q means minor radius and safety factor, respectively). In conventional tokamak plasmas without ITB, q increases monotonically from the center towards the edge, i.e. the magnetic shear is positive across the whole plasma radius. Only in conjunction with ITB, due to the large off-axis bootstrap current, does the q profile become non-monotonic (Figure 2.2). Therefore, internal transport

barriers are produced by modification of the current profile using external heating and current drive effects. This process is called plasma tailoring. In large tokamaks, tailoring of the current profile towards flat and/or hollow profiles can only be carried out during a low-performance phase, which is referred to as the "prelude". The reason is the strong dependence of the plasma current resistive diffusion time t_R upon the electron temperature and plasma size, leading to unacceptably long-term evolution of the current profile if this is not tailored in a low-performance phase [14].

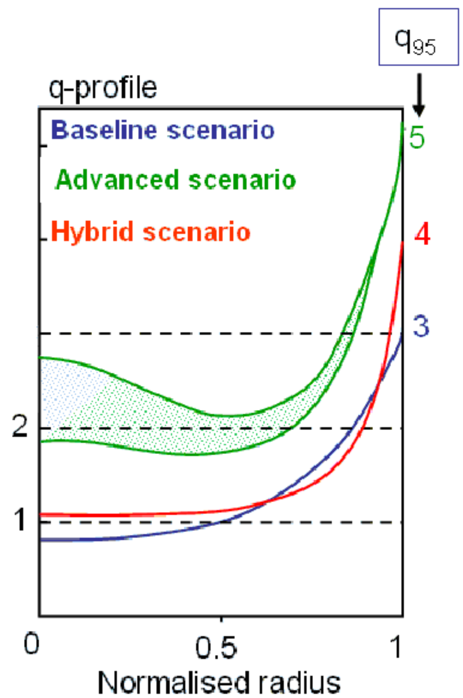


Figure 2.2: *q-profile for different scenarios.*

As mentioned above, internal transport barriers in tokamaks are produced by modifications of the current profile. The most common technique how to generate low or negative central magnetic shear is to apply auxiliary heating power during the initial current ramp-up phase of the tokamak discharge and thus to some extent "freeze" the initial skin current profile. Depending on the different heating methods and their different ion and electron heating partition, ITBs of the ions, electrons or both are generated [15].

In JET, ITBs have been combined with L- and H-mode edge conditions. The latter include edge transport barriers with both Type-III and Type-I ELMs and have been termed double transport barriers or double barrier (DB) mode [16]. This combination of both barriers is advantageous since it can lead to:

- higher fusion performance with better confinement properties at high beta;

- broader pressure profile with improved MHD stability against destabilizing n=1 modes providing a route towards high β_N operation ;
- a higher fraction of bootstrap current thanks to edge pressure gradient by naturally increasing the off-axis non-inductive current driven outside the ITB; off-axis current outside the ITB is indeed required to maintain broad current density profiles.

Chapter 3

Results

3.1 Introduction

As was mentioned in previous chapter, the reference confinement scenario used for extrapolation to burning fusion plasma is based on the ELMy H-mode which exhibits a transport barrier at the plasma edge. Often confinement modes with internal transport barriers (ITB) are also referred to as advanced tokamak scenarios. The development of discharge scenarios with weak and strong internal transport barriers (ITB) is one of the primary goals of research at JET (undertaken by the Scenario 2, or S2 Task Force). Such scenarios are the key to non-inductive, steady-state operation in future devices and are being developed as ITER candidate scenarios. Although similar experiments are performed on many other tokamaks, there has never been a systematic attempt to compare the edge conditions in these advanced scenarios with those of the ELMy (Edge Localised Mode) H-mode plasma (studied by the Scenario 1, or S1 Task Force at JET) which will be the baseline Q_{DT}=10 scenario on ITER and which has been extensively studied with respect to edge physics. Such comparative studies take on even greater importance now with the imminent arrival of the new all-metal, ITER-like first wall planned for JET. Two types of study are possible: the detailed investigation of individual discharges (from S1 and S2), including numerical fluid code modelling and a statistical approach in which a large number of edge and core plasma quantities are compared across a large discharge database to assess the level of similarity in the edge plasma of each type of scenario. I have begun by following the latter, statistical approach, compiling a database of more than 80 relevant quantities which characterize plasma geometry, basic plasma parameters including profiles of electron temperature T_e, density n_e, ion temperature T_i, and all edge and scrape-off layer (SOL) properties which are currently measured on JET. This has been done using a database creation utility, MDB, written at CRPP-EPFL Lausanne, Switzerland and running under MATLAB. Each quantity is represented in this database by its average value in a user specified time window at any number of intervals in any given discharge.

3.2 MDB database

At JET, signals from all diagnostic systems are digitised and stored in a central database. The sampling frequencies depend on the requirements and abilities of individual diagnostics and vary from a few measurements per second up to about one million per second. In total, more than one billion readings of diagnostic data are recorded per JET pulse, each reading with 12 or 16 bits. In other words, every JET pulse produces almost 2 GBytes of raw diagnostics data, so that as much as 50 GBytes are stored daily. Most of the data need further processing - this is done automatically where possible by dedicated computer codes, but in many cases human intervention and/or data validation is required. The processed data are stored separately from raw data. All data are accessible to all scientists on the JET site and, moreover, any scientist from any EFDA Association can work with the data from his home institute via the technique of Remote Access. Many Associations and Contractors continue to develop new diagnostics for JET or upgrade the present ones. At the same time, JET serves as a unique test bed for the development of diagnostics for the future fusion reactor machine, the ITER.

To compare both main JET operating regimes (standard ELMy H-mode and advanced regime with ITB), I have choosen a statistical approach in which a large number of edge and core plasma quantities are compared across a large JET discharge database to assess the level of similarity of both scenarios. Database contains more than 80 relevant quantities which characterise plasma geometry, basic plasma parameters including profiles of electron temperature T_e, density n_e, ion temperature T_i, and all edge and scrape-off layer (SOL) properties which are currently measured on JET. This has been done using a database creation utility, MDB, written at CRPP-EPFL Lausanne, Switzerland and running under MATLAB.

A database built with TCV MDB tool is most commonly a set of very interesting quantities (variables) taken for very interesting shots at very interesting times (samples). It can be seen as a table in which each column represents a variables and each row a sample. The samples are defined by one or more key variables, where SHOT and TIME is the example. Their list is given in the manual entry file *man* file, one of the three files constituting a database. The list of variables together with the definition of their properties is given in the variable description file *mdb* file. These properties specify where the value of the variable for each sample comes from (in this case it is a manual entry entered in the man file) and how the sample time is selected on a time signal. The variables themselves are stored as MATLAB variables with the same name in a MAT-file forming the data file. For the data processing I used the mdb setting: '*processing=poly0*' which calculates a mean value of the signal samples inside the time window of width (-0.25 + t, t + 0.25), where t denotes the specified time [8].

3.3 Set-up and evaluation of the database

The database used for comparison both standard ELMy H-mode (a reference ITER scenario) and advanced tokamak scenario with internal transport barrier contains 66 JET discharges. The discharges were performed during different campaigns and can be divided into four groups:

- ***S*1 regime**[1] (23 discharges): 70236-70239, 70241-70243, 70245-70247, 70540-70542, 70544-70553.
  S1 shots were performed during two experimental sessions "Divertor geometry studies - ITER like". The first are characterized by high plasma current of 2.5 MA and total heating power of about 16 MW. The latter session is characterized by low NBI and ICRH heating of total power ~9 MW.

- ***S*2 regime**[2] (10 discharges): 69987, 70274, 70275, 70292, 70300, 70333, 70355, 70358, 70361, 70362.
  These shots are characterized by no additional gas seeding.

- ***S*2 regime with Neon seeding** (29 discharges): 69974, 69976-69982, 69984, 70276, 70281-70283, 70285-70287, 70289, 70291, 70301, 70334-70341, 70359, 70360.

- ***S*2 regime with Nitrogen seeding** (4 discharges): 70293-70295, 70297
  These shots were performed during two experimental session S2 Impurity seeding in AT scenarios characterized by B_T = 3.1 T and I_p = 2 MA. The results presented in this paper are related to nitrogen seeding in type-III ELMy H-modes and this nitrogen seeded type-III ELMy regime leads to a partially detached H-mode.

The JET MK2GB divertor with new load bearing septum replacement plate (LBSRP) enables high triangularity plasmas ($\delta \sim 0.45$) with a larger volume than was previously possible. Figure 3.1 shows the high triangularity configuration that was used for study of impurities influence on plasma performance. These experiment run at high field and current of 3.1 T and 2 MA with q_{95} ~5. The outer strike point is placed onto the LBSRP and the inner strike point is positioned on the upper vertical tile. This configuration was set up in view of the future ITER like wall project in JET. The LBSRP will be the only

[1] S1 - Standard ELMy H-mode studied by S1 Task Force at JET
[2] S2 - regime with ITB studied by S2 Task Force at JET

divertor tile made out of solid tungsten and will have a higher power handling capability than the other divertor tiles. The inner vertical divertor tile, like all the rest, will be made from carbon with a thin tungsten coating. The power load obtained in this configuration is asymmetric with the higher load on the outer divertor.

It was not the aim of these S2 discharges with additional gas seeding to produce high performance ITB plasmas. For this reason the heating scheme applied was not, as is usual for ITB experiments, adjusted to get a current profile that enables maximum confinement. The heating scheme was fixed to a 'best guess' in order to be close to optimised shear, i.e. to obtain a rather flat current profile. As a consequence no strong ITBs were obtained during these experiments.

3.3.1 Impurities

The gas recipes used in these experiments were carried out in feed forward mode, i.e. a pre-programmed gas waveform was applied to the plasma. Impurity seeding blips of 0.25s intermittency are applied as the injection modules would not open at the required lowest averaged level of $5 \cdot 10^{20}$ electrons/s of neon seeding. The gas recipes used in these experiments can be categorised as follows with increasing level of radiated power:

- No-gas reference discharges do not have additional D_2 or impurity injection. They are used to probe the effect of machine conditioning on plasma behaviour, as experiments were carried out spread over a 3 months period.

- Deuterium only discharges have a base level of deuterium gas seeding of $3.5 \cdot 10^{21}$ electrons/s and feature higher frequency ELMs.

- Impurity seeded discharges have a varying level of either Ne ($5 \cdot 10^{20} - 2 \cdot 10^{21}$ electrons/s) or N_2 ($5 \cdot 10^{21}$ - $2 \cdot 10^{22}$ electrons/s) seeding. Because of lack of experimental time much less discharges were seeded with N_2 than with Ne.

- Impurity and deuterium discharges have a combination of either Ne ($1 \cdot 10^{21}$ electrons/s) or nitrogen ($1.5 \cdot 10^{22}$ electrons/s) and deuterium ($1.5 \cdot 10^{22}$ electrons/s) seeding.

It was found experimentally that in order to obtain the same level of radiated power from nitrogen injection an order of magnitude more gas injection was required than in the neon seeded case . Nitrogen was seeded into the divertor private region, whereas for neon a scan was performed for three different injection locations; the divertor private region (GIM11)[3], from the top of the main chamber (GIM5) and from the bottom of the main chamber (GIM9) (Figure 3.1).

[3]GIM = Gas Injection Module

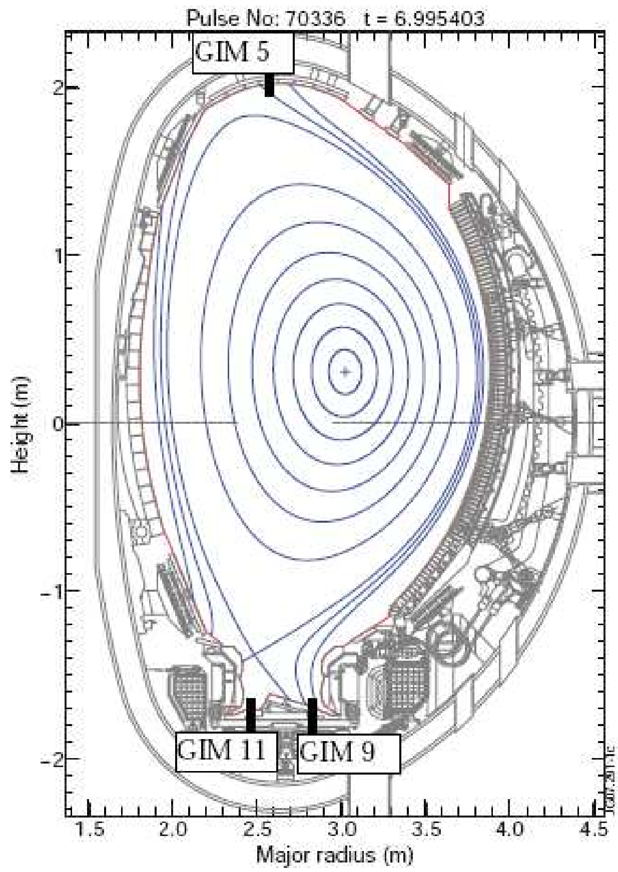


Figure 3.1: *High triangularity configuration. The outer strikepoint is placed on the load bearing septum replacements plate, which will have the highest power handling in the ITER like wall. The gas injection locations for Ne are indicated as GIM5, GIM9 and GIM11.* N_2 *was injected from GIM11 only.*

3.3.2 Temperature and density profile

The temperature and density profiles are measured on JET by the LIDAR system, which is based on Thomson scattering effect. The density and temperature profiles for both S1 and S2 regimes are shown in Figure 3.2. As it can be seen, the density is characterised by flat profiles in both regimes, unlike the temperature. The S1 plasmas have almost twice higher density. Temperature exhibits more peaked profiles in S2, reaching almost twice higher central values compared to S1, as expected. Moreover, neon and nitrogen impurities have no deleterious effect on plasma temperature and density in plasma core.

3.3.3 Radiation pattern

One of the most severe problems for fusion reactors is the power load on the divertor taret plates. Technically, only steady state power loads of about 10 MW m^{-2} are acceptable. In order to reduce the power load in the divertor to this value, radiation cooling by seeding of

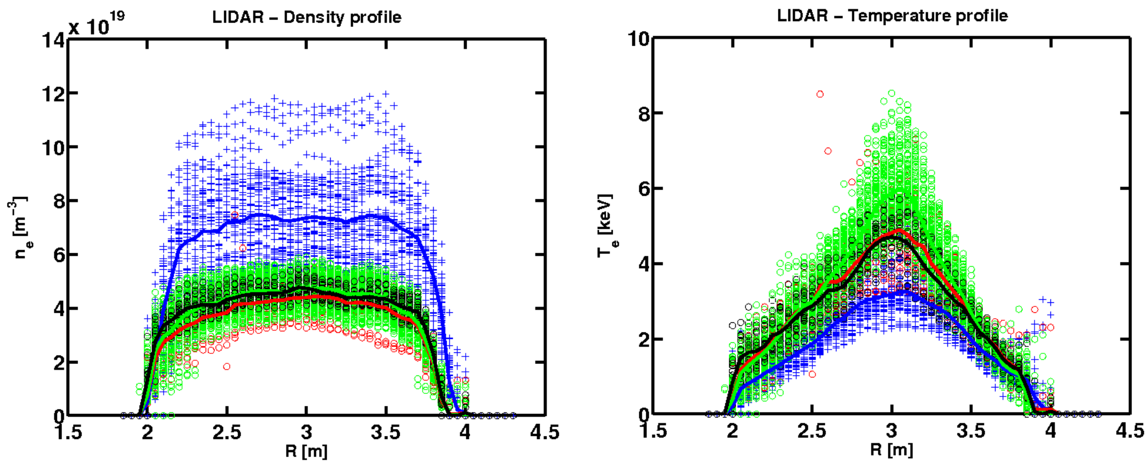


Figure 3.2: *LIDAR density (left) and temperature (right) profiles measurement.*

impurities might be necessary. For the ITER reference scenario a radiative power fraction of ≈75% is reguired [17]. Currently, in most fusion devices, carbon, which is the most commonly used divertor target and wall material, determines the radiative power fraction. However, looking to future devices like ITER with tungsten divertor and metallic walls (Be or W), radiation due to intrinsic impurities will be minimal and seeding of additional impurities becomes essential.

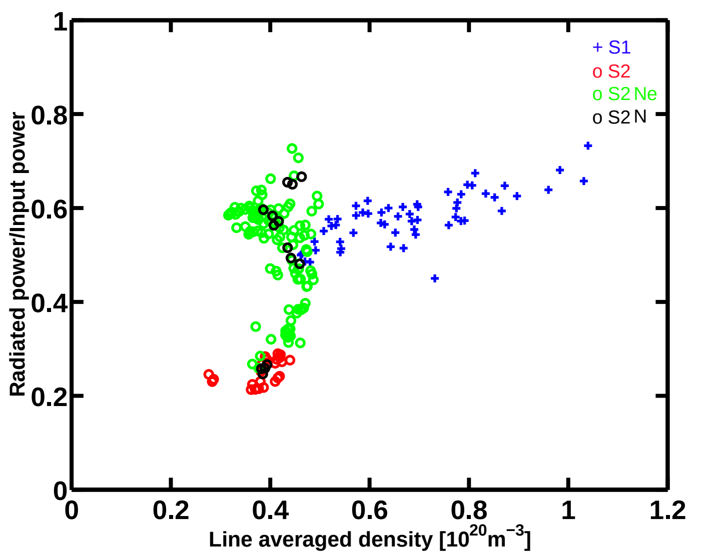


Figure 3.3: *Radiative power fraction.*

Pure hydrogen plasma emits electromagnetic radiation. Microscopically this is caused due to the acceleration of the charged particles. The electrons are accelerated in two ways. Firstly, they are accelerated by collisions and then the resulting radiation is called bremsstrahlung. Secondly they are subject to the acceleration of their cyclotron motion.

The presence of impurities in the plasma produces energy losses through line radiation. A measurement of the total radiation emitted from the plasma is important for evaluation of the energy balance. Power radiated fraction is a important characteristic especially in point of view of ITER. It is a ratio of total radiated power from plasma and the total input power into plasma. The radiated power fraction dependence on density is shown in Figure 3.3. As it can be seen the radiated power fraction in S1 regime increases linearly with density and reaches up to 70%, i.e. 70% of total input power is lost via radiation. On the other hand, the radiated power fraction for S2 regime is only about 25%. But, for impurities seeded discharges the radiative power fraction reaches up to 70%. There is too narrow density range in S2 to asses density dependence of radiated power fraction. This is caused by fact that the JET tokamak has limited amount of additional heating power (regime with ITB requires more input power than standard EMLy H-mode).

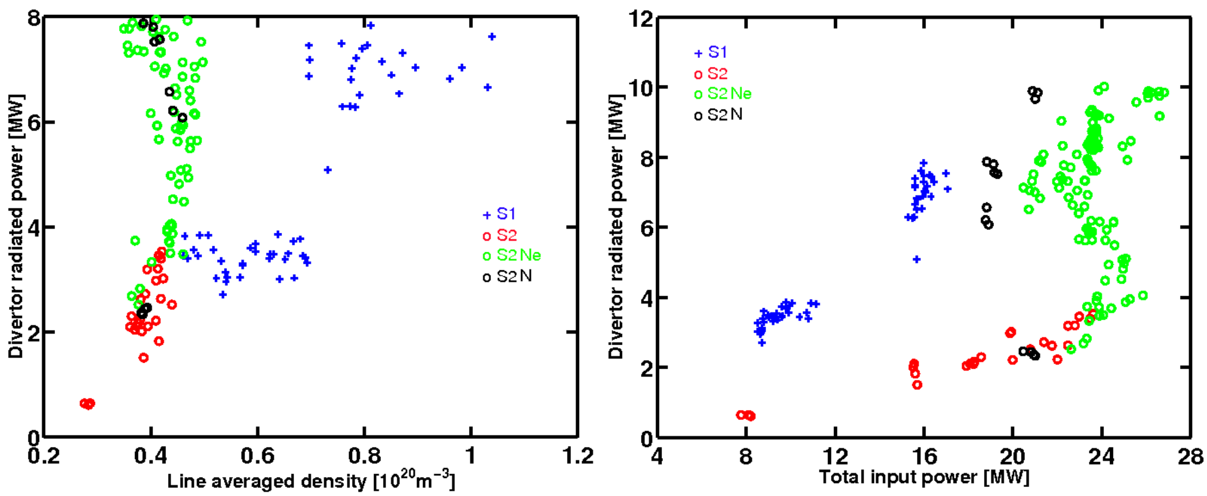


Figure 3.4: *Divertor radiated power as a function of density (left) and total input power (right).*

The total radiated power found in divertor region as a function of density and of total input power is shown in Figure 3.4 (left panel). The S2 plasmas have lower density and hotter edges. And as it shown the S2 plasmas radiate less than the colder edges in S1 plasmas. The two S1 'clouds' correspond to the two groups of discharges in database which differ in amount of additional heating power. The divertor radiated power increases with density for S2 discharges. But for S1 shots, taking into account two distinct levels of input power, the divertor radiated power remains constant. Lower ability to radiate out the energy from the divertor in S2 regime without impurity seeding is apparent also in Figure 3.4 (right panel). Moreover, it can be nicely seen the transition of S2 regime into 'S1 like' in dependence on amount of injected impurities. Here, the dependence of divertor radiated power versus total input power is plotted.

Radiative power fraction in dependence on total gas seeded amount is shown in Figure 3.5. It can be clearly seen that the radiated power fraction as a function of neon fuelling amount remains constant of about 60%. The minimum neon fuelling amount for this value is $\sim 4 \cdot 10^{21}$ electrons. The situation is almost the same for nitrogen seeded discharges. The minimum nitrogen fuelling amount necessary for reaching 60% of radiated power fraction is $\sim 7 \cdot 10^{22}$ electrons.

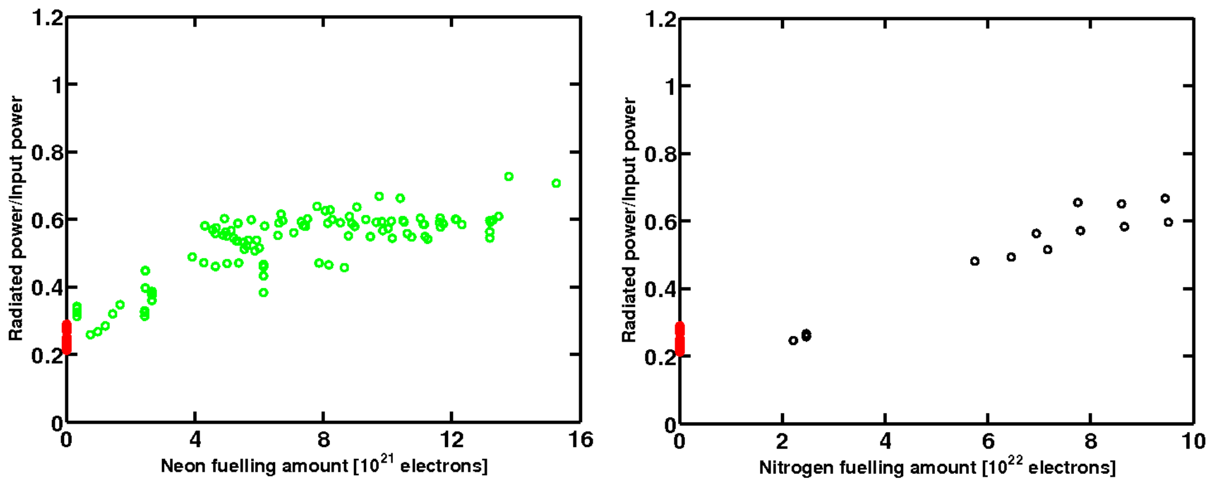


Figure 3.5: *Radiative power fraction in dependence on total gas seeded amount.*

3.3.4 Energy balance

Another very important quantity is the energy balance characterised by dependance of total energy arriving on the divertor tiles on total "available" energy (input - radiated energy), shown in Figure 3.6. The total available energy was calculated by integration of a bolometric signals across the whole discharge. The energy balance is not perfect for both regimes. The situation is worst in S1 regime. Slightly more energy arrives to divertor tiles in S1 regime than there seems to be available. On the other hand, it seems that some amount of energy in S2 regime is lost. This can be explained by fact that some the magnetic field lines are opened also in the upper part of the vacuum vessel (see Figure 3.1).

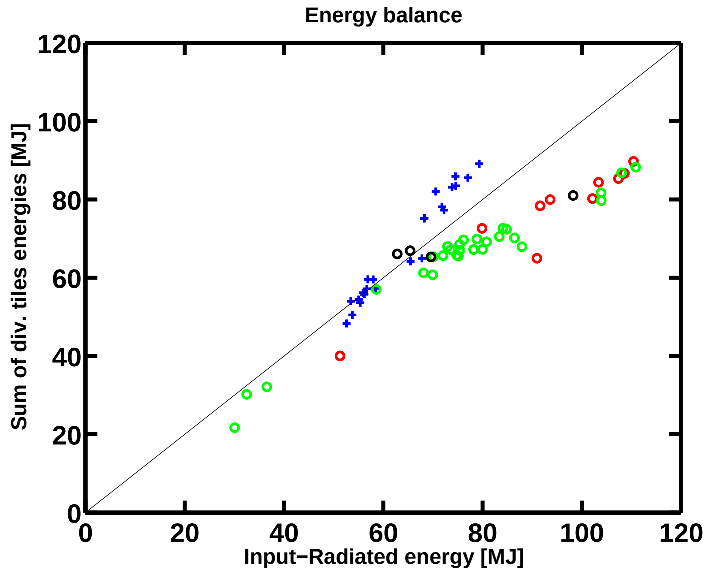


Figure 3.6: *Energy balance.*

Summary

This report gives an overview of comparison of the basic plasma parameters in various regimes of the JET tokamak. First regime is called the ELMing H-mode (S1), which exhibits a transport barrier at the plasma edge and periodic relaxations of edge pressure profile (ELMs). The second one is characterised by Internal Transport Barrier (ITB) in plasma core (S2) and is a key to non-inductive, steady-state operation of future devices.

In order to compare the both operating regimes a set of 23 discharges performed in S1 regime, 10 discharges to represent S2 regime without additional impurity seeding, 29 discharges contains neon seeding and 4 discharges characterised by nitrogen seeding were selected. More than 80 relevant quantities including plasma geometry, basic plasma parameters, profiles of electron temperature T_e, density n_e, ion temperature T_i, and all edge and scrape-off layer (SOL) properties which are currently measured on JET, were used to characterize these two regimes. The proper time windows, where the main plasma parameters are stationary for each discharge, were selected. For each time window, the mean values of the evaluated quantities were computed and stored using the MDB database creation tool.

The density is characterised by flat profiles in both regimes. S1 regime operates with about two times higher density compared to S2. Temperature exhibits more peaked profiles in S2, reaching almost twice as high central values compared to S1.

Radiated power fraction in S1 regime increases linearly with density and reaches up to 70%. On the contrary, the radiated power fraction for S2 regime without seeding of impurities is only about 25%. Discharges with additional impurities seeding reach the radiative power fraction of about 60%. This finding is extremely important with respect to ITER as advanced ITER regimes with only 25% of radiated power fraction would mean unacceptably high heat loads to the divertor structure. Therefore, using neon or nitrogen as a cooling mechanism is necessary. For both gases exists a minimal value for reaching a high radiative power fraction.

The energy balance is not perfect for both regimes. The situation is worst in S1 regime where it seems to be more energy arriving to divertor tiles than there seems to be available. On the other hand, some amount of energy in S2 regime is lost.

In conclusion, it was found that for ITER advanced regimes, additional extrinsic impurity seeding or further increase of density in divertor is needed in order to increase radiation power fraction and as a result to keep the power loads on divertor targets sufficiently low.

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