# The Physics Base for ITER and DEMO

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## 1. Introduction

Nuclear Fusion research using magnetic confinement aims at confining a plasma consisting of hydrogen isotopes, hot and dense enough such that fusion processes between the nuclei lead to a net energy gain. Magnetic confinement is realised in toroidal geometry where helical magnetic field lines form a set of nested magnetic surfaces as shown in Fig. 1. The helical field lines can either be formed by a combination of planar external coils and a current in the plasma (‘tokamak’, left part of Fig. 1) or completely by more complicated coil set (‘stellarator’, right part of Fig. 1). In general, the charged plasma particles follow the magnetic field lines and hence gradients in density and temperature can be sustained between the magnetic surfaces. If successful, magnetically confined plasmas could result in a sustainable energy source with favourable environmental properties.

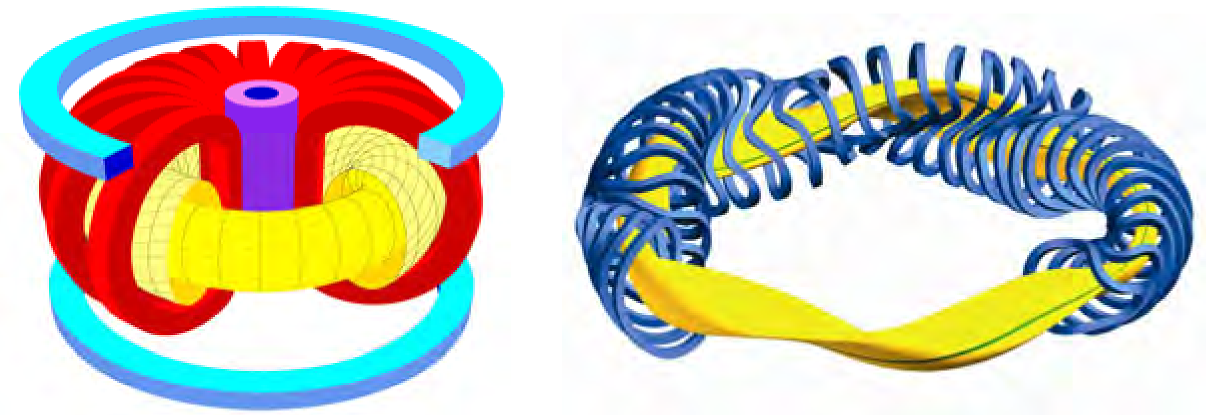


*Fig. 1: Toroidal magnetic geometry for confining a fusion plasma. Left: axisymmetric ‘tokamak’ device with planar coils (red) and central transformer (violet) to induce a toroidal plasma current. Right: ’Stellarator’ device with 3-dimensional coils – this configuration does not need an intrinsic plasma current for confinement.*

To reach this goal, a plasma consisting of a 50:50 mix of Deuterium and Hydrogen has to be heated to temperatures of the order of $k_BT \approx 20$ keV and the number density should be of the order $n = 10^{20}$ m$^{-3}$. Then, fusion reactions will occur according to

$$D+T \rightarrow {}^4He + n + 17.6\ \text{MeV} \tag{1}$$

where the released energy is split according to conservation of energy and momentum into 1/5 for the $\alpha$-particle and 4/5 for the neutron. The latter will escape the magnetic confinement and heat the wall of the reactor; the $\alpha$-particle however is also confined in the plasma and will, via collisions, heat the plasma. This self-heating by the fusion reactions is supposed to cover the majority of the energy loss from the hot plasma by radiation, convection and conduction. This postulate leads to the so-called Lawson Criterion (see e.g. [1]) which demands that the product of density and energy confinement time exceeds $n\tau_E \approx 2 \times 10^{20}$ m$^{-3}$ s at the above mentioned temperature of roughly 20 keV. Here, $\tau_E$ is the energy confinement time which characterises the quality of the heat insulation of the magnetic confinement device. In stationary conditions, it can be expressed as $\tau_E = W/P$ where $W$ is the thermal energy of the plasma and $P$ the heating power needed to sustain it.

Progress in magnetic confinement devices all around the world has led to the creation of plasma that are dense and hot enough to roughly fulfil the Lawson criterion, but at present, the energy confinement time is not high enough to obtain a sufficient fraction of self-heating. Typically, energy confinement times of several 100 ms can be realised on the largest present day experiments while a reactor will require $\tau_E$ to be of the order of several seconds. However, the experimental results together with the theoretical understanding obtained over the last decade have given confidence that a machine can be build that will fulfil the requirements for dominant self-heating. This device, ITER [2], is presently under construction in a multi-national effort at Cadarache, France, and supposed to go into operation by the end of the decade. If successful, it should be followed by a device that generates net electrical power and is the direct step to a commercial reactor. This step is called DEMO [3], and its physics requirements will be somewhat more stringent than those for ITER. It is the aim of this paper to outline the physics requirements for ITER and DEMO and to compare them with the present status of research, hence highlighting areas which will need special attention in the coming years. The discussion is focused on the plasma physics requirements; details about the technology aspects can be found elsewhere.

## 2. Main topics in Fusion Plasma Physics

In the previous section, we have already mentioned the conditions that need to be achieved for a successful fusion reactor based on the magnetic confinement principle. Looking in more detail, one can separate four basic areas of fusion plasma physics that need attention for the system to succeed:

*Transport* of heat and particles determines the amount of heating needed to obtain the necessary temperatures. First estimates of the cross-field transport based on binary collisions, carried out in the 1950-1960s, gave quite favourable predictions, but it was found experimentally that the heat loss is much bigger than initially thought because the strong gradients drive turbulent transport that can exceeds by orders of magnitude the losses predicted by binary collision theory [4]. Since the quantitative prediction of turbulent transport is quite difficult, empirical scalings using experimental data from magnetic confinement devices were assembled to predict the energy confinement time $\tau_E$ in future devices. A widely used scaling is the so-called ITER98(p,y) scaling [5] that predicts $\tau_E$ based on a power law dependence of the engineering parameters used to control the confinement device. In particular, this scaling has been used to evaluate the ITER design point such that it would enable ITER to reach its goals if heat losses follow this scaling. It is hence common to quote experimentally reached data points in relation to the prediction by this scaling through the quantity H (also called ‘H-factor’)

$$H = \tau_{E,exp}/\tau_{E,ITER98(p,y)}\,. \qquad (2)$$

In the following sections, we will adopt this approach.

*Stability* of the magnetically confined plasma is a major issue. In the optimum temperature range around 20 keV, the fusion reaction rate $R_{DT} = n_D n_T\langle\sigma v\rangle$ can be approximately expressed as $R_{DT} \sim n^2T^2$ [6] and since the kinetic pressure of an ideal plasma is $p = n\,k_BT$, the fusion power density will scale roughly as $P_{fus} \sim p^2$. Hence, raising the plasma pressure will lead to larger fusion power. On the other hand, the kinetic pressure also is a source for magnetohydrodynamic (MHD) instabilities which set a limit to the maximum achievable pressure. An important dimensionless number characterising the instability drive is the ratio of kinetic pressure to magnetic pressure $B^2/(2\mu_0)$:

$$\beta = 2\mu_0 \langle p \rangle / B^2 \qquad (3)$$

which can also be viewed as a number describing the efficiency of the use of the magnetic field since the strength of magnetic field necessary for confinement largely determines the cost of the device while as stated above, $p$ determines the fusion power output. Typical $\beta$-values needed for economic reactors will be of the order of several %. These have been reached in present day devices, but coming close to these values has also revealed that MHD instabilities can occur in that region. Hence, understanding the physics of these instabilities to avoid or control them is an important line of research. In particular, for the tokamak configuration (see Fig. 1, left), ideal MHD theory predicts that the maximum value of $\beta$ should be related to the toroidal plasma current $I_p$, the minor radius of the torus $a$ and the magnetic field strength $B$ by $\beta_{max} \sim I_p/(aB)$. This so-called Troyon-limit [7] has been confirmed experimentally and hence the experimental values are often quoted in relation to this limit as $\beta_N$, the 'normalised $\beta$':

$$\beta_N = \beta/(I_p/(aB)). \qquad (4)$$

In the following sections, we will follow this approach.

Concerning the *self-heating by $\alpha$-particles*, the importance of the internal heating is characterised by the fraction of $\alpha$-heating power $P_\alpha$ to the total power $P_{tot} = P_\alpha + P_{AUX}$ where $P_{AUX}$ is the auxiliary power used to heat the plasma by external means. Since the $\alpha$-particles carry 1/5 of the energy released in the fusion reaction (see also Section 1), this can be directly related to $Q$, the ratio of fusion power $P_{fus}$ to auxiliary heating power by

$$Q = P_{fus}/P_{AUX} \Rightarrow P_\alpha/P_{tot} = Q/(Q+5). \qquad (5)$$

In the following, we will use $Q$ to characterise the amount of $\alpha$-power in a discharge. For $Q \rightarrow \infty$, the plasma approaches the ignited state and control of the heating power is no longer directly possible by adjusting the external heating power. Hence, burn control will become an important subject under these conditions. In addition, a large fraction of suprathermal $\alpha$-particles represents a source of free energy that can drive instabilities which might redistribute or even eject the $\alpha$-particles, reducing the efficiency of the self-heating process.

Finally, the *exhaust* problem mainly comes from the need to remove the power that leaves the plasma along magnetic field lines to the first wall since the interaction region, typically realised in the form of a magnetic divertor, is usually very narrow. The main technical limitation is set by the heat flux that can be removed in steady state from a divertor component. A typical value is of the order of 10 $MW/m^2$. Since the width of the interaction region does not scale strongly with machine size, a criterion commonly used is

$$P_{tot}/R_0 = (P_{AUX}+P_\alpha)/R_0 \, , \qquad (6)$$

with $R_0$ being the major radius of the torus [8]. A common approach to limit the heat flux to the divertor components at given $P_{tot}$ is to add impurities in the plasma boundary that will convert some of the power into electromagnetic radiation that is distributed evenly over the whole inner wall. Care has to be taken that this does not dilute the plasma or create too much radiation loss in the plasma centre where the heating power is needed to balance the losses from conduction and convection.

The criteria (2), (4), (5) and (6) can hence be used to characterise the physics requirements for future reactor-grade devices and to compare them with the parameters reached in present day devices. This will be done in the following two sections.

## 3. Requirements for ITER and DEMO

In this section we discuss the requirements for ITER and DEMO concerning the parameters introduced in the previous section. As pointed out before, to reach significant α-heating, we need $Q >> 1$. An analysis of the confinement scaling with machine size $R_0$, confinement quality $H$ and normalised pressure $\beta_N$ shows that $Q$ depends mainly on $R_0$ and $H$ and only weakly on $\beta_N$ in the form [9]

$$Q \propto \frac{H^{3.23} \beta_N^{\;0.1} R_0^{\;2.7}}{const - H^{3.23} \beta_N^{\;0.1} R_0^{\;2.7}} \quad . \tag{7}$$

Hence fixing $H=1$ (i.e the scaling used for ITER), largely determines the size of the machine, which for the ITER choice of $Q$ = 10 (meaning $P_\alpha = 2P_{AUX}$) yields $R_0$ = 6.2 m. Due to the strong nonlinearity of (7) close to ignition, an ignited device like DEMO will not be much larger, and for the same $H$ = 1 assumption, turns out to be of the size $R_0 \approx$ 7.5 m. This can be seen in the left part of Fig. 2 which shows the dependence of $Q$ on $R_0$.

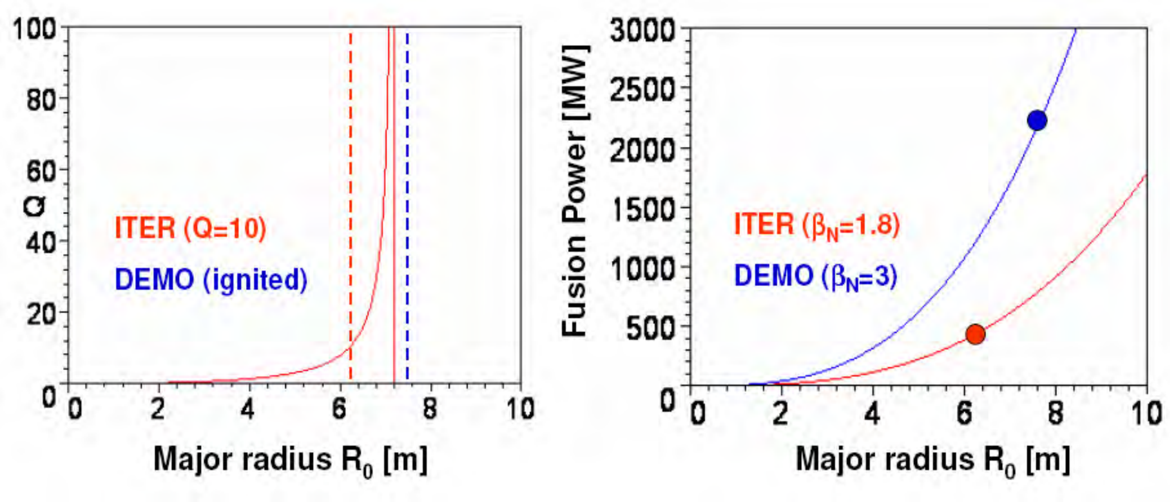



*Fig. 2: Dependence of Q (left, for H = 1) and $P_{fus}$ (right, for two different values of $\beta_N$) on major radius.*

The right part of Fig. 2 shows the dependence of fusion power on major radius, which, different from $Q$, does not depend on $H$, but strongly on $\beta_N$:

$$P_{fus} \sim \beta_N^2 \, R_0^3 \; . \tag{8}$$

For ITER, a conservative value of $\beta_N$ = 1.8 has been chosen to be well below present ideal stability limits. The choice of a higher $\beta_N$ for DEMO can be motivated by the fact that fusion power plants will carry a burden of a relatively large power needed for the balance of plant due to the need for cooling the superconducting coils and hence, in order to have acceptable recirculating power will necessarily be large units. However, due to the $R_0^3$ dependence of $P_{fus}$, an economic power plant will largely increase the requirements for exhaust characterised by $P/R_0$ when compared to ITER.

For a DEMO based on the tokamak concept, the values of $H$ and $\beta_N$ are also important for the pulse length. As introduced above 1, a tokamak needs a strong toroidal plasma current (typically in the MA range) to establish the magnetic geometry needed to confine the plasma. In present day devices, this current is mainly driven by a central transformer coil (violet solenoid in Fig. 1), which means an intrinsically pulsed device since primary current in the transformer coil has to be ramped down continuously to compensate the resistive loss of magnetic flux in the plasma. One method to prolong the pulse is to drive current with the auxiliary heating systems, which in turn has an impact on the economic viability of the system (meaning that the plasma is effectively no longer ignited). Another method makes use of the fact that in a toroidal plasma, a finite radial pressure gradient drives a thermo-electric (so-called 'bootstrap') current in the toroidal direction [10]. Hence, the fraction of non-inductively driven current increases with normalised plasma pressure $\beta_N$. This can be seen from Fig. 3 which shows the pulse length and recirculating power for different values of $f_{CD}$, the ratio of current driven by the auxiliary heating systems and $\beta_N$:

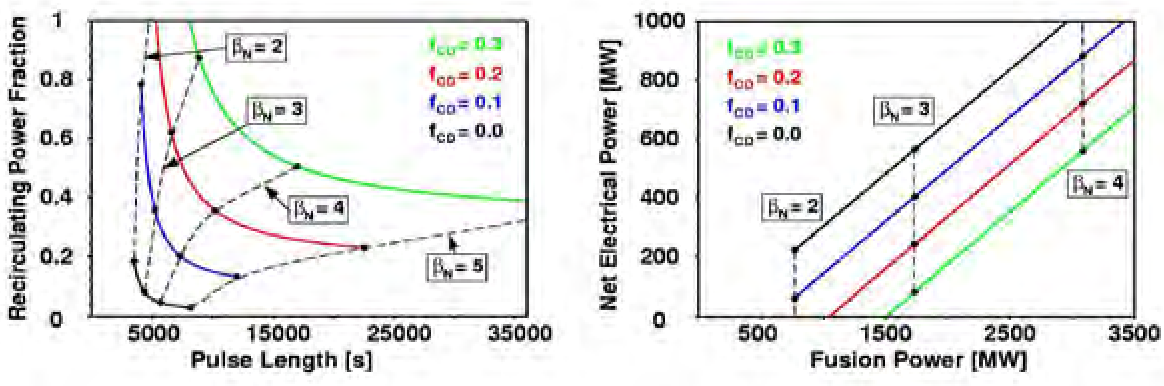



*Fig. 3: Left: Parametric dependence of pulse length and recirculating power for a variation of the externally driven current fraction $f_{CD}$ and normalised plasma pressure $\beta_N$ for typical tokamak DEMO parameters ($R_0$ = 7.5 m). The impact of the externally driven current on the economic viability is demonstrated in the right plot showing the net electrical power generated as function of fusion power (thermodynamic efficiency was assumed $\eta$ = 0.33).*

As can be seen, the power used to externally drive current has a severe impact on the recirculating power and, as shown in the right part of Fig. 3, on the net electrical power generated for given fusion power. This effect can be mitigated by chosing to operate at higher normalised pressure $\beta_N$. At the highest values of $\beta_N$ studied, very long pulses can be achieved, approaching steady state (i.e. fully noninductive operation), but as we shall show in the next sections, these values of $\beta_N$ challenge significantly the stability properties that have so far been achieved experimentally. It is hence at present not clear if a tokamak DEMO will have to be pulsed or can be steady state.

| | ITER | DEMO |
|---|---|---|
| $H$ | 1-1.2 | 1.2-1.4 |
| $\beta_N$ | 2 | 4-5 |
| $Q$ | 10 | 50 |
| $P/R$ | 20 | 65 |

*Table 1: Physics requirements for ITER and DEMO.*

Table 1 summarises the physics requirements that we have derived above. It can be seen that the biggest differences between ITER and DEMO are in the area of stability ($\beta_N$) and, most pronounced, exhaust ($P/R$). In the next section, we will review the present physics base, both experimentally and theoretically, in order to assess where major improvements are still needed to arrive at an economically viable reactor.

## 4. Present status of physics research

In this section, we review the present physic base in the areas of transport, stability, α-heating and exhaust and assess where major progress is needed to arrive at the values foreseen for ITER and DEMO. It is assumed that the DEMO parameters are not far from a reactor such that DEMO largely represents the requirements for a commercial fusion power plant.

### 4.1 Transport

In the area of transport, recent progress in both theory and experiment has led to a better understanding of the above mentioned turbulent transport of heat and particles that dominates under most conditions in present day experiments. It was realised over the last decade that the underlying gradient driven microinstabilities that give rise to the development of turbulence have a threshold behaviour w.r.t. the gradients such that the gradient-flux relation between temperature gradient and heat flux is nonlinear. Experimentally, the normalised gradients $\nabla T/T$ take on values not far from the onset value for turbulence which can also be understood as a 'critical gradient' phenomenon. This results in so-called 'profile stiffness', i.e. for given temperature at the plasma edge, the central temperature can be determined from the critical gradient and the profile are self-similar such that $T_1(r) = c\ T_2(r)$ if $T_1$ and $T_2$ are characterised by different edge temperature, e.g. due to different heating power applied to the plasma [11]. Recent theoretical progress allows to estimate the parameter dependence of the critical value under a variety of experimental conditions, hence giving confidence in our predictive capability beyond that of pure empirical scalings developed in the 1990s.

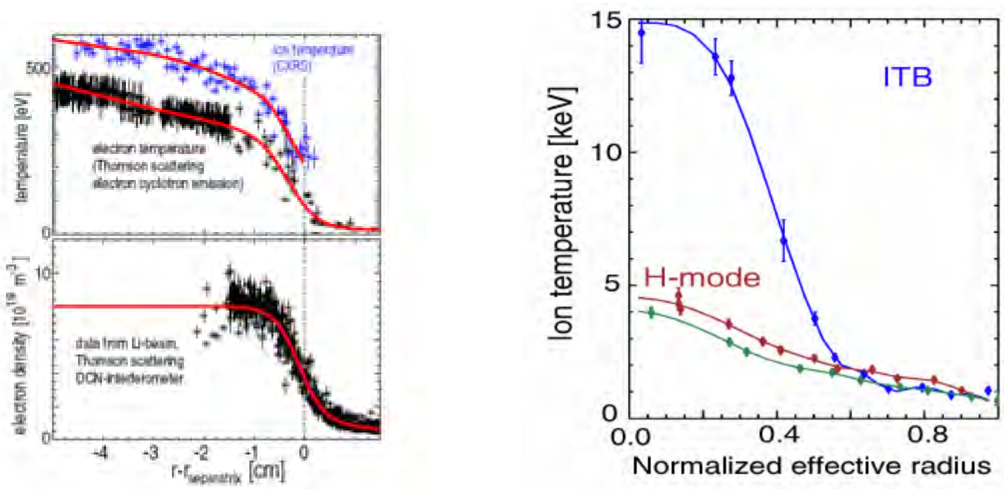


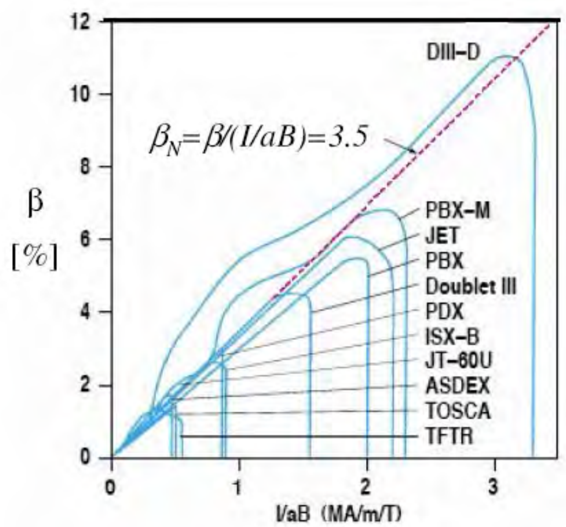

*Fig. 4: Two examples of transport barriers that lead to improved confinement via turbulence suppression in a part of the plasma. Edge transport barrier (left) typical for H-Mode conditions and internal transport barrier with further improvement in confinement.*

Improved confinement with respect to the above described turbulent transport has been realised with the discovery of so-called transport barriers, i.e. narrow regions in which the turbulence is suppressed and the gradients can by far exceed the critical values. Fig. 4 shows two examples from the ASDEX Upgrade tokamak in Garching, Germany: On the left side, an edge transport barrier can be seen, where the zone with steep gradients due to the suppressed turbulence covers the very edge of the confined plasma, i.e. just 1-2 cm out of the 50 cm minor radius of the plasma under consideration. This situation is typical for the so-called 'H-Mode' (high confinement mode) [12] that serves as a reference for ITER and corresponds to a confinement quality of H=1. On the right side of Fig. 4, an internal transport barrier can be seen that breaks profile stiffness at around half radius and leads to a tripling of the central temperature w.r.t. H-mode conditions. This mode of operation is not as well established as the H-mode, but holds the promise for further improvement ($H > 1$, as assumed for DEMO in table 1).

Our present understanding is that the suppression comes from a narrow zone of strongly sheared differential plasma rotation that decorrelates the turbulent eddies, thus significantly reducing their radial size [13]. However, the mechanism for generating the (experimentally observed) sheared rotation is not yet clearly identified and no self-consistent ab-initio turbulence simulation of the formation of transport barriers exists to date. Hence, this area is still subject to intense experimental and theoretical research.

### 4.2 Stability

Concerning the stability against pressure driven MHD instabilities, the predictions of ideal MHD stability leading to the Troyon scaling (underlying Eqn. (4)) have largely been verified experimentally. This is shown in the left part of Fig. 5 which shows the envelope of $\beta_N$ values reached in different devices together with the prediction from the Troyon scaling for a value of $\beta_{N,max} = 3.5$ [14].



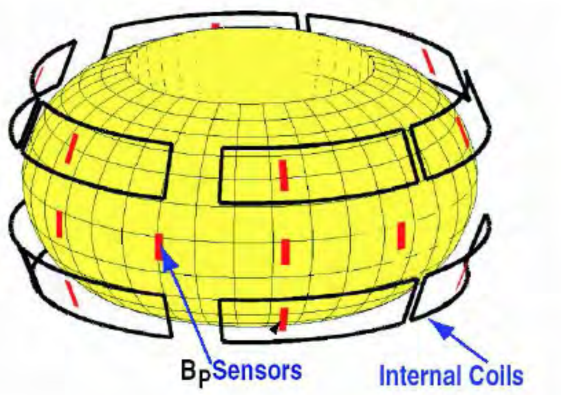




*Fig. 5: Envelope of experimentally achieved $\beta$-values indicating the relevance of the ideal MHD limit $\beta$ ~ I/(aB) (left) and schematic drawing of an internal coil set used for active control of ideal MHD instabilities on the DIII-D tokamak (USA).*

It can be seen that $\beta_{N,max}$ = 3.5 can be reached in practically all devices, with some experimental points even exceeding this value. Recent experiments applying active control of MHD instabilities such as counteracting the growing helical perturbation magnetic fields by actively controlled internal coils such as the ones shown in the right hand side of Fig. 5 may lead to reliable operation above the limits shown in the left part of Fig. 5 and are hence an important field of research in MHD stability.

We also note here that MHD instabilities which can develop due to finite resistivity effects and limit $\beta_N$ to values below that of ideal MHD have been observed in H-mode discharges in many experimental devices, but their active control using local current drive from heating systems seems in reach based on recent experimental results [15].

### 4.3 α-heating

Concerning the plasma heating by fusion generated α-particles, all present day devices are too small to reach a Q-value that would allow the experimental study of dominant α-heating. In D-T experiments in the world's largest tokamak device, JET, located in Culham, UK, 16 MW of fusion power were generated in a transient manner (left side of Fig. 6) and a clear heating effect could be seen by varying the isotope mix and finding a maximum temperature when the mix was close to the optimum 50:50 (shown in the right part of Fig. 6) [16]. Hence, there is some confidence that α-heating will work, but the results are obtained rather in a trace limit than when it is dominating. It will be one of the main goals of ITER to establish these conditions and study the nonlinear dynamics of α-heating.

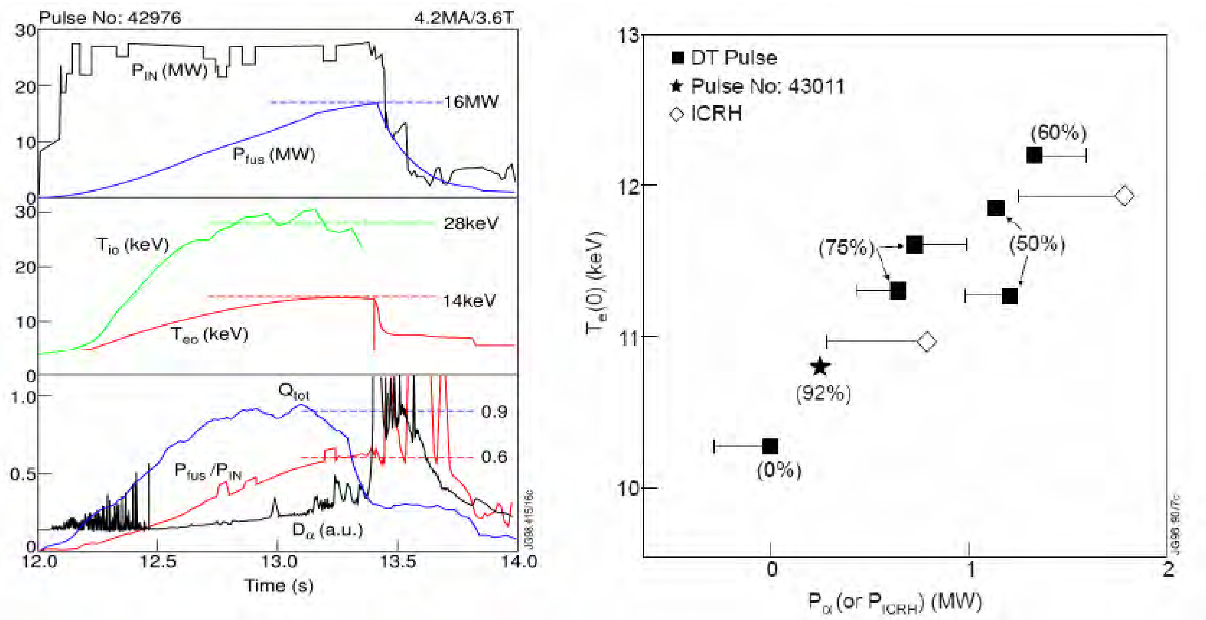



*Fig. 6: Demonstration of generating fusion power in a DT plasma in the JET tokamak (left) and experimental results that prove the effect of a-heating through a scan of D-T isotope mix.*

It has been mentioned above that there is a concern that the suprathermal α-particles could drive instabilities that could lead to an enhanced loss of α-particles and hence a reduction of the efficiency of α-heating as compared to classical slowing down. Such effects can be simulated in present day devices since the heating systems used can generate large populations of fast particles. Indeed, the excitation of instabilities by fast particles has been seen in several experiments and the observations are consistent with linear stability calculations [17]. However, the nonlinear dynamics of the interaction between the fast particles and the instabilities is much harder to model and a comprehensive predictive capability does not yet exist. This is especially true since the ratio of fast particle orbit width to the plasma radius and hence the radial extension of the instability will be much smaller in

ITER than in present day experiments leading to cross-coupling at different scales. The area of interaction between fast particles and MHD instabilities is hence another field of very active research in fusion plasma physics.

#### 4.4 Exhaust

In the area of exhaust, it has become clear that in order to progress to reactor type experiments, an integrated approach has to be taken that satisfies the sometimes conflicting requirements together. It is now clear that the retention of the Tritium fuel in a Carbon first wall can pose a serious limitation and moreover, the erosion of Carbon wall elements will be too high to give an acceptable life time for a power reactor. This has led to renewed interest in metallic wall materials, which were abandoned in favour of carbon in the late 1980s since at that time it was found that impurities released from a metallic wall may lead to unacceptable central radiation losses from the plasma.

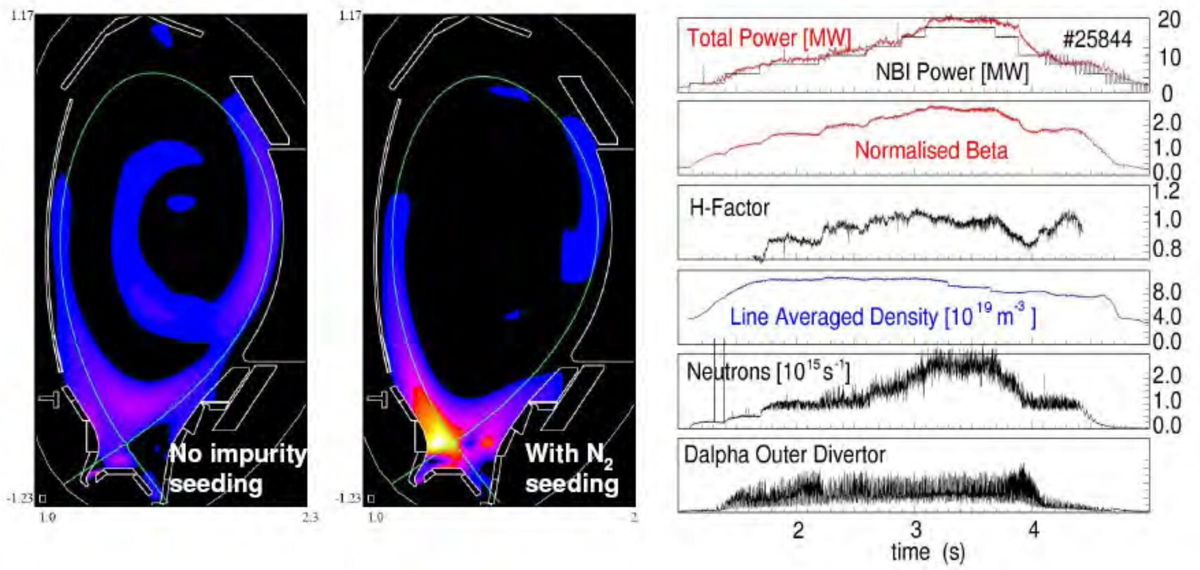



*Fig. 7: Demonstration of exhaust at P/R = 13 MW/m using impurity seeding to increase the divertor radiation in a high performance H-mode discharge in ASDEX Upgrade with fully W-covered first wall. Left: bolometric reconstruction of the total radiation power without and with impurity seeding. Right: time traces indicating the simultaneous achievement of good stability ($\beta_N$ = 2.7) and good confinement (H=1) even at the highest P/R.*

However, recent experiments on the ASDEX Upgrade tokamak using a fully Tungsten-covered first wall have shown that by tailoring the plasma edge, this material can be compatible even with high power discharges [18]. As mentioned above, a key to success is the controlled addition of impurities that radiate in the plasma edge region, thereby reducing the power flow hitting the first wall through charged particles. Fig. 7 shows an example where by the use of feedback controlled Nitrogen seeding, the radiation in the plasma wall interaction zone could be substantially increased without enhancing the core radiation losses. As a consequence, it was possible to demonstrate high plasma performance in terms of $\beta_N$ and $H$ at high $P/R$.

While these results are quite encouraging for ITER, a look at table 1 shows that for DEMO, substantial progress is needed in this area. In particular, modelling indicates that a mix of different impurity species may be needed for seeding in order to obtain enough peripheral radiation, since different impurities have their maximum radiation at different temperatures [19]. The modelling also suggests that some core radiation may be necessary as well to reduce the power flux in charged particles to an acceptable level. These approaches will have to be tested on present day devices and on ITER in preparation for DEMO.

## 5. Summary and Outlook

We have outlined the main areas of physics research needed in fusion plasma physics for future reactor grade devices such as ITER and DEMO. Simple criteria have been presented for the areas of transport, stability, α-heating and exhaust and the requirements for ITER and DEMO as well as the present understanding and experimental achievements were reviewed. These are summarised in table 2.

| | ITER (Q=10) | DEMO | achieved |
|---|---|---|---|
| *H* | 1-1.2 | 1.2-1.4 | ≤1.5 |
| $\beta_N$ | 2 | 4-5 | 3-4 |
| *Q* | 10 | 50 | 0.6 |
| *P/R* | 20 | 65 | 15 |

*Table 2: Same as table 1, but also showing the experimentally achieved parameters.*

It can be seen that the values relevant for ITER have more or less been achieved in present day devices, with the noticeable exception of α-heating, which is one of the main scientific goals of ITER. In that sense, ITER represents a conservative approach in terms of choice of plasma performance. For DEMO, substantial progress will be needed in the area of stability and exhaust to come to a design point which can prove the economic viability of fusion as an energy source. Hence, research in the next years will especially focus on these areas to improve our fundamental understanding of the underlying physics to be able to progress in this direction.

Finally, we note here that several other areas in which progress would be quite helpful, such as e.g. reaching high plasma density, could not be treated in this simple approach and the reader is referred to the literature for further discussion of these areas [20].

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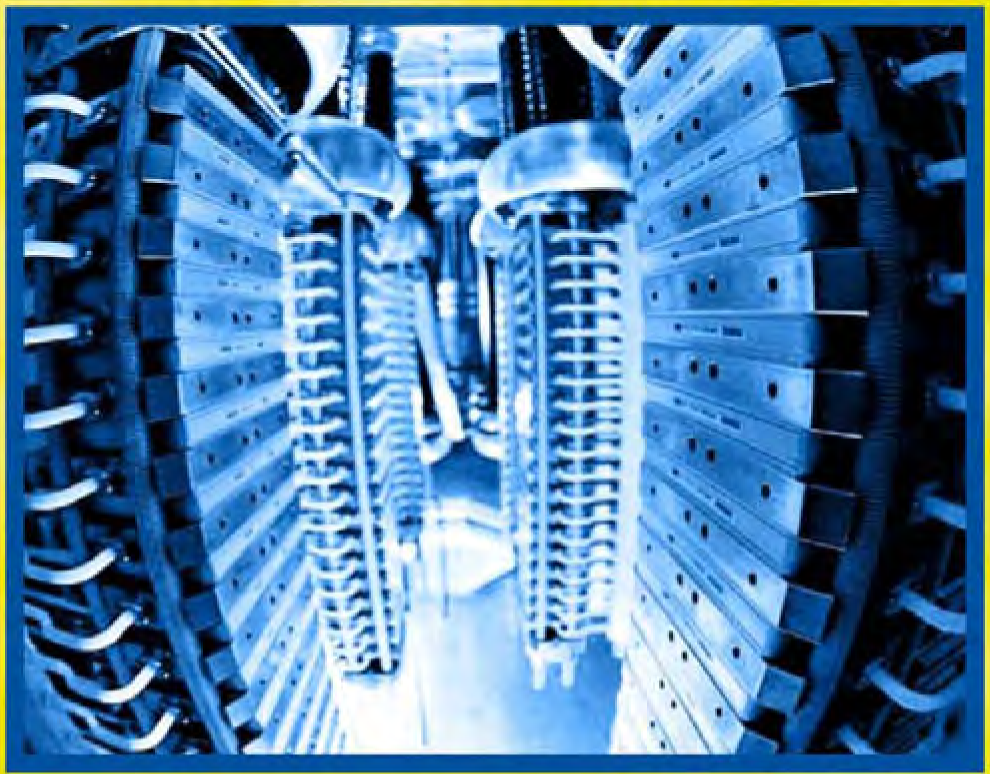
Arbeitskreis Energie





# Energie

## Technologien für die Zukunft

Vorträge auf der DPG-Frühjahrstagung in Bonn 2010

Herausgegeben von Hardo Bruhns

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Arbeitskreis Energie in der Deutschen Physikalischen Gesellschaft
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Bad Honnef, April 2011

# Frühjahrstagung des Arbeitskreises Energie in der Deutschen Physikalischen Gesellschaft Bonn, 15. und 16. März 2010

# Hauptvorträge

## Inhaltsverzeichnis

Der vorliegende Band fasst schriftliche Ausarbeitungen der Hauptvorträge der AKE Tagung des Jahres 2010 in Bonn zusammen. Leider ist es nicht gelungen, von allen Vortragenden Manuskripte zu erhalten. Die Präsentationsfolien aller Hauptvorträge können auf der Webseite des Arbeitskreises über:

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(von dort zu dem Archiv des AKE weiterklicken) eingesehen werden. Allen, die zu diesem Sammelband beigetragen haben, sei an dieser Stelle sehr herzlich gedankt.

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