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Recent Steps Towards a Controlled Thermonuclear Fusion Reactor with Results from the JET Tokamak Device

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ABSTRACT.

One of the three possible long term options for providing mankind with sufficient energy is controlled thermonuclear fusion. It is still in a stage of basic research, i.e. the conditions for producing a high fusion reaction rate for a sufficiently long period of time have not yet been mastered. However, the relevant figure of merit (the product $n \cdot \tau \cdot T$ of particle density, energy confinement time and temperature) has been improved by almost four orders of magnitude over the past 20 years.

Among the various approaches towards this goal the method using magnetic confinement of very hot plasmas is in the lead, and within this research line the tokamak devices have been most successful.

A present, the largest of the tokamak experiment is JET which is operated by the European Community (in England). The plasma parameters such as density and temperatures which have been reached, as well as the observed and yet unsolved problems, are close to the regime of values needed in a fusion reactor.

Among the recent advances the first time usage at JET of the real fusion reactants, deuterium and tritium, in some discharges stands out. In these pulses a significant amount of fusion power (1.7 MW) was generated.

In the near future the research line with tokamaks will be pursued vigorously throughout the world, for example with further extensive research programmes on JET (EC), TFTR (US), JT-60 (Japan), ITER (world wide), and on numerous smaller devices.

1. INTRODUCTION

Since the discovery that large amounts of energy can be released from nuclear fusion reactions, research has been performed with the aim of tapping this source of energy for practical purposes.

One of the largest cross sections is found for the reaction

$$_2D + _3T \rightarrow _4He\ (+\ 3.52\ MeV) + _1n^0\ (+\ 14.06\ MeV)$$

The deuterium (D) is very abundant in water, and tritium (T) can be produced by the reactions

$$_6Li + _1n^0 \rightarrow _3T + _4He\ (+\ 4.86\ MeV)$$

$$_7Li + _1n^0 \rightarrow _3T + _4He + _1n^0\ (-\ 2.87\ MeV)$$

from lithium (Li) which also is available everywhere and in huge quantities.

At present, mankind uses up the fossil energy resources, coal and oil. In the long term their deposits will be depleted, and only three serious options for energy production remain open:

(a) solar, (b) fission (breeding), (c) fusion.

Although strict economy and local optimisation in the usage of energy are highly desirable, solar energy alone seems to be insufficient to support the (growing) human population and the average standard of living. Among the two nuclear options (b) and (c), fission has already been clearly demonstrated to work while fusion yet holds the promise of substantial advantages:

(1) Inherent safety because it has to operate with low fuel inventory (only for about 30s of burning) and at low pressure in the reactor chamber;

(2) The reaction products are not radioactive, and there is hope that the reactor structures activated by the reaction neutrons could be fabricated from materials with fast-decaying radioactive isotopes. Thus the waste problem would be greatly diminished;

(3) The basic fuels, e.g. D and Li, are very widely available in the oceans and in the Earth's crust, can be cheaply extracted, and would last for thousands of years.

Fusion reactions provide the energy of the sun and other stars, and have on Earth been (mis-)used in the H-bombs. This indicates that in a controlled fusion reactor very extreme conditions as compared to our terrestrial environment are prevalent. D and T must be heated up into the temperature range of 100 to 200 million degrees. At these temperatures the gas molecules and atoms have disintegrated and form a plasma consisting of separated ions and electrons.

2. FIGURES-OF-MERIT DIAGRAMME

From a power balance for a (hypothetical) fusion reactor, i.e. by equating gains from the above reactions and external heating with losses (due to radiation, heat conduction, convection etc.) the so-called "triple product diagramme" for $n_i \tau_E T_i$ versus T_i (Fig. 1) can be constructed. Here n_i stands for the particle density of the reacting ions (D^+, T^+), T_i for their temperature, and τ_E measures in summary the energy loss rate in form of an "energy confinement time".

To obtain a positive power balance not only the temperature has to be in the 10-20 keV range (1 keV $\hat{=}$ 11.6 x 10^6 ^{0}K) but also the triple product must exceed

$$n_i \tau_E T_i \geq 5 \cdot 10^{21} \text{ m}^{-3}\text{s keV},$$

as indicated for an area in the top right-hand side of Fig. 1.

The upper left corner cannot be accessed because even for hydrogenic ions (i.e. no He or other impurities) the losses through bremsstrahlung exceed any possible gains by fusion reactions. For obvious economical reasons the reacting plasma should be self-sustaining. Therefore, the energy of the produced α-particles is supposed to be absorbed by the plasma to keep the temperature(s) sufficiently high. The energy of the neutrons must be evaluated in a surrounding blanket wall. The region to the lower right hand side in Fig. 1 is the so-called "hot ion regime". A reactor could in principle be operated there but the ions would always require external heating, because the fusion α-particles heat mostly the electrons, and in this region the equipartition between electron and ion temperatures is too slow.

The shaded line where the power of the α-particles alone suffices to compensate for all energy losses of the plasma is called "ignition line". If one demands only the total reaction power (of α-particles plus the one of neutrons) to balance the plasma losses, then "break-even" is reached and indicated by the line "Q_{DT} = 1". In general, the distance to the various lines is used as a measure of performance for any fusion plasma.

3. PRACTICAL APPROACHES TO FUSION

In the stars the sheer size and gravitational forces permit a fusion energy release. If one leaves out of consideration the physically very interesting case of "muon catalysed fusion" [1], on Earth the spatial extension of a very hot plasma is restricted with the help of magnetic fields or through inertia.

In the schemes of inertial confinement a solid pellet is irradiated with intense laser, or electron, or ion beams, and the fusion conditions are created on implosion of the fuel before the material is expanding.

In the magnetic confinement schemes the Lorentz force on charged particles is essential. In *linear* configurations such as *magnetic mirrors* or some *pinches* the plasma needs to be reflected at the ends. In order to avoid this difficulty many *toroidal* configurations have been devised such as stellarators, spheromaks, toroidal pinches, and tokamaks. The advances achieved with all these schemes are every second or third year summarised in IAEA-Conferences on "Plasma Physics and Controlled Nuclear Fusion Research" [2].

Of all these approaches the biggest strides forward have been achieved by the tokamak system (Fig. 2). A toroidal $\vec{B}$-field component is created by a set of coils around a ring shaped plasma vessel. A large transformer produces a voltage in toroidal direction, and drives, after electrical breakdown, a plasma current. This toroidal current is in turn connected with a poloidal $\vec{B}$-field component, so that the (total) $\vec{B}$-field lines wind helically around a centre line inside the vessel. Additional vertical $\vec{B}$-field components must be provided with external coils to shape the plasma and to control its vertical and radial position.

The (induced) current through the plasma not only provides the necessary poloidal field but also heats the plasma by resistive dissipation. For this reason, and to have better energy and α-particle confinement, a high plasma current would be desirable. Unfortunately, its size is limited by the ratio between the toroidal and poloidal field components in such a way that, following a helical $\vec{B}$-field line, the "safety factor"

$$q \equiv \frac{\text{number of toroidal turns}}{\text{number of poloidal turns}} > 1 \,,$$

otherwise the whole configuration is unstable. The value of q varies from the plasma centre to the boundary where it is typically $2 < q_{\text{bound.}} < 4$.

An elongated (non-circular) plasma shape allows an increase in current without changing the q.

Since the early tokamak experiments at the Kurchatov Institute in Moscow and at the Australian National Univerity in Camberra, a large number of such devices has been studied worldwide. The latest complete survey of the world's fusion experiments (not only tokamaks) is published by the IAEA in 1991 [3]. Four very large tokamaks have been operated during the last decade, JT-60 in Japan, TFTR (Princeton) and D-III (San Diego) in the USA, and JET (Culham, England) in Europe. Their main parameters are collected in Table 1.

4. TOKAMAK PERFORMANCE

These tokamaks have produced plasmas which almost fulfil the requirements for thermonuclear reactors. Ion temperatures of more than 380 million degrees [4] have been reached, particle densities of up to 4×10^{20} m^{-3}, and energy confinement time between 1 and 2 seconds [5]. Such maximum values have not been achieved (yet) in the *same discharge*. However, also the measured triple products [5]

$$n\tau_E T \approx 9 \cdot 10^{20}\ \mathrm{m^{-3} s\ keV}$$

are very close to the $Q_{DT} = 1$ line in Fig.1.

Considering the development of fusion research, an enormous improvement in these figures of merit can be observed. In Fig.1 the right hand side ordinate indicates the calendar years when the corresponding $n\tau T/T$-values have been measured. Over the last 20 years obviously several orders of magnitude have been gained.

Because of the finite number of volt seconds in the transformer, tokamaks are operated in pulses. A typical evolution of a JET discharge is depicted in Fig. 3. After the build-up of the toroidal $\vec{B}$-field to about 2.5 Tesla, the current starts rising to its plateau value of 3 MA. When the Ohmic heating power levels of, additional heating from outside is applied through neutral atom beams injection ("NBI", 14 MW). In other discharges alternatively radio frequency waves of similar power level are applied, sometimes simultaneously with beams.

The plasma density is determined mostly by recycling at the walls, by additional gas which is puffed-in at the boundary, and by the injected particles from NBI. The volume averaged density of ~ 2×10^{19} m^{-3} shown here is relatively low. The additional trace of the average ion charge Z_{eff} indicates the rise in the concentration of (non-hydrogenic) impurities released through contact of the plasma with the wall.

While during the Ohmic heating phase the electron temperature T_e and the ion temperature T_i are almost equal, the external heating methods improve predominantly T_i. During the increase of T_e the plasma resistivity, and with it the Ohmic heating power (P_{ohm}), is lowered, and with the inflowing wall material the radiation losses (P_{rad}) grow, at times very dramatically.

The maximum toroidal magnetic field for JET is 3.4 Tesla. Smaller tokamaks have used up to 9 Tesla [6]. The maximum currents (of 7MA) have been reached at JET [7] for several seconds.

5. THE JET PRELIMINARY DEUTERIUM/TRITIUM DISCHARGES

The pulse sketched in Fig. 3 was one of the JET discharges with the highest performance, i.e. with the highest neutron rates *from D/D fusion* reactions. For later comparison with Fig. 4b the relevant span of time is considered in some more detail in Fig. 4a for the same discharge (the origin of the time axis has been shifted to t = 40 s from the time axis in Fig. 3).

After an Ohmically heated plasma of $<n> \approx 2 \times 10^{19}$ m^{-3} and $T_e \approx T_i \approx$ 3-4 keV has been established, neutral beam power of 5 MW is injected which then is increased at (t = 12 s) to 15MW. Until t = 13.3s density, temperatures, internal plasma energy and neutron output increase steadily. A favourable so-called "H-mode" period with improved confinement is reached during the heating period. It will below be discussed further . At t = 13.3s the Z_{eff} suddenly rises, indicating a strongly increased influx of impurities from overheated parts of the wall. Ion temperature and neutron emission rate collapse immediately. This phenomena, called "Carbon-Bloom" (because the plasma touches a carbon tile of the wall) is observed in many JET discharges when high heating power is applied.

Impurity related problems of this kind are of major concern for future tokamak experiments where even higher wall loads must be expected, Nevertheless during the high T_i-period one of the best triple products $n\tau T$ of Fig. 1 was produced.

With respect to these "measured performance" points in Fig. 1 one should remember that all these data are somewhat fictitious. The temperatures, densities and confinement times have been measured in reality not for a Deuterium/Tritium mixture but for other (nonradioactive) gases such as H and D. The points in the diagramme have then been obtained by extrapolating to a 50/50 mixture of D/T and to the corresponding fictitious fusion reaction rates and energy gains.

The *open questions* for the use of real tritium do hardly concern the reaction cross section. They can be divided into a *plasma physical* and a *technological* group. The plasma physical uncertainties are connected with the unusual ratio e/m of tritium which could lead to demixing (from D), and also to additional resonances in (macroscopic) plasma oscillations and waves. Furthermore, there are the unknown effects of fusion generated α-particles which enter the plasma as a highly non-thermal plasma species, and which in a reactor plasma should constitute the main heat source.

On the technological side, the direct handling of the radioactive tritium, safety procedures, the penetration and retention of tritium in the walls, the cleaning techniques for its removal, and the effects of high neutron fluxes on the experimental structures and on the diagnostics, have to be dealt with both from technical and from administrative points of view.

The first experiments with an actual D/T mixture in a near-reactor plasma have been carried out at JET in November 1991. The tritium was fed as a neutral atom beam into a deuterium background plasma. Two of the 16 JET neutral beam sources injected tritium (24 A with 78 keV), the others supplied deuterium and energy (about 132 A with 135 keV, and 40 A with 75 kV). In this way a maximum ratio of average densities of $\langle n_T \rangle / \langle n_T + n_D \rangle = 0.11$ was achieved [8] in two discharges. In a number of trial discharges a much smaller ratio ($\approx$ 0.001) has also been used. Fig. 4b summarises the measurements in comparison with Fig. 4a. Both discharges were made under most similar experimental conditions, Fig. 4a with D only, Fig. 4b with D/T.

The main difference is, as expected, in the fusion neutron signal which has a peak of $6 \cdot 10^{17} s^{-1}$. The total (integrated) yield of neutrons was 7.2×10^{17} which corresponds to a peak fusion power of about 1.7 MW and a released fusion energy of about 2 MJ.
The other plasma parameters are, in fact, very similar; in particular both show a collapse of the ion temperature and neutron rates by "Carbon-Bloom" almost at the same time.

With respect to the above mentioned open physics questions, these JET experiments can only be regarded as a first step. Within the uncertainties of measurements, the tritium plasma did behave as expected. Thus there are indications that the tritium diffuses in the same way as the deuterium. The density of the fusion α-particles was too small to expect any influence on the plasma behaviour. The characteristic parameter

($\beta_\alpha \equiv$ energy density of α-particles/energy density of magnetic field)

reached a maximum of only $\beta_\alpha \approx 2 \times 10^{-3}$.

For extrapolations to a reactor plasma one must keep in mind that the *total power* input to run the discharge (fields, beams, etc.) is of the order of 500 MW. Also, the chosen discharge type was of the hot-ion-mode, i.e. a plasma with a large share of non-thermalised ions (lower right hand side of Fig. 1), and finally that the tritium was deposited directly into the plasma centre by beams, a method which would be very difficult to use in a reactor.

Much more conclusive and reassuring are the results on the above technological questions. For the first time high power neutral beams have been operated successfully with tritium. In total 1003 (± 70) Ci (37 TBq) of tritium were fed to the injector boxes, and of this amount 54 (± 6) Ci were injected into the torus.

By application of various methods of discharge cleaning the tritium was removed efficiently from the torus walls within a period of 20 days. Within the error bar of the measurements more than 98% of the tritium could be recovered, i.e. 1090 Ci (± 100) [9]. Measurements of the T-distribution within the carbon tiles show that about 98% are contained in the outermost 4mm layer, the remaining 2% more or less uniformly spread over the 20mm tile thickness [10]. The decay of the radioactivity caused by the neutrons in the experimental structures followed also clearly the calculated predictions.

6. ADVANCES IN FUSION PLASMA PHYSICS

Over the decades of research the empirical knowledge and the proper theory of hot fusion plasmas has increased enormously. However, there are still a good number of very important phenomena which are not yet understood, not qualitatively and even less quantitatively. This is largely a consequence of the general characteristics of plasma physics where most important expressions and equations are non-linear and multi-dimensional, and where usually many effects are interlinked and interact simultaneously.

Both non-linearity and complexity require, in general, numerical analysis and extensive computations. Some plasma aspects such as macroscopic equilibrium configurations and their stability can be described through approximating the plasma (section 6.1) by a conducting fluid in a magnetic field ("magneto-hydro-dynamics" ≡ MHD). Heating processes and non-inductive current drive (sections 6.2 and 6.5) already require kinetic theory, i.e. non-Maxwellian velocity distribution functions, mixed with continuum theory for electromagnetic wave propagation. Plasma transport (section 6.4) is still escaping any proper theoretical grasp.

6.1 Configurational Equilibrium and Stability

Plasma equilibria for given external and internal current distributions are computed, for toroidal symmetry, from the local force balance (in cgs-units)

$$\frac{1}{c}\left(\vec{j} \times \vec{B}\right) = \nabla p,$$

where p denotes the total plasma pressure, $\vec{B}$ the magnetic field, and $\vec{j}$ the current density. Although $\vec{j}$ can still not be measured accurately, there seems to be amazingly good agreement with the observed equilibrium. The external currents necessary for a desired plasma shape, and for shifts of its position can well be predicted from such computations. Thus, very long pulses (e.g. 60 sec. in JET [11]) can be produced with feed back control.

Not all such equilibria are stable. Extensive calculations have produced MHD stability criteria. One example ($q > 1$, in section 3) has already been mentioned above. Another one, called Troyon limit, has been derived from a large number of computations numerical-empirically, and concerns a limit on the parameter β, similar to the previously given β_α, and defined by

$$\beta \equiv \frac{\text{thermal energy density}}{\text{magnetic field energy density}} .$$

Physically this limit describes the onset of MHD-instabilities of ballooning and kink type. Fig. 5 compares the highest measured JET-β-values with the Troyon limit, and finds good agreement within the error bars.

The best β-values of about 11% [12] have been obtained by the DIII-D tokamak, using an extremely elongated plasma shape with a ratio of 2.35 between the

(elliptic) cross section axes. Such elongations are, however, difficult to keep vertically stable.
The higher the β, the more economic use is made of confining $\vec{B}$-field strength. In this respect the observed limits are rather low. Theory has predicted the existence of another stable range at higher β, separated by an unstable gap from the lower one. However, there is no clear experimental evidence that this higher β regime is accessible.

Besides these successes it remains often difficult to obtain good agreement between observations and theory, for example when the onset of the instability depends critically on local profile features, or when non-MHD effects become important. Also, the nonlinear evolution of an MHD instability, its saturation limits, are mostly beyond the present capabilities.
The characteristic times of MHD phenomena is very short (Alfvén transit times, order 10^{-6} sec.). Two very important phenomena, (which are not yet fully understood), show such time scales during the sudden destruction of the whole, or of parts of the plasma configuration. These are the "*disruptions*" where the plasma current stops momentarily, and the (internal) "*sawtooth oscillations*", most apparent in T_e (Fig. 6). Both are connected with the formation of magnetic islands (helical deformations) around the surfaces where $q = 1$ and/or $q = 2$; however the trigger(s) of these fast events and the (non-linear) evolution are not clear for all cases.
It is possible to stabilize sawtooth oscillations by the presence of high energy ions produced by absorption of magnetosonic waves from ICRH (ion cyclotron resonance heating) [13]. In this way up to 8 sec. of sawtooth free discharge periods have been obtained [14]. Fig. 6 shows an example. It seems also possible to stabilize sawteeth by modifying the $\vec{j}$-profile in the plasma core by lower hybrid waves.

6.2 Densities

Even below the β-limits it is not easy to increase the density. From earlier experiments to so-called "Murakami limit" (with R denoting the major plasma radius and n_e the electron density)

$$n_e Rq/B_{tor} < 12 \times 10^{19}\ m^{-2}T^{-1}$$

was empirically derived. Beyond this limit the plasma disrupted.

In these experiments the particles were supplied by gas puffing from the boundary, resulting in flat profiles characteristic for a boundary source. With central fuelling by injection of solid fuel pellets, or by neutral beams, profiles peaking in the interior could be produced, and the Murakami limit could be overcome. Fig. 7 presents the evolution of density profiles in a JET discharge [15]. After initial gas puffing one pellet was injected at t = 44.5 s and another two at t = 45.4 s. The speed of the pellets needs to be in the km/sec range, and much higher for a reactor plasma. Even neutral beams, to penetrate to a reactor plasma core would need energies in the MeV range. Because such high speed systems would be totally uneconomical, neutral beams from accelerated negative ions are being studied but not yet available at sufficient power levels.

A special problem for the density of the hydrogenic ions is caused by impurity ions with higher charge Z, to which also the reaction product He (α-particles) must be counted. Even if the radiation losses are tolerable they can reduce strongly the fusion reactions by *dilution*.

6.3 Temperatures

The high temperatures needed for fusion reactions can today be produced reliably. After break-down the tokamak plasma is heated by the plasma current. Further temperature increase is achieved by injecting *neutral beams* or *radio frequency waves*. When the threshold of fusion reactions has been reached, the fusion α-particles should be the only heat source for the plasma to maintain the reaction. No limitation for the ion temperature has been found. It seems to be more difficult to increase the electron temperature which in JET is observed to saturate with increasing RF power input.

With *Ohmic heating power* alone (which in JET typically reaches a few MW) temperatures of 4-5 keV can be obtained. Only for very low densities the electron temperature T_e exceeds appreciably T_i. The resistivity decreases with the electron temperature, and so does the heating power [see Fig. 3 during NBI]. This resistivity decrease has on the other hand the advantage that only a small driving voltage is required for the tokamak current in a hot, reactor relevant plasma.

Only neutral atoms can penetrate the magnetic field which confines the tokamak plasma. To create *neutral beams,* ions are accelerated through a grid system to the required energy of typically 50-150 keV. The ion beams are then directed through neutral gas and partially neutralised by charge exchange, the remaining still ionized beam majority is dumped. Typical power levels are noted in Table 1. In the plasma the atoms are re-ionized by collisions with ions or electrons, or undergo again charge-exchange, depending on the local plasma parameters. This deposition is well described by classical theory as seems to be the slowing-down process of the fast ions in the plasma background. During injection the ion-distribution function is, of course, non-Maxwellian, and the high energy component can produce non-thermal fusion reactions (as in the JET D/T experiment described in Section 5).

Beams provide, of course, not only energy but also fuel particles and, often not so desirable, momentum which can disturb the symmetry. The difficulties in extrapolating the application of beams to a reactor plasma have already been mentioned.

Corresponding to resonances with the plasma particles, electromagnetic waves mainly in three frequency regimes are excited *("Radio Frequency Heating").* These are the ion cyclotron resonances heating (ICRH) in the 20-100 MHz range, the electron cyclotron resonance heating (ECRH) in the range of 40-100 GHz and the lower hybrid resonance heating (LHRH) in an intermediate range around 1 GHz.

From the large tokamaks ICRH has turned out to be the most successful. Generators which can provide power of tens of megawatts (see Table 1) are available. The frequency can be chosen to a desired energy deposition profile (e.g. near the plasma centre), and there is no accessibility problem for the waves. Earlier antennae had caused strong release of impurities; this has been ameliorated by improved design [16]. The efficiency of ICRH in present experiments is about 60%.
Because the generators cannot accept large reflected power, the coupling of the waves into the plasma needs care; the resistance of the antennae seems to depend on the plasma edge parameters. As a solution, feed-back shifting of the plasma to keep the resistance constant with changing edge parameters, seems

viable [17]. The theory of ICRH has been developed over the last decade mostly at JET [18]. Consistently combining wave propagation and kinetic (Fokker-Planck type) particle responses in large computer codes yields good agreement with observations.

Power from ECRH, in resonance with the electrons, couples very easily into the plasma, and energy deposition can be very well localised. In this high frequency range the wave guide antennae need not to be close to the plasma surface, thus avoiding problems with impurities and mechanical forces. A wider application of this attractive method is still prevented because powerful gyrotrons at sufficiently high frequencies are not (yet) available.

The main difficulty of LHRH lies in the strong dependence of the wave propagation on plasma density. It is difficult to access a high density plasma core, and the wave guide antenna must be very near the plasma edge for efficient coupling. Lower hybrid waves are more favoured for current drive (see Section 6.5).

6.4 Fluxes

The confinement times τ_E in the triple product (Fig. 1) are dominated by transport processes, i.e. mainly by *particle and heat fluxes.* In spite of huge efforts over several decades these important plasma processes have not yet been clarified.

Observations reveal that some transport coefficients such as the electrical resistivity are roughly "classical", i.e. as predicted by theory. This is also found for energy exchange rates between various ion species, and between ions and electrons, as well as for the slowing-down times of fast, non-Maxwellian particles in a thermal plasma background. These phenomena depend essentially on the local (Coulomb) collision frequencies, and the measurements seem to confirm this "classical" effect.

On the other hand, large discrepancies (of factors 10 to 100) between measured data and theory are found for particle fluxes, and for the fluxes of momentum and heat perpendicular to the confining magnetic field. The difference to the above group is the fact that these fluxes are not only governed by collisions but also by the shapes of the particle orbits, as has been worked out most consistently in the

"neo-classical theory" [19]. Numerous attempts to explain such "*anomalous transport*" by unstable waves (e.g. drift wave instabilities) or by more general theories of weak and strong turbulence have as yet failed [20].
The pre-conditions for specific instabilities and turbulence are in most cases fulfilled only in parts of the plasma. Therefore, even the evaluation of these theoretical models needs to be done by extensive computations with "plasma transport codes" [21]. With their help consistent density, temperature, current etc. profile evolutions can be obtained and be compared with measured data.
In view of the continuing lack of success of deductive theory, these transport codes have also been used inversely to find consistent empirical formulas when the measured data are used as input. The obtained empirical formulae can at least be used to describe the data (although without understanding the underlying physics).

An even cruder form of empirical transport description is found by regression analysis of a global confinement time (for the plasma as a whole) against externally controllable variables such as plasma current, heating power input, average particle number, type of fuel etc. These so-called "*scaling laws*" are at present used for many practical estimates on existing tokamaks and also for extrapolating to future fusion devices.

A few important features of observed transport (Fig. 8) should be mentioned. There is a disturbing degradation of τ_E with increasing input of heating power into the plasma. This decay of confinement makes it at least more difficult to reach ignition. Confinement, however, is observed to improve with higher plasma current (Fig. 8a).

Extrapolating to a reactor with the most commonly observed absolute values for τ_E ("L-mode" confinement) leads to an unrealistically large size tokamak. In recent years, fortunately, several new operating conditions have been discovered, where the τ_E improves by considerable factors (up to 3), although only temporarily. These are the "H-mode" (Fig. 8b) or "S-mode" confinement regimes. The transition from L- to H- modes seem to be connected with changes in the boundary plasma; there is however, no generally accepted explanation yet. The improved confinement would allow technically acceptable next step tokamak devices capable of reaching ignition.

6.5 Towards Stationarity

Because of the limited flux (volt seconds) in the transformer the tokamak current, and thus the plasma, is operated in pulses. For a fusion reactor one would, however, greatly prefer stationary operation.

The low resistance of a hot plasma would allow transformer induced currents for many minutes, durations which certainly would be sufficient to demonstrate the existence of a controllable, ignited and burning, fusion plasma. Nevertheless, considerable effort is being spent already to overcome this future deficiency of the tokamak concept.

For many years, superconducting toroidal coil systems (for the main toroidal field component) have been built and tested, for example in the Russian tokamak T-15 and the French Tore-Supra. The Japanese TRIAM even succeeded with superconducting fields as high as 8 Tesla [22].

More difficulties are encountered for a stationary plasma current. It has to be either sustained by *non-inductive* drive or by A.C. operation. The latter was tested recently at JET [11] with current reversal between two successive discharges.

Both neutral beams and RF-waves can be applied to drive current. Neutral beams have been shown to drive currents for 2.5 seconds in DIII [22]. For a reactor, of course, the already mentioned penetration problems do exist.
Lower Hybrid waves have shown the highest efficiency (driven current per unit RF power). In JT-60 a 1.5 MA plasma current was driven by 4.2 MW at a frequency of 6GHz [24]. The efficiency $\eta = 0.36 \times 10^{20}$ Am^{-2}/W is close to the level required for a reactor, although the density, as it is characteristic for all Lower Hybrid applications, was low.

At least part of a non-inductive current could be driven by a classical effect from temperature and density gradients, i.e. by the so-called *bootstrap current* [25]. Inspite of the failure of neoclassical theory with respect to transport, the bootstrap current seems to have been observed. In JET [26] recently 70% of a 1 MA current have been attributed to bootstrap action while the temperature gradients were sustained by ICRH. In JT-60, 80% of a 400 kA current were estimated to be bootstrap driven [24]. For gradients in a reactor plasma the current could almost exclusively be provided by the bootstrap effect.

7. OPEN PROBLEMS AND PLANS FOR FUTURE RESEARCH

There are still a number of problems which are "on the critical path" to the demonstration of fusion power and for which no solution is yet known. They might be summarised as the investigation and achievement of:

(a) ignition and burning, i.e. the α-particles from the fusion reactions alone should sustain the high thermal energy level of the plasma;

(b) a long period (at least minutes) of well controlled (in spatial distribution and time) high temperatures, densities of all particle species, and magnetic field components, including the condition of the first wall.

With respect to point (a) the behaviour of small traces of α-particles has already been studied for example in the above described JET D/T experiments [8]. With ICRH ^{3}He - ions in the MeV energy range can be produced and their slowing-down and confinement be studied [18]. The slowing-down seems to be classical, and the confinement to be not worse than for the background plasma; however, these ICRH accelerated ions are strongly anisotropic with their high velocity components only perpendicular to the $\vec{B}$-field. There are theories which suggest for isotropic α-particles a less favourable behaviour [27].

The next experiments with the aim to expand our experience with tritium, α-particles and neutrons are scheduled to begin in July 1993 [28] at the TFTR device in Princeton (USA). Several hundred high performance D/T - pulses are foreseen until October 1994. Major points in the research plan concern the behaviour of α-particles and their interaction with the background plasma, various tritium fuelling methods including tritium pellet injection. As a technological problem the decommissioning of radio-activated tokamak structures will be investigated starting in 1995.

At JET another series of a few D/T discharges in a divertor configuration is being considered for 1994, and according to present planning full-scale D/T-operation (similar to TFTR) will be undertaken in 1995 and 1996.

In neither device, however, ignition can be expected to occur. The plasma evolution to ignition could be studied in a high-B-field (~ 10 Tesla) tokamak such as IGNITOR which is being designed [29].

With respect to the above point (b) a key difficulty is the poisoning of the plasma by impurities from the walls (or finally in a reactor plasma also from the "He-ash"). The coming JET program is focussed on this problem. Divertor coils are being installed near the inside bottom of the vessel to produce a so-called "X-point" configuration where the confined plasma is kept at a distance from the wall and where the particle and energy outfluxes are concentrated onto special dump plates [30].
Very long pulse discharges to study plasma control are in discussion for a future JT-60 upgrade version.

It should be noticed and emphasised that many of the important fusion plasma problems have turned out to be the same in the large tokamaks (see Table 1) and in the small ones. This holds, for example, for anomalous transport, current disruption, sawtooth oscillation, impurities, current drive, etc. All these problems can, therefore, well be studied in smaller devices most likely with less expense in time and money, and with more flexibility.

A big next step tokamak device which is supposed to demonstrate solutions for the combined above areas (a) and (b), and which is also aimed at the evaluation of the neutrons emerging from a stably burning plasma (e.g. blanket features, tritium breeding) is being planned in a world-wide collaborative effort, under the name of ITER (International Thermonuclear Experimental Reactor) [31]. In Fig. 8 its (preliminary) dimensions are compared to those of JET. A detailed design for ITER is expected to be available before the year 2000.

8. GENERAL CONCLUSIONS

Nuclear fusion is a most promising option for future energy production for mankind.
As for almost any scientific problem the way to a fusion reactor can be divided in two principle steps:

> *Step 1:* prove of existence; this is in our case to show that the fusion energy production conditions (as they are prevalent e.g. in the interior of stars) can be established on Earth.

Step 2: Optimization with respect to economy, safety, environmental issues, etc.

Fusion has certainly not completed the first step, and must therefore still be considered as *basic research* with possible speculative extrapolations to a practical reactor. This holds for any predictions on development time scales and final applicability.

With the tokamak scheme, enormous progress towards achieving reactor plasma parameters has been made over the past decades. Temperature above 20 keV, densities above 10^{20} m^{-3}, energy confinement above 1s, represent single reactor conditions and have been achieved, and the triple product $n_i\tau_E T_i$ is now near break-even, i.e. near the "Q_{DT} = 1" - line.

Although a number of formidable problems, such as ignition and burn control, plasma purity, protection of the first wall, and others are not yet solved, I can also see no principle obstacles ahead for further progress (with a tokamak). A similar statement seems to hold for the foreseeable fusion technology.

An integrating design of a large tokamak of ITER type is certainly useful, and such a device will finally have to be built. Many not yet understood plasma physics problems can and should be investigated on smaller devices without delay. This seems to me more economical both in costs and in time, and would obviously reduce the risks for a success of an ITER type device and a fusion reactor.

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TABLE 1

Dimensions of the four largest tokamaks in the world and the design parameters of ITER.

	TFTR	DIII-D	JET	JT-60	ITER
Minor radius [m]	0.85	0.67 1.36	1.25 horz. 2.1 vert	0.95	2.15 4.3
Major radius [m]	2.48	1.67	2.96	3.0	6.0
Toroidal magnetic field [tesla]	5.2	2.2	3.5	4.5	4.85
Plasma current [MA]	3.0	3.5	7.0	2.7	22
Pulse length [s]	5	5	60	10	2000
Additional Heating Power					
Neutral beam [MW]	32	16	21	20	75
ICRH [MW]	6.3	2	18(24)	6	115*
LH [MW]	0	0	10	24	45
ECRH [MW]	0	2	0	0	20

* ICRH is alternative to NB

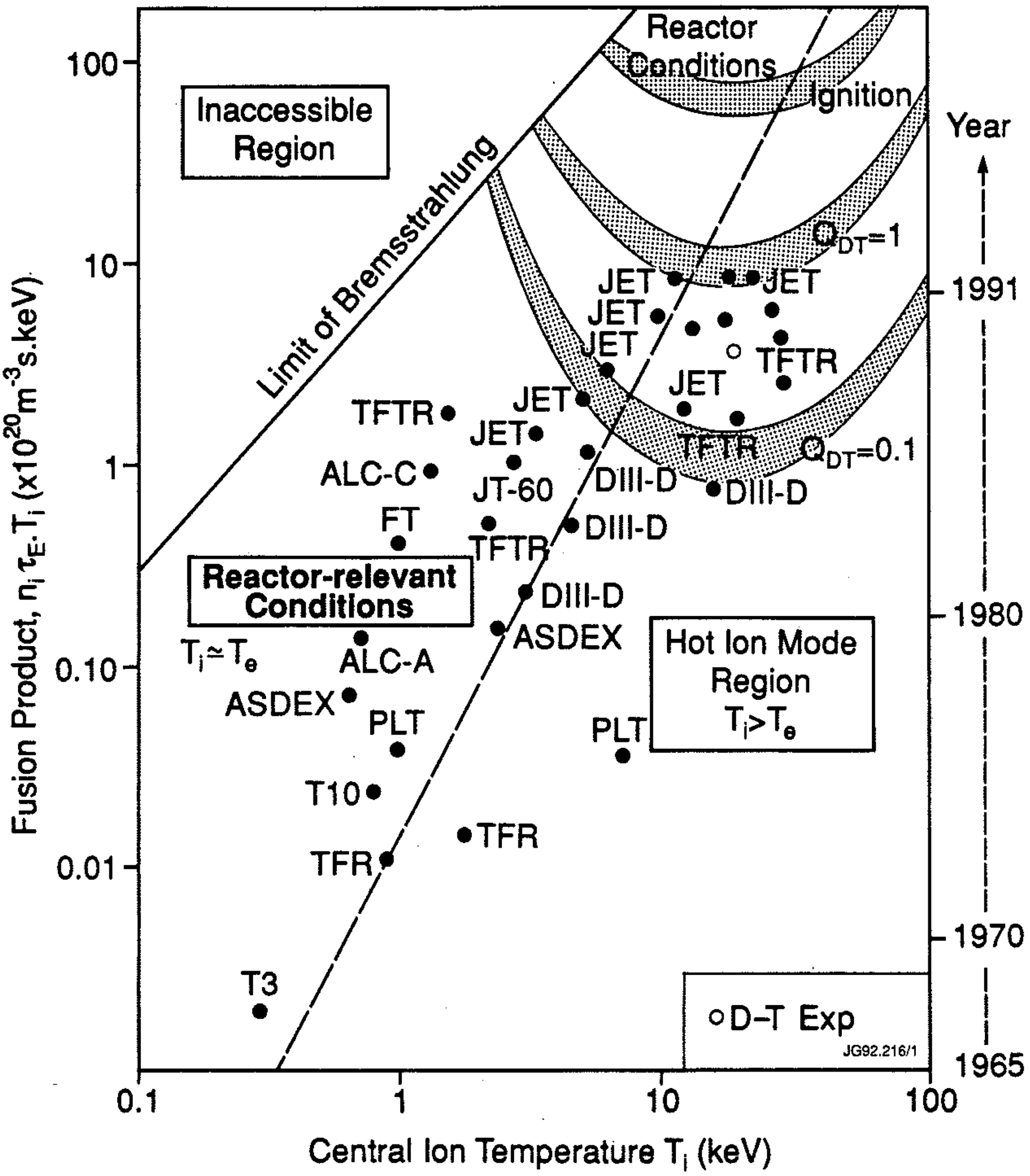

Fig. 1 Fusion Figures of Merit. Triple product $n_i \tau_E T_i$ versus ion temperature T_i. To the left, typical year for which the experimental results were achieved.

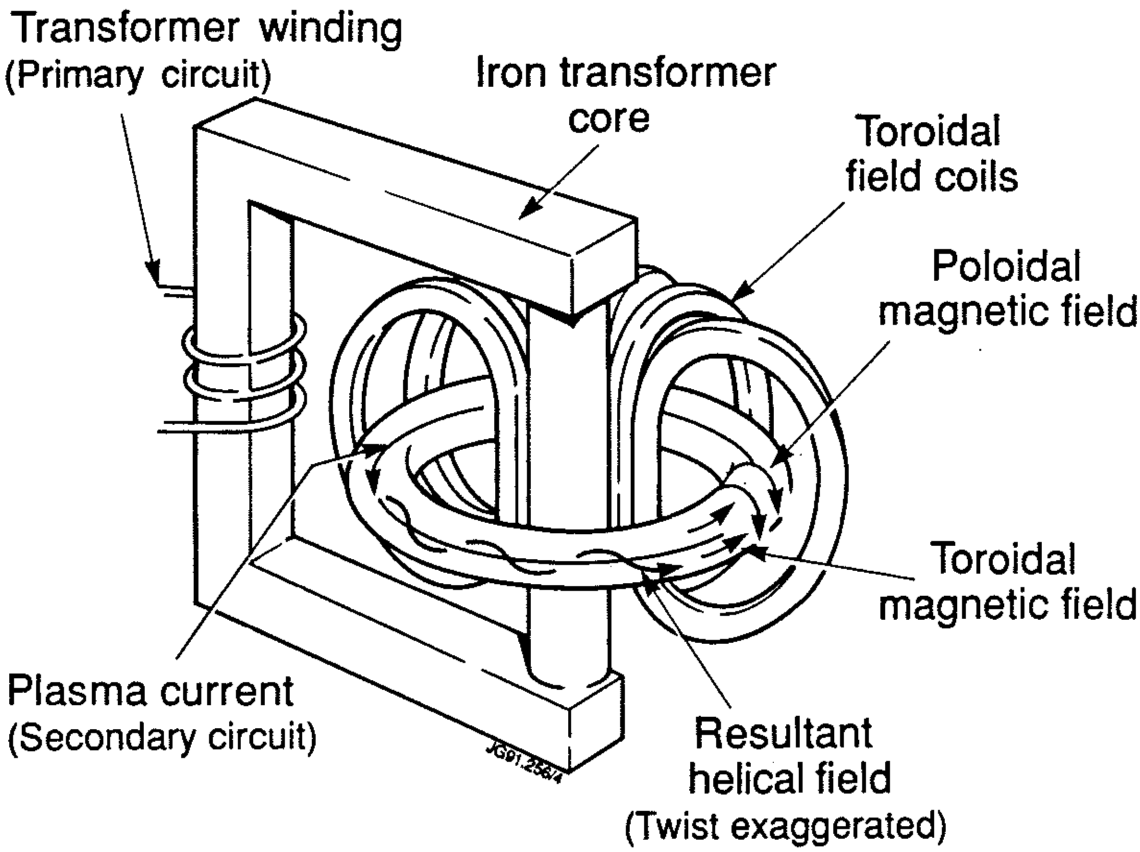

Fig. 2 Tokamak concept.

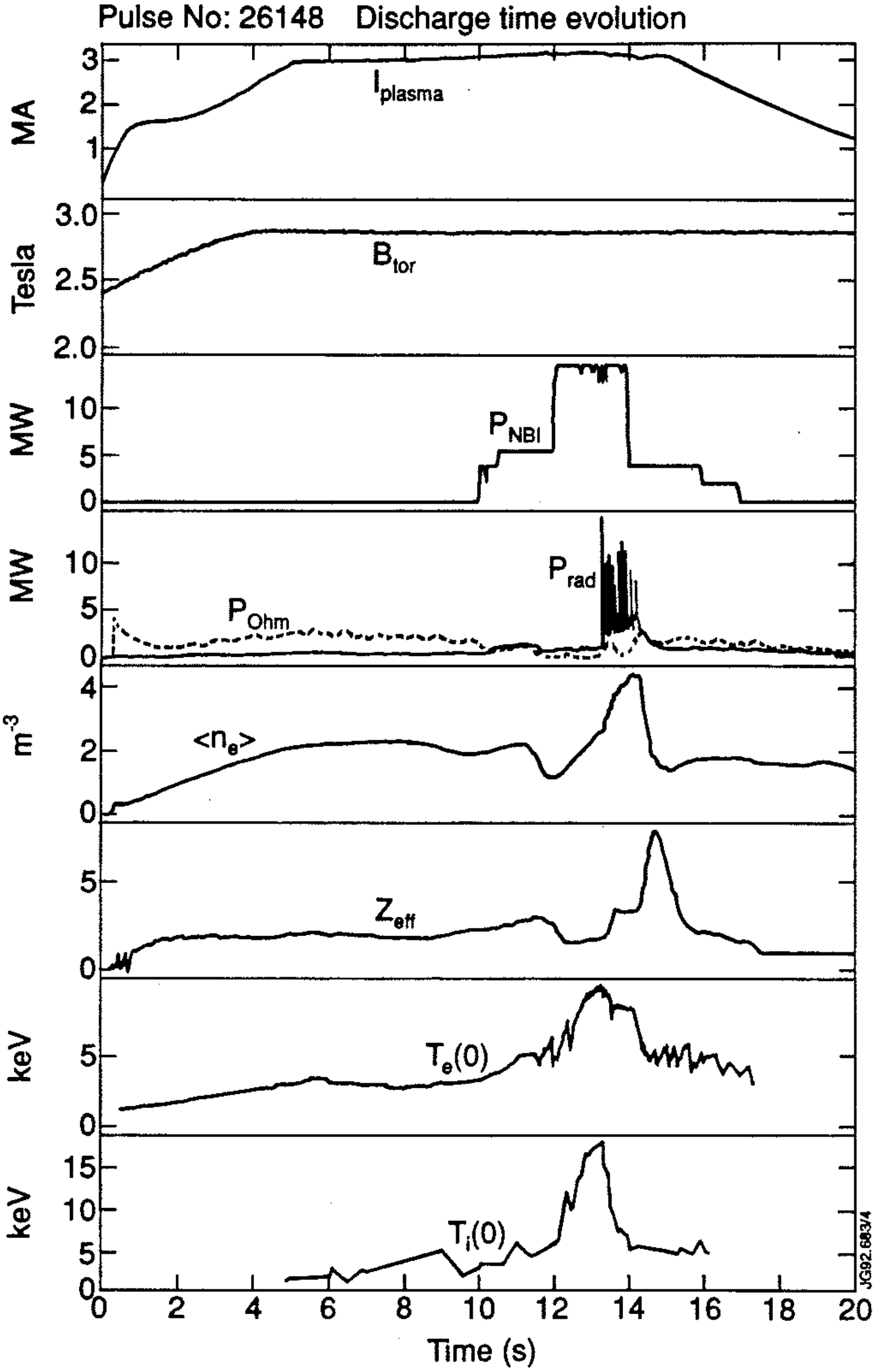

Fig. 3 Evolution of typical JET discharge (#26087) parameters.

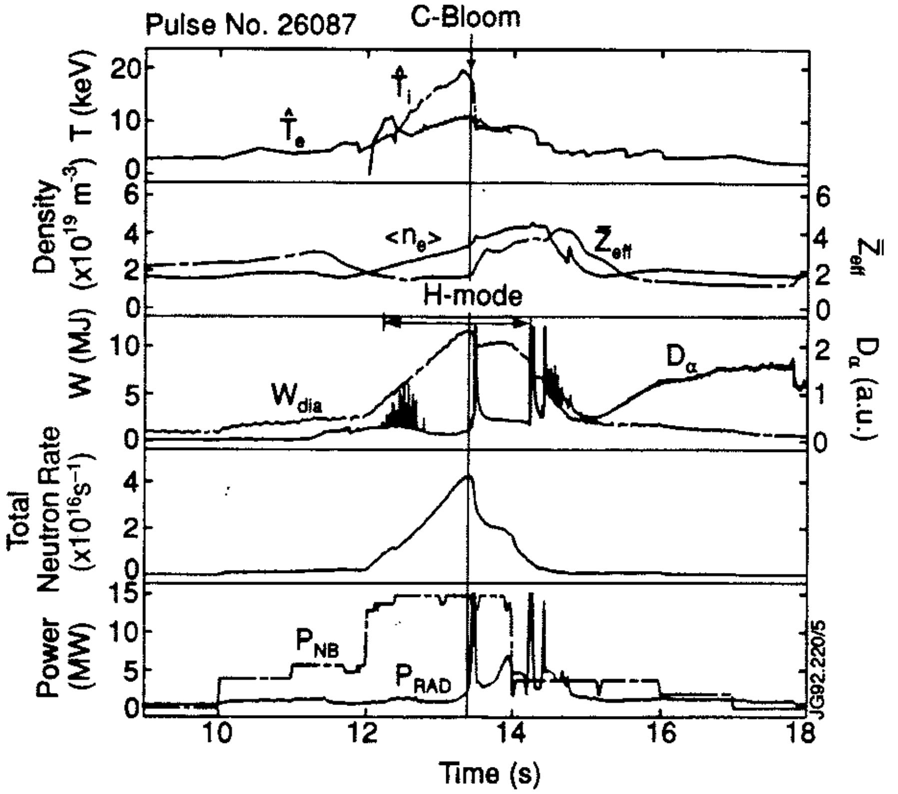

Fig. 4a Details of the discharge parameters for JET discharge #26087 in deuterium. Central electron and ion temperatures T_e and T_i, line-averaged electron density $<n_e>$, effective charge, Z_{eff}, energy content W_{dia}, and D_α line emission, neutron rate, neutral beam power, P_{NB}, and radiated power P_{RAD}, are shown during the heating period.

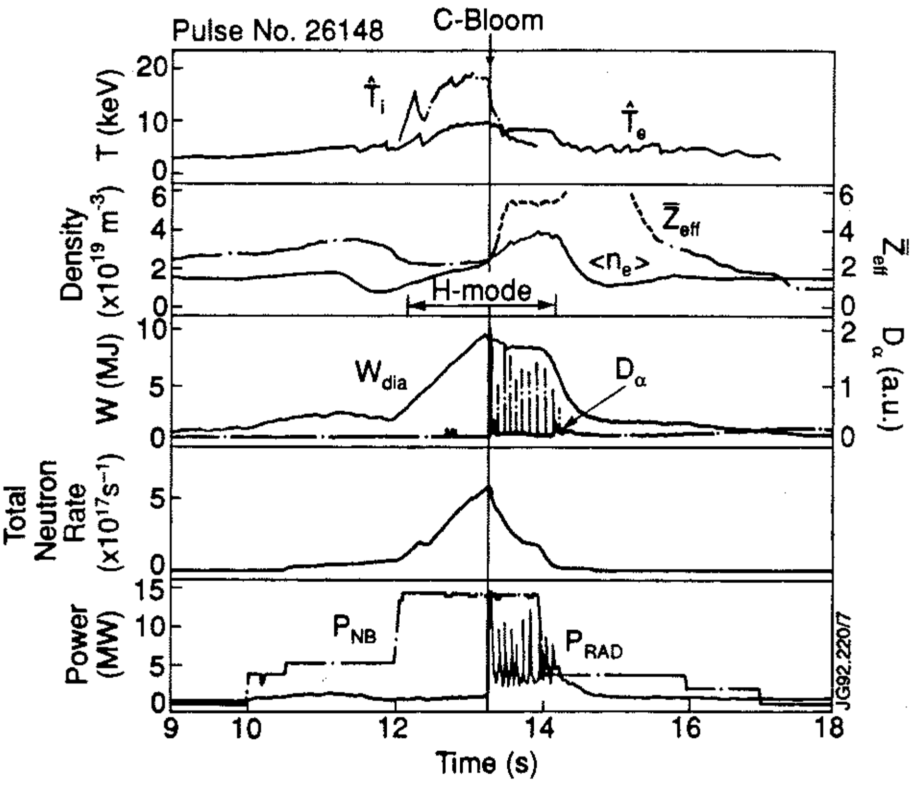

Fig. 4b The corresponding traces for JET discharge #26148 in *deuterium and tritium.*

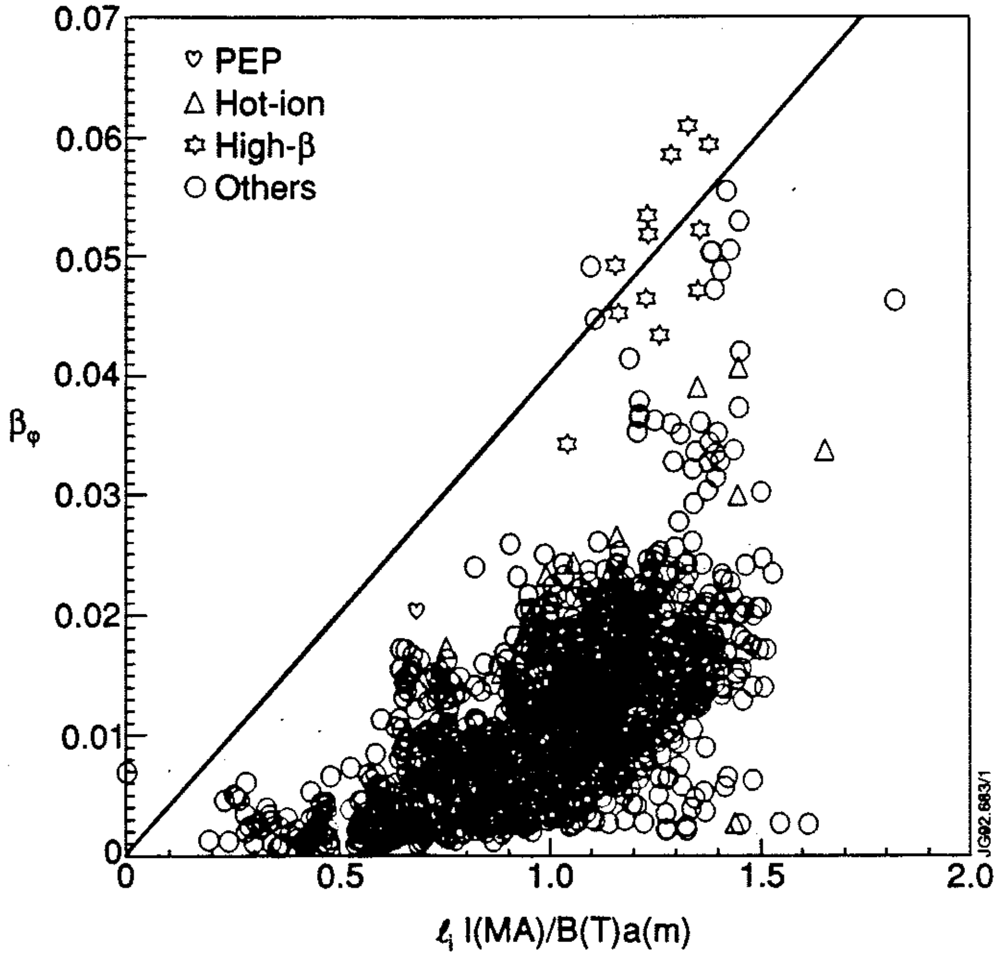

Fig. 5 Maximum toroidal β_φ-value as a function of $l_i I/aB$, where l_i is the internal plasma inductance and a the minor plasma radius. The line indicates the Troyon limit.

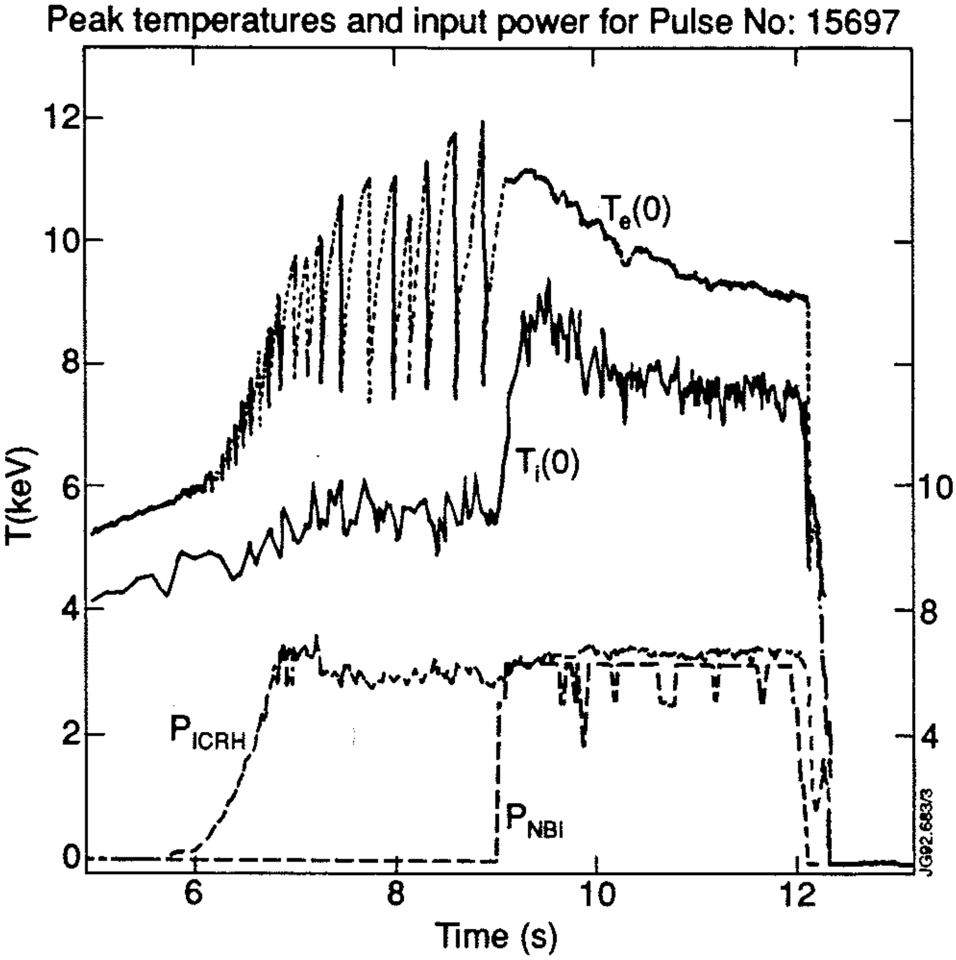

Fig. 6 Evolution of the central electron and ion temperatures, neutral beam and RF power for JET discharge #15697.

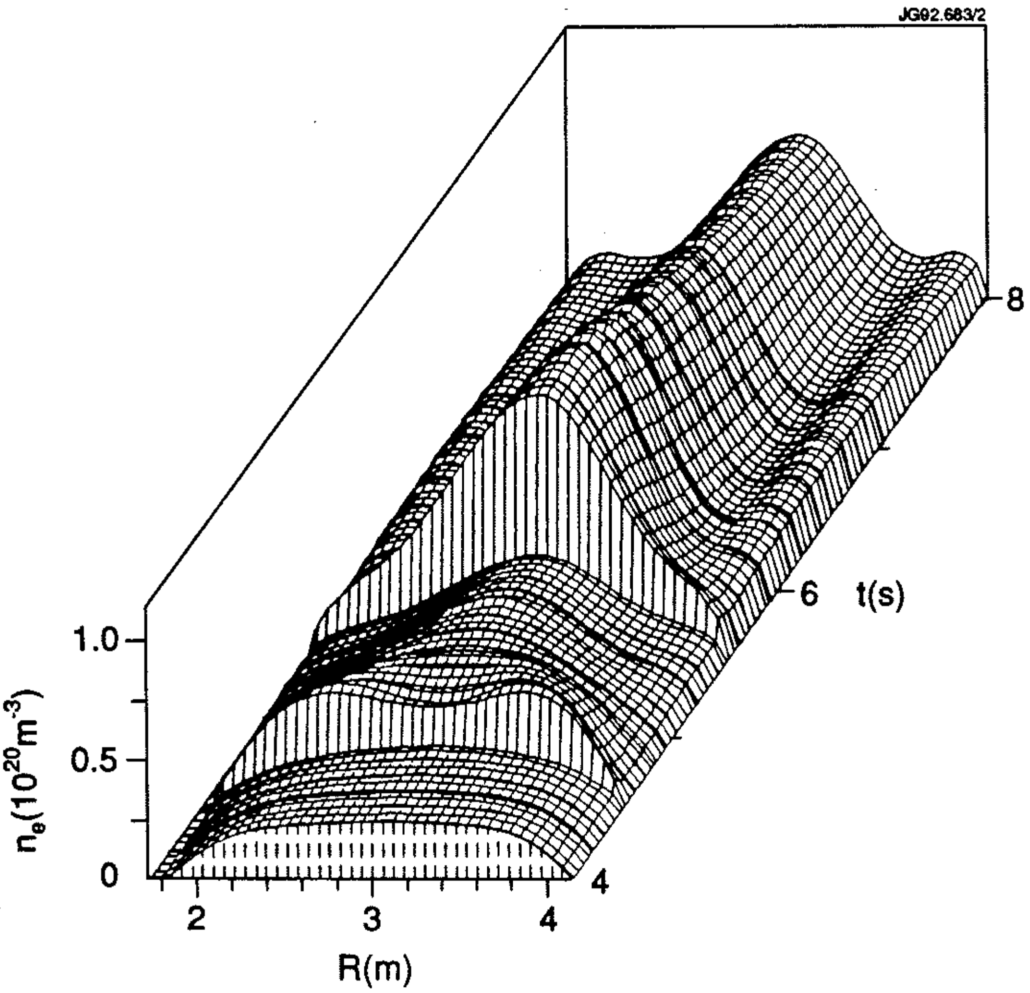

Fig. 7 Evolution of the density profile for the JET discharge #13572. A 4 mm pellet is injected at t ≈ 44.5 s and a 2.7 mm and a 4 mm pellet are injected at t ≈ 45.5 s. The discharge starts at t = 40 s.

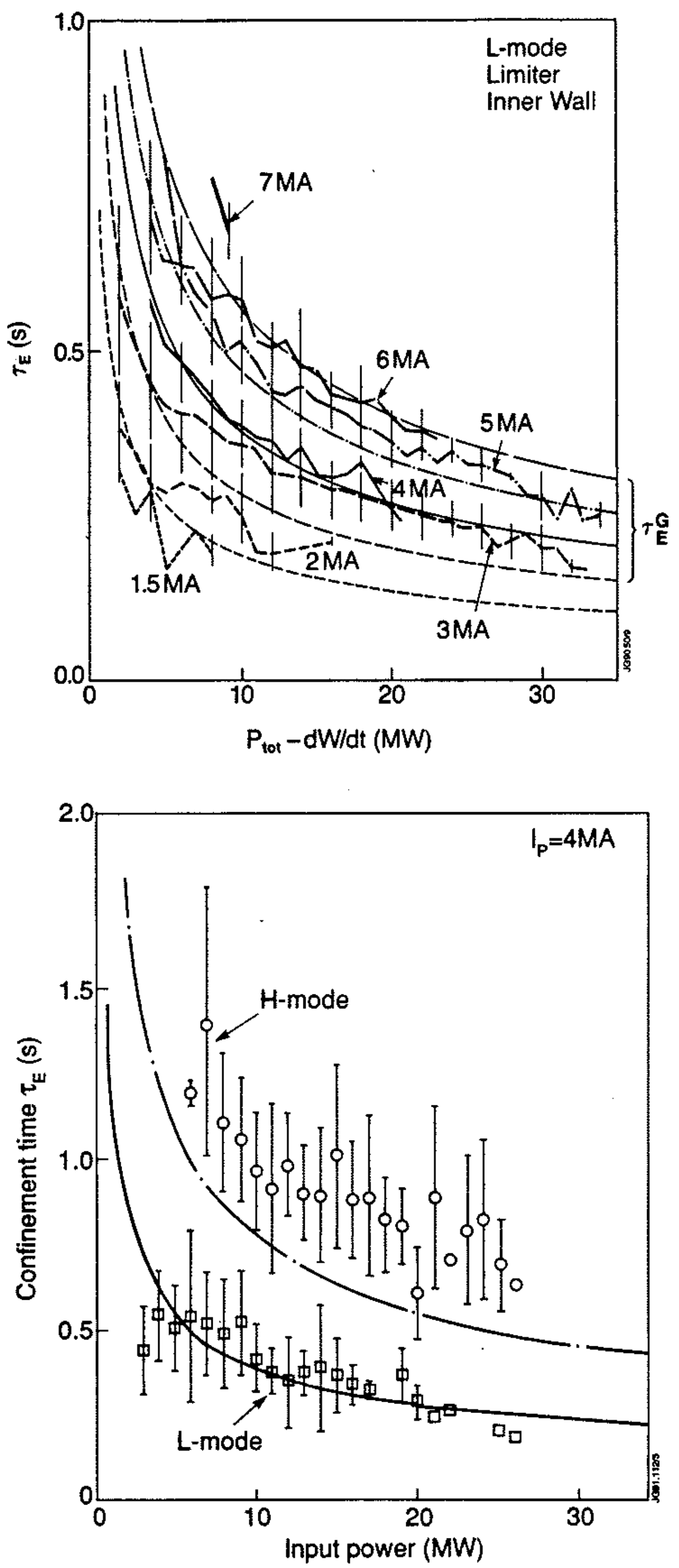

Fig. 8 Energy confinement times versus input power (minus rate of change in thermal plasma energy) content.

a) The smooth curves represent the L-mode scaling, the irregular curves are variations of energy confinement for different discharges with similar current. The vertical lines indicate the spread among different discharges in estimating the energy confinement time.

b) Comparison of energy confinement times for H- and L-modes during steady state for 4MA plasma currents.

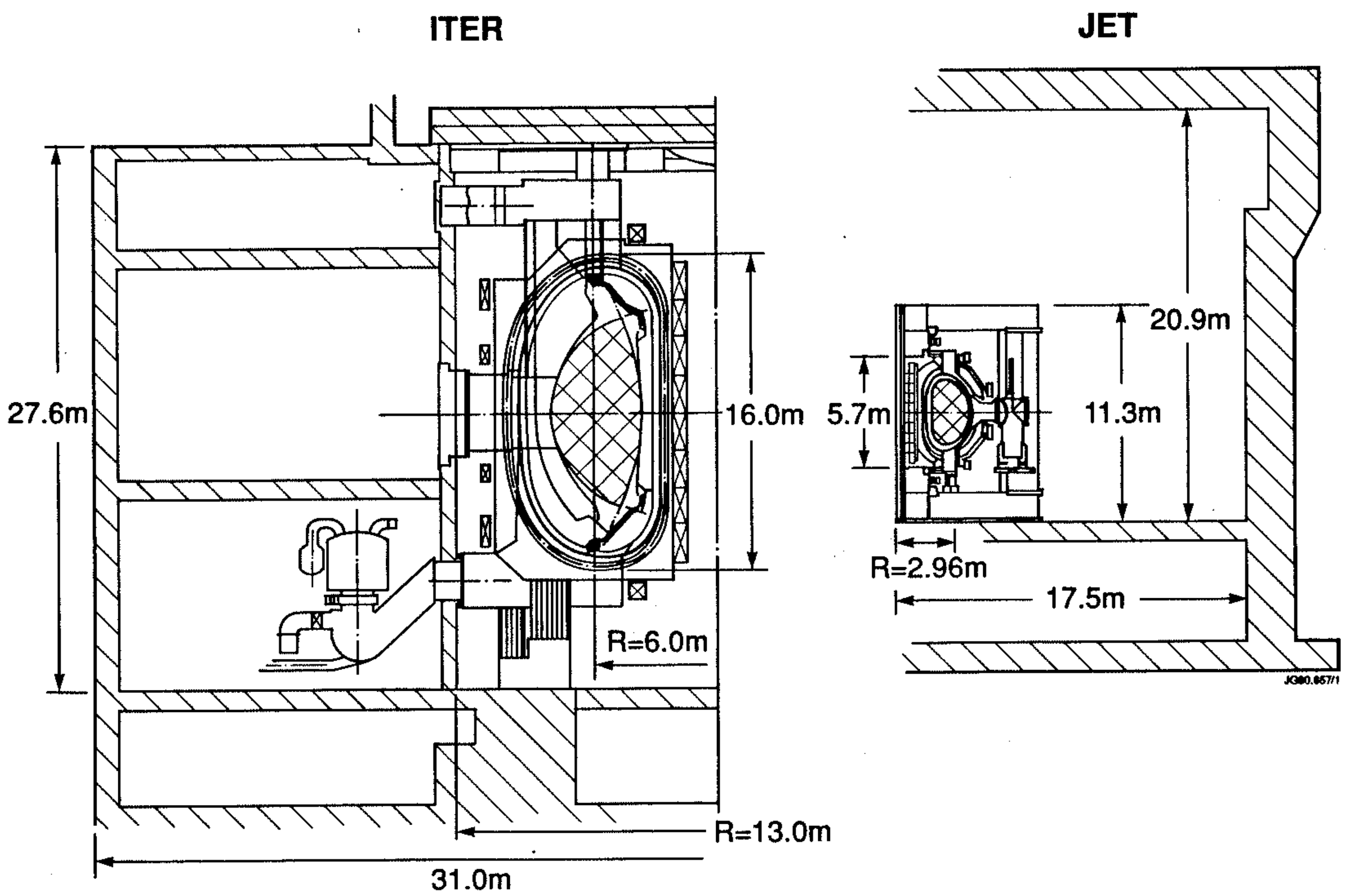

Fig. 9 Cross-section of the planned ITER tokamak in comparison with JET dimensions.

Appendix I

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