Magnetic Confinement Fusion: Recent Results at JET and Plans for the Future

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Abstract

The JET tokamak experiment has approached the plasma conditions needed in a thermonuclear reactor based on magnetic confinement concepts. In single deuterium discharges, breakeven has been achieved and, for the first time with deuterium-tritium fuels, ~1.7MW of fusion power was achieved in a 2s pulse. The total energy release was 2MJ. These results were obtained transiently, limited by an high impurity influx. For long pulse high power operation, plasma dilution has been identified as a major threat to a reactor. Improved impurity control in the pumped divertor configuration in a New Phase of JET (1992-1996) is envisaged. Experimental results support a plasma model based on a single phenomenon and MHD limits. Together, these are used to define the size and operating conditions of a reactor. A Next Step device would demonstrate the scientific feasibility of ignition under reactor conditions and this is discussed within the context of an international collaborative programme.

1. INTRODUCTION

The basic principle of the fusion process is the fusing of light nuclei to form heavier ones and the accompanying release of substantial energy. For a reactor, there are several possible fusion reactions, but the easiest to achieve is between deuterium and tritium. The D-T reaction is:

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

At the temperatures needed for this reaction to occur the D-T fuel is in the plasma state, comprising a mixture of charged particles (nuclei and electrons), which can be contained by magnetic fields. The most effective magnetic configuration is the toroidal tokamak device, of which JET is the largest in operation.

For a D-T fusion reactor, the triple product of the temperature (T_i), density (n_i) and energy confinement time (τ_E) must exceed the value ($n_i.\tau_E.T_i$) of $5x10^{21}m^{-3}skeV$. Typically, for magnetic confinement concepts, this requires:

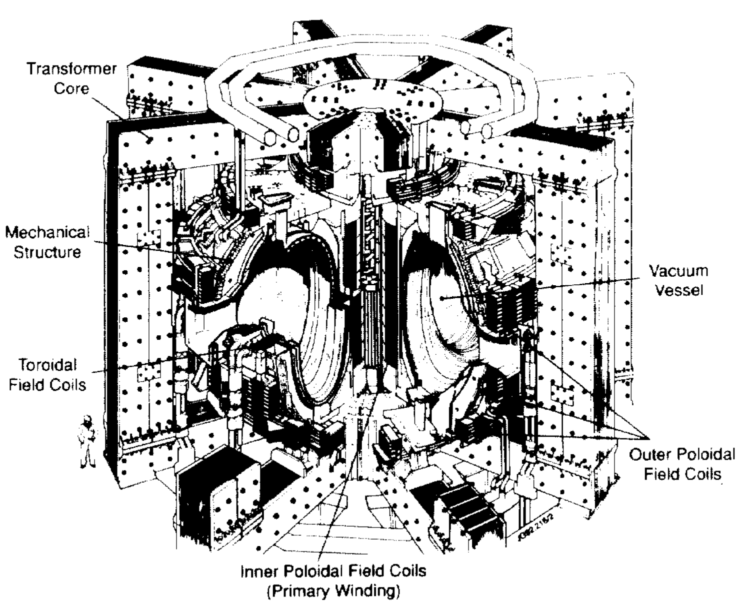
Central ion temperature, $T_i \sim$ 10-20keV
Central ion density, $n_i \sim$ $2\text{-}3x10^{20}m^{-3}$
Global confinement time, $\tau_E \sim$ 1 - 2s

During the early 1970's, it was clear that the achievement of near-reactor conditions required much larger machines, which were likely to be beyond the resources of individual countries. In 1973, it was decided in Europe that a large device, the Joint European Torus (JET), should be built as a joint venture.

JET is the largest project in the coordinated programme of EURATOM, whose fusion programme is designed to lead ultimately to construction of an energy producing reactor. Its strategy is based on sequential construction of major apparatus such as JET, a Next Step device, and a demonstration reactor, supported by medium sized specialized tokamaks.

The objective of JET is to obtain and study plasma in conditions approaching those needed in a thermonuclear reactor [1]. By mid-1983, the construction of JET was completed on schedule and the research programme started. To date, JET (Fig.1) has successfully contained plasmas of thermonuclear grade, and reached near breakeven conditions in single discharges. These results have also produced a clearer picture of energy and particle transport, resulting in the development of a particular model which describes and predicts plasma behaviour.

Furthermore, moderate extrapolation of latest results and considerations of model predictions allow the size and performance of



Parameter	Design Values	Achieved Values
Major Radius	2.96m	2.5-3.4m
Minor Radius (Horizontal)	1.25m	0.8-1.2m
Minor Radius (Vertical)	2.10m	0.8-2.1m
Toroidal Field	3.45T	3.45T
Plasma Current	4.8MA	7.0MA
NB Power	20MW	21MW
ICRF Power	15MW	22MW

Fig. 1: The JET Tokamak

a thermonuclear reactor to be largely defined. Most critical for a reactor is the control of impurities and the exhaust of helium ash at high power. To consolidate the model and provide further information on density and impurity control, a New Phase of JET is underway. A Next Step device will then bridge the gap from present knowledge to that required to construct a reactor.

2. THE BASIC TOKAMAK CONFIGURATION

The tokamak is the most advanced concept for containing magnetically a hot dense plasma [2]. A toroidal, axisymmetric plasma is confined by the combination of a large toroidal magnetic field, a smaller poloidal magnetic field (created by a toroidal current through the plasma) and the superposition of magnetic fields created by toroidal coils external to the plasma. The position and shape of the plasma cross-section is determined by the magnetic fields generated by these external coils.

The current circulating in the tokamak heats the plasma resistively. However, temperatures are expected to be limited below ignition by the decrease in resistivity with increasing temperature. Auxiliary heating is then required to reach higher temperatures; for example, injection of beams of high energy neutral particles (NB); and electromagnetic waves in different frequency ranges, such as ion cyclotron resonance heating (ICRH), and lower hybrid heating. In an ignited D-T plasma, collisional heating due to the thermalization of energetic alpha-particles will be dominant.

The heating effectiveness is determined by the thermal insulation of the plasma measured by the global energy confinement time, τ_E. Unfortunately, energy confinement is worse than would be expected on the basis of kinetic theory with binary collisions between particles (the so-called neo-classical theory) and a theoretical model for the anomalously poor insulation is needed. Empirical scaling laws for the energy confinement time have been

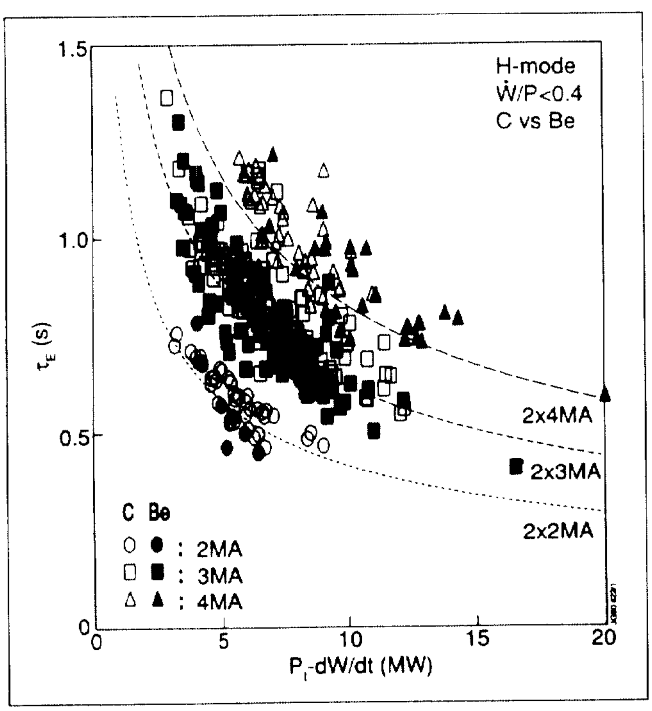


Fig. 2: Global energy confinement time (τ_E) during the H-mode as a function of net input power

derived on the basis of statistical fits to experimental data. The scalings which characterize discharges with additional heating (the low confinement or L-regime) are quite different from, and more pessimistic than, those for Ohmic heating alone. However, the expectations of L-regime scalings have been exceeded by up to a factor of about three in some regimes of plasma operation, the most notable of which is the H-regime (or high confinement mode).

The main methods of increasing the plasma density are: injection of cold gas, high energy neutral particles and frozen solid pellets. Since the central region shows better confinement than the overall plasma, central fuelling is highly desirable.

The plasma environment and the system chosen to define the plasma edge and to exhaust particles and energy is also important. The first-wall that the plasma encounters can be a copious source of impurities to cool and poison the hot plasma. Therefore, a careful choice of configuration and first-wall material must be made, as this determines the extent of the impurity problem. One option is a material limiter, in which a solid structure defines the plasma boundary. An alternative is a poloidal magnetic divertor (X-point magnetic configuration), in which the plasma boundary is defined by the transition between closed, nested magnetic surfaces and open magnetic field lines, which eventually intersect target plates away from the main plasma. A divertor can be considered as a limiter, remote from the plasma.

3. ACHIEVEMENTS OF THE JET PROGRAMME

3.1 The JET Tokamak

JET is a high current, high power tokamak with a low-Z first wall [3] (Fig.1). The technical design specifications of JET have been achieved in all parameters and exceeded in several cases. The plasma current of 7MA in the limiter configuration and the current duration of up to 60s at 2MA are world records. Neutral beam (NB) injection has been brought up to full power (~21MW) and ion cyclotron resonance heating (ICRH) power has also been increased to ~22MW in the plasma. In combination, these systems have delivered 36MW to the plasma.

3.2 Energy Confinement

JET can also operate with a magnetic limiter configuration, which is foreseen for a Next Step tokamak. In this configuration, a regime of higher energy confinement (H-mode) has been observed with confinement times exceeding twice the normal regime (L-mode). In both regimes, confinement improves with increasing current but degrades with increasing heating power (Fig. 2), independent of heating method. The plasma thermal energy does not increase in proportion to heating power, and considerably more power is needed to increase the plasma temperature and energy. Confinement is not affected by the impurity mix (carbon or beryllium in deuterium plasmas), but at high power levels, impurities become a problem to further improving plasma parameters.

3.3 Performance in Deuterium Plasmas

Improved plasma purity was achieved in JET using beryllium as a first-wall material, by sweeping the X-point and by using strong gas-puffing in the divertor region. This resulted in high central ion temperatures (the hot-ion mode with T_i~20-30keV) and improved plasma performance, with the fusion triple product ($n_i.\tau_E.T_i$) increasing significantly. Such improved fusion performance could

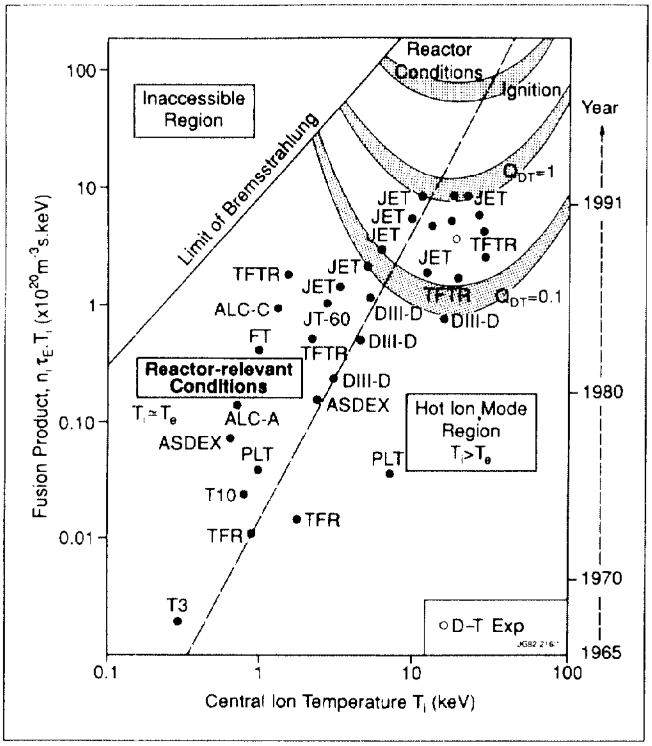


Fig. 3: Overall performance of the fusion product as a function of central ion temperature for a number of tokamaks

otherwise have been achieved only with a substantial increase in energy confinement.

In a deuterium hot-ion H-mode plasma, T_i reached 19keV, τ_E, was 1.2s, with a record fusion product ($n_i.\tau_E.T_i$) of 9×10^{20}m^{-3}skeV, and the neutron yield was the highest achieved at 4.3×10^{16} ns^{-1}. Simulation for a D-T mixture showed that 11MW of fusion power would have been obtained transiently with 15MW of NB power. This would have reached near breakeven conditions, within a factor 6 of that required for a reactor. Similar results with ICRH were obtained at medium temperatures, with T_e~T_i~10keV.

The overall fusion triple product as a function of ion temperature is shown in Fig.3 for JET and a number of other tokamaks.

3.4 Performance in Deuterium-Tritium Mixtures

Towards the end of 1991, the performance of JET plasmas had improved sufficiently to warrant the first tokamak experiments using a deuterium-tritium (D-T) fuel mixture [4]. Tritium neutral beams were injected into a deuterium plasma, heated by deuterium neutral beams (Fig.4). This introduced ~10% tritium into the machine, and a significant amount of power was obtained from controlled nuclear fusion reactions. The peak fusion power generated ~1.7MW in a high power pulse lasting for 2 seconds, giving a total energy release of 2MJ. This was clearly a major step forward in the development of fusion as a new source of energy.

4. IMPURITY CONTROL IN JET AND A REACTOR

4.1 The JET Experience

Plasma dilution is a major threat to a reactor and impurity control under high power conditions has always been considered a key scientific and technical issue. Impurity production has been reduced both **passively** (by proper choice of plasma-facing

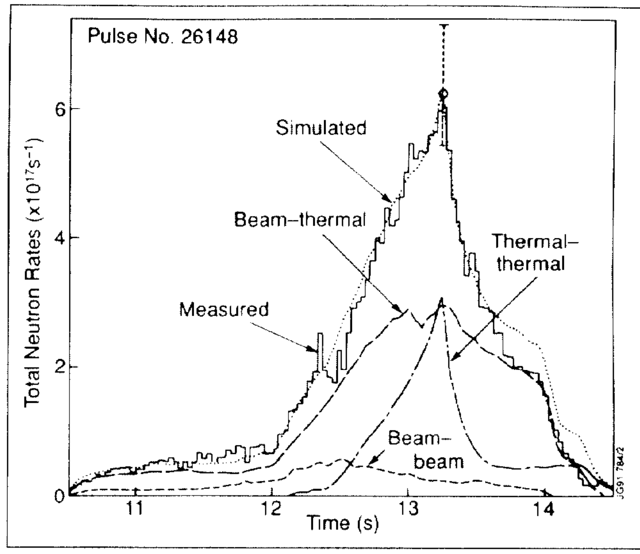


Fig.4: Experimental measurements and simulations of the total neutron rates (mainly 14MeV neutrons) for a D-T pulse in JET

components) and **actively** (by sweeping the plasma across the targets where the interaction is often localised).

Up to 1988, JET operated with a carbon first wall (carbon tiles and wall carbonisation). The attainment of high plasma performance was limited by impurity influxes, mostly carbon and oxygen, from the walls. The impurities diluted the plasma fuel, decreasing the fusion reactivity and increasing radiative energy losses. Excessively high impurity influxes were observed during high power heating and led to a rapid deterioration of fusion performance.

From 1989, JET operated with a beryllium first-wall. Due to its low atomic number, beryllium was expected to lead to superior plasma performance, resulting in much reduced radiative losses compared with carbon. It also has the advantage of acting as a getter for oxygen. Subsequently, experimental campaigns confirmed these expectations. The chief effect of beryllium is to improve plasma purity and, as a result, to increase plasma performance.

However, as in all high performance discharges, the high power phase is transient, lasting for less than 1s. It could not be sustained in the steady state: the impurity influx observed with carbon walls also occurs with beryllium and causes a degradation of plasma parameters. This emphasises the need for improved methods of impurity control in fusion devices.

4.2 The New Phase of JET

Early in 1992, a New Phase began [5] with the aim of demonstrating effective methods of impurity control in operating conditions close to those of a Next Step with a stationary plasma of 'thermonuclear grade' in an axisymmetric pumped divertor configuration. Specifically, the New Phase should demonstrate: control of impurities generated at the divertor targets; decrease of heat load on the targets; control of plasma density; an exhaust capability; and a realistic model of particle transport.

First results should be available in 1993 and the Project will continue to end of 1996. Overall, the results should allow determination of the size and geometry needed to realise impurity control in a Next-Step; allow a choice of suitable plasma facing components; and demonstrate the operational domain for such a device.

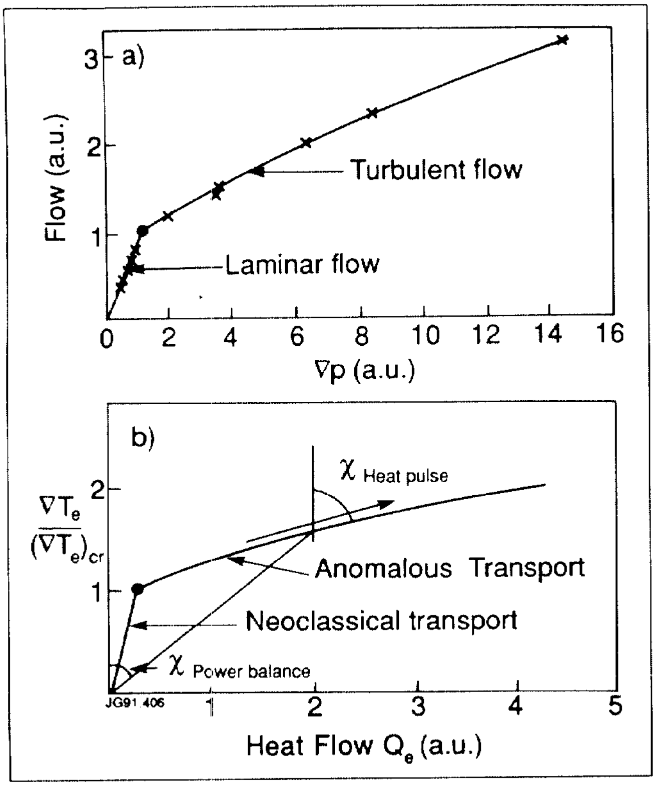


Fig.5: (a) the relationship governing liquid flow through a pipe. When the Reynolds number, R, reaches a critical value, R_c, extra resistance restricts the flow which increases with the value of $\check{R}$ (curvature of the curve in the turbulent flow regime); (b) for given temperature, density, magnetic field, etc., the dependence of the heat flow with electron temperature gradient in the critical temperature gradient model shows the same behaviour: when ∇T reaches $(\nabla T)_c$, anomalous transport appears which increases the heat flow. This anomalous transport also varies non-linearly with the ratio $(\nabla T_e)/(\nabla T_e)_{cr}$.

5. A TRANSPORT MODEL

5.1 Formulation of a Plasma Model

Explaining the anomalous transport in tokamaks by the presence of turbulence is widely accepted. An analogy with turbulence in fluid mechanics can be developed [6]. The dimensionless Reynolds number, R, can be constructed from the physics quantities entering the Navier-Stokes equation and turbulence develops when R exceeds a critical value, R_C. Experimental data show that the laws ruling the flow change when the pressure gradient, i.e. the driving force, is such that $R>R_C$ (see Fig.5(a)).

Such a change in the energy and particle flows is also observed in a tokamak. Since the tokamak is an open thermodynamic system, heat flow could influence its stability. To conform with thermodynamics, the driving force for the heat transport in steady state should be the temperature gradient. The equivalent of the Reynolds number in fluid mechanics should be the ratio, $\nabla T/(\nabla T)_{cr}$ - a threshold value above which turbulence develops and the heat transport is enhanced. This behaviour is illustrated in Fig.5(b). With ohmic heating alone the temperature gradient should limit itself to a value close to, but above, the onset of turbulence. In the presence of powerful additional heating the confinement properties should be entirely controlled by the anomalous thermal transport. Universal to all turbulent phenomena is a non-linearity between the flow and driving force and a delay in the onset and propagation of turbulence.

It is certainly not unreasonable to assume that magnetic confinement is affected by magnetic turbulence. In particular, macroscopic changes in the magnetic topology seem to be responsible for the total loss of confinement observed during major plasma disruptions. An attractive hypothesis is that a single basis, the magnetic topology, underlies the various phenomena observed in a tokamak, at least where atomic physics does not play a role. Tokamak physics would then be dominated by tearing and micro-tearing modes. The topology would consist not only of well-nested magnetic surfaces but also of small magnetic islands surrounded by chaotic field lines connecting radially hot and cold regions.

Experimental observations support a model for anomalous transport based on a single phenomenon and MHD limits. This **Critical Electron Temperature Gradient** model of anomalous heat and particle transport features: electrons which determine the degree of confinement degradation; ion anomalous transport with heat diffusivity χ_i linked to electron heat diffusivity χ_e; anomalous particle diffusivities, D, for ions and electrons, proportional to χ; and anomalous inward particle convection.

Specifically, above a critical threshold, $(\nabla T_e)_{cr}$, in the electron temperature gradient, the transport is anomalous and greater than the underlying neoclassical transport. The electrons are primarily responsible for the anomalous transport, but ion heat and particle transport are also anomalous. The general expressions for the anomalous conductive heat fluxes have been specified [7].

Plasma profiles under various conditions and in several tokamaks are well described by the model which also exhibits the following experimental features: consistency with physics constraints, global scaling laws and statistical analysis; a limitation in the electron temperature; no intrinsic degradation of ion confinement with ion heating power; similar behaviour of particle and heat transport; the propagation of heat and density pulses; and the same transport model in the plasma interior for L and H-regimes.

With such a model, we begin to have the predictive capability needed to define the parameters and operating conditions of a reactor, including impurity levels.

6. A FIRST REACTOR

6.1 Definition of a First Reactor

A first reactor will be a full ignition, high power device (1-2GW electrical). This will include: auxiliary heating; D-T fuelling; divertor with high power handling and low erosion; exhaust for impurities and helium ash; first wall with high resilience to 14MeV neutrons; hot blanket to breed tritium; and plasma control.

The parameters of a first reactor are defined by technology and physics predictions. The plasma minor radius must be twice the thickness of the tritium breeding blanket (~3m) and the elongation can be ~2. A practical aspect ratio (~2.5) sets the major radius at 8-9m. Safe operation can be assumed for a cylindrical safety factor 1.6-1.8. Plasma physics requirements can be fulfilled with a toroidal field ≤5T. This defines a plasma current of ~25-30MA.

The reactor will operate with: $T_i \sim T_e$; confinement in L-mode; high density, relatively flat density profile; and impurity control.

6.2 Application of the Model to a First Reactor

Reactor plasmas using the L-mode transport model have been tested against JET results. A D-T mixture is assumed and helium ash accumulation and pumping is included. Assuming the provision

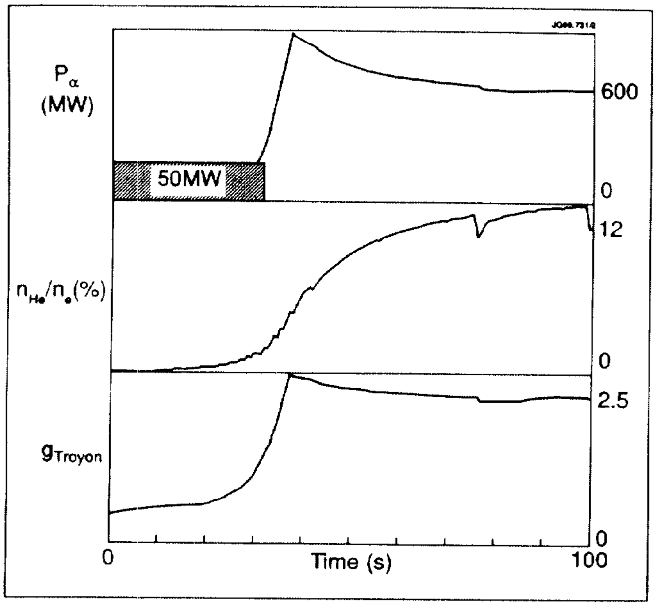


Fig.6: Model of a first reactor plasma using the L-mode transport model tested against JET results

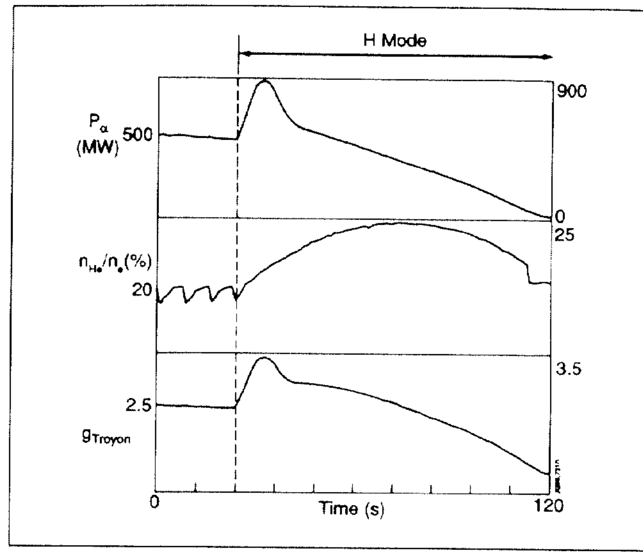


Fig.7: Long term deficiencies due to helium poisoning in the H-mode

of adequate impurity control, ignition is maintained in this reactor (R=8m, a=3m, B_T=4.5T, I_p=30MA, κ=2) after switching off 50MW of ICRH (Fig.6). At ignition, the slightly hollow density profiles with edge fuelling are sufficient to fuel the centre. However, helium poisoning alone can quench the ignition without adequate pumping. To exhaust sufficient helium and maintain ignition requires pumping ~0.3g of D-T per second. Exhaust and impurity control are essential. In fact, while the H-mode has short term benefits for approaching ignition, the long term deficiencies due to helium poisoning are evident (see Fig.7). Steady ignition conditions can be achieved with a specific level of helium ash.

With impurity control, ignition is achieved with edge fuelling and high pumping; high density and relatively flat profile; L-mode confinement; and recirculation of ~0.3g of D - T per second.

7. THE NEXT STEP: A REACTOR CORE

The overall aim of a Next Step would be to demonstrate fusion as an energy source. Therefore, it should: demonstrate sustained high power operation (1-3GW thermal power); operate semi-continuously (~1/2 hour); study the ignition domain; study operating conditions; define first wall technology; define exhaust and fuelling requirements; provide a testbed for the study and validation of tritium breeding blanket modules in reactor conditions; achieve a cost/unit thermal output relevant to the establishment of fusion as a potential economic energy source; achieve a high level of safety and have minimum effect on the environment.

These objectives could be achieved in a tokamak with I_p up to 25MA, B_T≈5T, major radius ≈7m, minor radius ≈3m, and elongation of 1.6. Impurities would be controlled actively by high density operation and a pumped divertor. The approach to ignition would utilise ICRH with H-mode confinement, while long pulse ignition (~1/2 hr) would be sustained with X-point L-mode confinement at high power. With sustained ignition conditions, blanket modules could be tested under neutron fluxes of up to 2MWm^{-2}. In addition, advanced divertors and concept development aimed at improved efficiency must also be pursued.

International collaboration, such as the proposed ITER (International Thermonuclear Experimental Reactor) programme is the way to address the different Next Step issues. Several complementary facilities, including a 14MeV neutron source, each with separate, clearly defined objectives, would:

- reduce scientific and technological risks;
- allow flexibility to accommodate new concepts;
- provide a wider and more comprehensive data base and allow cross-checking of results;
- be more practical in managerial and industrial terms;
- offer flexibility in location and time scheduling.

This programme, with ITER as the first component, would minimise risks and overall costs and would ensure efficient use of world resources. In support, National Programmes comparable in size would be needed.

With concerted effort and determined international collaboration, world resources exist to proceed with such a programme, towards a Demonstration Reactor starting operation in ~2015.

8. ACKNOWLEDGEMENTS

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