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Abstract

Controlled thermonuclear fusion is potentially a major vast new energy source. A reactor based on nuclear fusion would be inherently safe, environmentally friendly, and fuels are cheap, abundant and widely available. 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 nuclei and the accompanying release of substantial energy. Controlled thermonuclear fusion is potentially a major vast new energy source suited to the industrialised world. A reactor based on nuclear fusion would be inherently safe and environmentally friendly. Furthermore, fuels are cheap, abundant and widely available.

A fusion reactor would be:

- **safe** because it would operate at low pressure, low fuel inventory and maximum reactivity. No significant surges of power would be possible since the fuel in the reactor at any time would be sufficient for only 30s of operation.
- **environmentally friendly** because it would not pollute the atmosphere and it would avoid the problems (such as "greenhouse" gases and acid rain) associated with the burning of fossil fuels. The hazard potential would be limited since there would be no long lived radioactive waste from the reaction products, including the fuel "ash". Radioactivity in the structure of the reactor would be minimised by careful selection of low-activation materials.
- **using fuels in abundant supply** because the principal fuel, deuterium, is readily available - it can be extracted from water - and supplies would last for millions of years. The first type of reactor would also use tritium which does not occur naturally but can be manufactured from

lithium in the reactor itself. Lithium reserves are plentiful in the Earth's crust and could last for more than a thousand years.

Two approaches towards controlled thermonuclear fusion are being pursued at present - one based on inertial confinement and the other based on magnetic confinement. A major step in the fusion programme would be the construction of the core of a first reactor. The magnetic confinement programme is now ready to take this step - ITER - with the most advanced concept for magnetic confinement, namely the toroidal tokamak configuration, of which JET is the largest in operation [1].

2. CONTROLLED THERMONUCLEAR FUSION

2.1 Basic Principles

The basic principle of the fusion process is the fusing or joining together of light nuclei to form heavier ones and in so doing a small quantity of mass is converted into a large amount of energy. Fusion is the process occurring in the sun where light atoms, heated to temperatures of about 15M°C (~1.5keV) fuse together. The rate at which fusion occurs in the sun is relatively slow; for a fusion reactor on earth, a higher rate is required and hence much higher temperatures are needed - typically in the range of 100-200 M°C (10-20keV), which is 10 times greater than the temperature in the centre of the Sun. In addition, in a reactor, a high enough concentration (or density) of fuel must be maintained at these temperatures for sufficient periods. For a reactor, there are several possible fusion reactions, but the one that is easiest to achieve is that between the two isotopes of hydrogen - deuterium and tritium. Deuterium can be easily and cheaply obtained from water and tritium can be manufactured in a fusion reactor from the light metal lithium. The reactions involved are:

Fusion Reaction	$D + T \rightarrow {}^4He + n$
Tritium Maunfacturing Reactions	${}^6Li + n \rightarrow T + {}^3He$
	${}^7Li + n \rightarrow T + {}^4He + n$

2.2 Magnetic Confinement

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). In a reactor, there must be sufficient fuel present and the energy losses must be kept low to ensure that more energy is released from the fusion reaction than is needed to heat the fuel and run the system. The plasma nuclei can be contained by gravitational forces, as in the sun, or by magnetic fields. For magnetic confinement, the effectiveness of the magnetic field in containing plasma and minimising losses can be measured by the time taken for the plasma to cool down. This is called the energy confinement time and needs to be between one and two seconds in a reactor - although the plasma will be contained for considerably longer. The power output depends on the amount (or density) of fuel present, but there is a limit on the amount that can be contained by the magnetic field. This is only a few thousandths of a gram per cubic metre, but this is sufficient to yield vast amounts of energy. Thus a fusion reactor must produce very high temperature plasmas of sufficient density and long enough energy confinement time to generate a net output of power.

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

3. THE TOKAMAK APPROACH TO MAGNETIC CONFINEMENT

3.1 The Tokamak Concept

The tokamak is the most advanced concept for containing magnetically a hot dense plasma [2]. It originated in the USSR and JET is now the largest device in operation. 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, the injection of beams of high energy neutral particles; 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 energy confinement time. 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 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: edge fuelling by the injection of cold gas and low speed frozen solid pellets; and central fuelling by high energy neutral particles and high speed frozen solid pellets.

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.

3.2 The Tokamak Programme towards a Reactor

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 the construction of an energy producing reactor. Its strategy is based on the 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

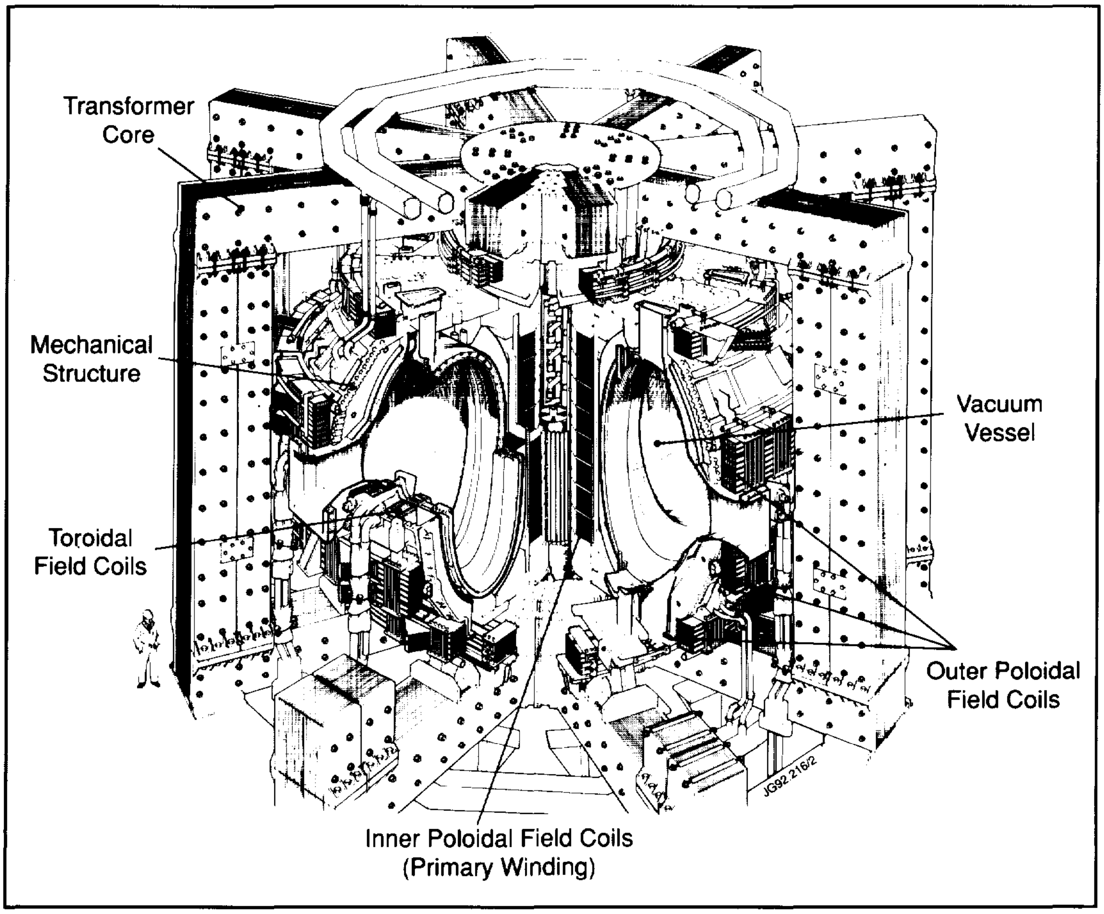

Fig. 1: The JET Tokamak

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 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.

4. STATUS OF TOKAMAK RESEARCH

4.1 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 (Table I). 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. JET can also operate with a magnetic limiter configuration, which is foreseen for a Next Step tokamak. In this configuration, the regime of higher energy confinement (H-mode) has been observed with confinement times exceeding twice those of the normal L-mode regime.

Table I

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

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

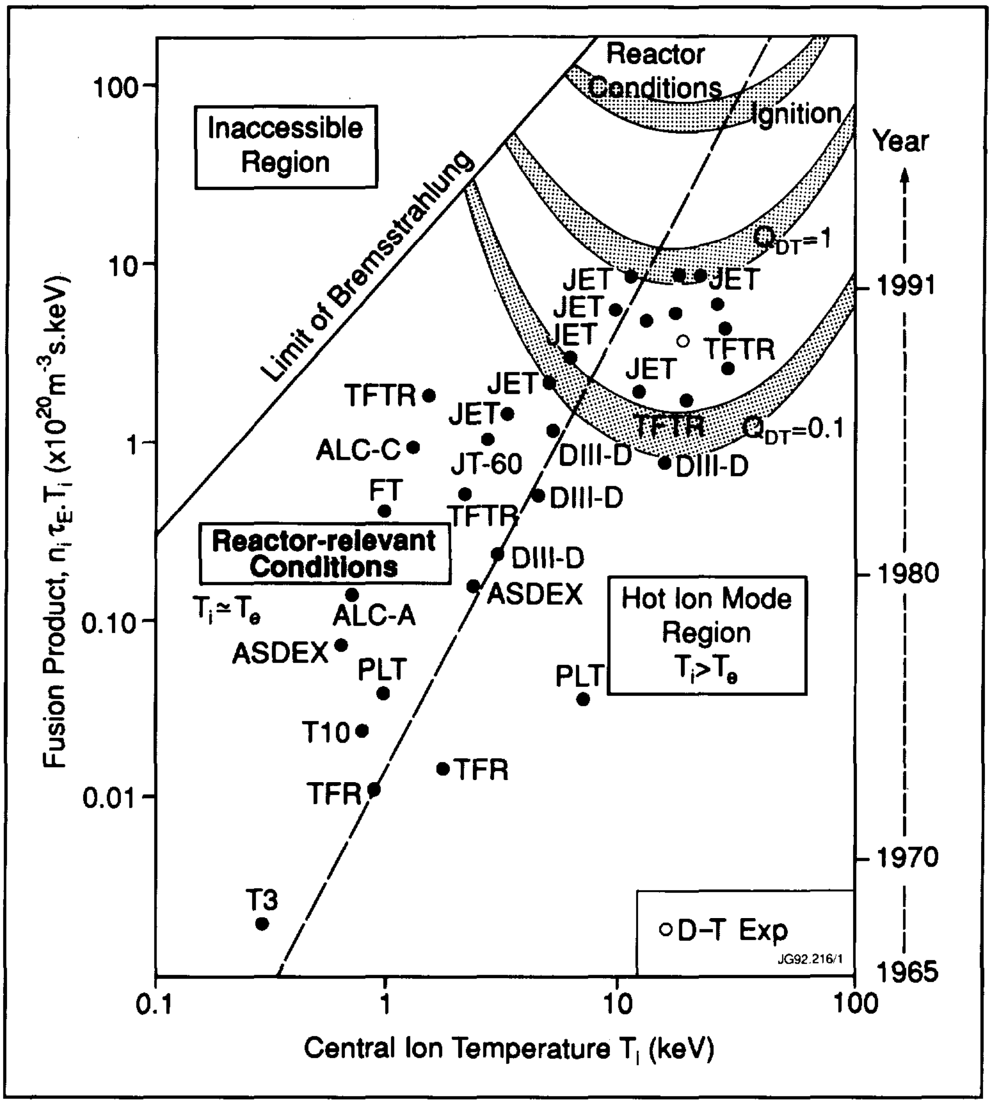

Fig. 2: Overall performance of the fusion product as a function of central ion temperature for a number of tokamaks

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 $9x10^{20}m^{-3}$skeV, and the neutron yield was the highest achieved at $4.3x10^{16}$ 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 been near breakeven and 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.2 for JET and a number of other tokamaks.

4.3 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.3). This introduced ~10% tritium into the machine, and a significant amount of power was obtained from controlled nuclear

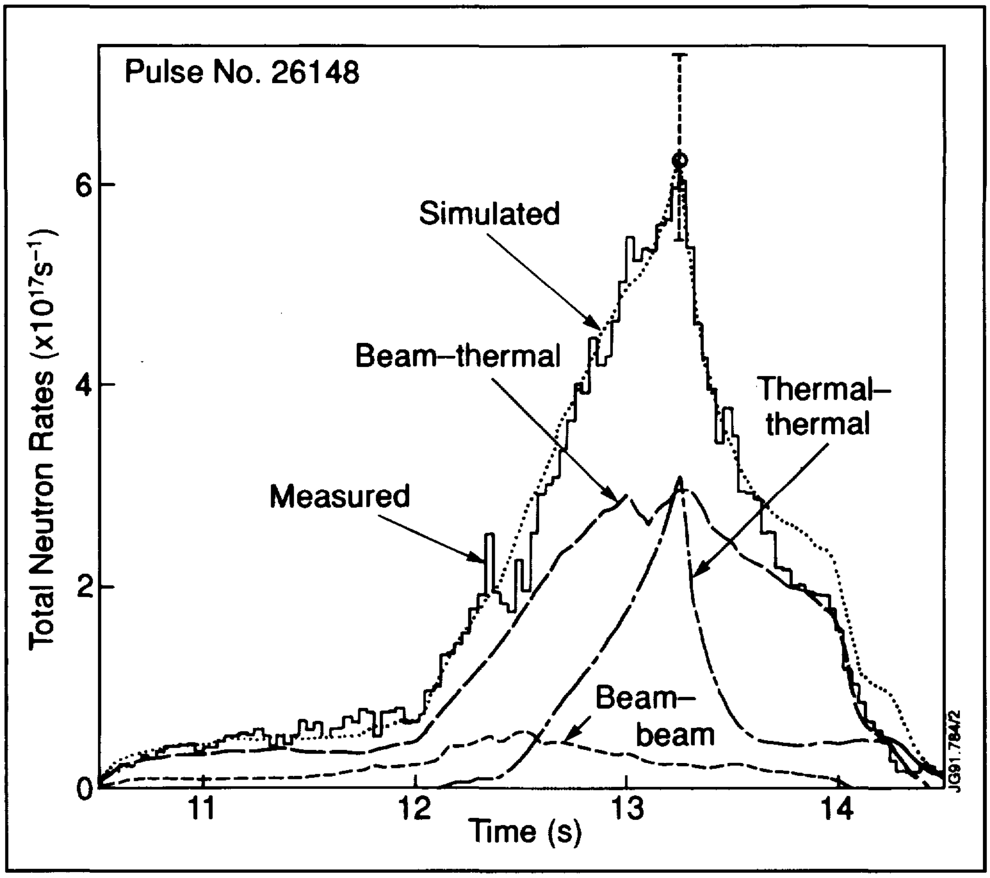

Fig.3: Experimental measurements and simulations of the total neutron rates (mainly 14MeV neutrons) for a D-T pulse in JET

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.

4.4 Impurity Control in JET and a Reactor

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 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.

4.5 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.

4.6 A Plasma Model

Explaining anomalous transport in tokamaks by the presence of turbulence is widely accepted. An analogy with turbulence in fluid mechanics can be developed [6]. It is certainly not unreasonable to assume that magnetic confinement is affected by magnetic turbulence. An attractive hypothesis is that a single basis, the magnetic topology, underlies the various phenomena observed in a tokamak. Experimental observations support a model for anomalous transport based on a single phenomenon - the Critical **Electron Temperature Gradient** model. Specifically, above a critical threshold 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. 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.

5. A FIRST REACTOR

5.1 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 (a~3m) and the elongation, k~2. A practical aspect ratio (~2.5) sets the major radius, R~8-9m. Plasma physics requirements can be fulfilled with a toroidal field ≤5T. This defines a plasma current of ~25-30MA.

The reactor will operate with: Ti ~ Te; confinement in L-mode; high density, relatively flat density profile; and impurity control.

5.2 Application of the Model to a First Reactor

The L-mode model of Section 4.6 has been tested against JET results and reactor plasmas have been simulated. Assuming the provision of adequate impurity control, ignition is maintained in a D-T reactor (R=8m, a=3m, BT=4.5T, Ip=30MA, k=2) after switching off 50MW of ICRH [7]. At ignition, the density profile is slightly hollow and edge fuelling is sufficient to fuel the centre. However, the accumulation of reaction products (helium ash) in the plasma centre can quench the ignition unless the ash can be transported to the plasma edge where it can be pumped. 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

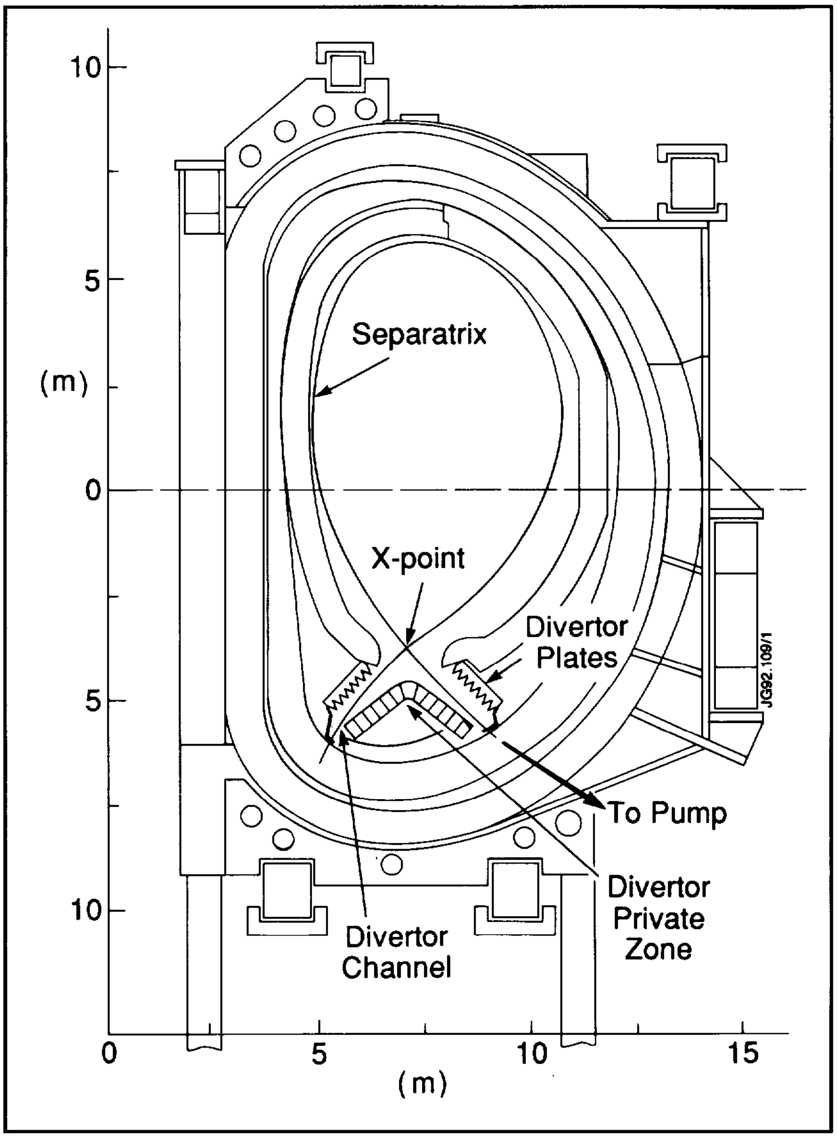

Fig.4: A possible 'Next Step' configuration

deficiencies due to helium accumulation are evident. Steady ignition conditions can be achieved with a specific level of helium ash.

6. 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, $B_T \approx 5T$, major radius ≈7m, minor radius ≈3m, and elongation of 1.6 (Fig.4). 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 be pursued.

7. WORLD COLLABORATIVE PROGRAMME

International collaboration is the way to address the different Next Step issues. The proposed ITER (International Thermonuclear Experimental Reactor) programme involves the four partners, Europe, Japan, Russian Federation and USA. 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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ANNEX

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