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Present State of Nuclear Fusion Research and Prospects for the Future

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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 bred 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 on magnetic confinement. A major step in the fusion programme would be the construction of the core of a first reactor. With the start of the Engineering Design Activities (EDA) of the International Thermonuclear Experimental Reactor (ITER) [1], the magnetic confinement programme is preparing to take this step with the most advanced concept for magnetic confinement, namely the toroidal tokamak, of which the Joint European Torus (JET) is the largest in operation [2].

2. CONTROLLED THERMONUCLEAR FUSION

2.1 Basic Principles

The basic principle of nuclear fusion is the fusing together of light nuclei to form heavier ones and in this process 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. At these temperatures, the rate at which fusion occurs 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 hotter 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, in principle, several possible fusion reactions, but the one that is easiest to achieve is that between the two isotopes of hydrogen - deuterium (D) and tritium (T) (Fig.1). Deuterium can be easily and cheaply obtained from water and tritium can be bred in a fusion reactor from the light metal lithium. The reactions involved are:

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

2.2 Magnetic Confinement

At the temperatures needed for the D-T 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 sufficiently low to ensure that more energy is released from the fusion reaction than is needed to heat the fuel and maintain the necessary temperature. 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 thermal losses can be measured by the time taken for the plasma to cool down after the source of heat is removed. 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, which is only a few thousandths of a gram per cubic metre, but is sufficient to yield vast amounts of energy. Thus a fusion reactor must produce high temperature plasmas of sufficient density that can be contained for long enough to generate a net output of power.

In fact, 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	~	10-20keV
Central ion density,	n_i	~	$2\text{-}3x10^{20}m^{-3}$
Global energy confinement time,	τ_E	~	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 [3]. It originated in the USSR and JET is the largest device in operation. A toroidal, axisymmetric plasma is confined by the combination of a large toroidal magnetic field, and a smaller poloidal magnetic field (created by a toroidal current through the plasma). The position and shape of the plasma cross-section is determined by the magnetic fields generated by the superposition of magnetic fields created by toroidal coils external to the plasma (Fig.2).

The current circulating in the tokamak heats the plasma resistively. However, temperatures are limited by the decrease in resistivity of the plasma 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 and lower hybrid heating. In a fusion reactor, collisional heating is dominant as the high energy alpha-particles (^{4}He) created during the fusion process thermalize with the background plasma.

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 resistive (or 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: shallower fuelling by the injection of cold gas and low speed frozen solid pellets; and deeper 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 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 [2].

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 smaller specialized devices.

The objective of JET is to obtain and study plasma in conditions approaching those needed in a thermonuclear reactor. By mid-1983, the construction of JET was completed on schedule and the research programme started. To date, JET (Fig.3) 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, providing the basis for a model to describe and predict 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 fuel "ash" at high power. A New Phase of JET is underway to provide further information on density and impurity control. A Next Step device, such as ITER, will then bridge the gap from present knowledge to that required to construct a demonstration 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 [4] (Fig.3). 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 a power of ~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 ITER. In this configuration, the regime of higher energy confinement (H-mode) has been observed with confinement times exceeding twice those of the 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
ICRH Power	15MW	22MW

4.2 Performance in Deuterium Plasmas

Improved plasma purity was achieved in JET using beryllium as a first-wall material [4], by using strong gas-puffing and by sweeping the magnetic configuration over the target plates 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.

During 1991, JET achieved, in a deuterium hot-ion H-mode plasma, $T_i \approx 19$keV, $\tau_E \approx 1.2$s, and a record fusion product ($n_i.\tau_E.T_i$) of $9 \times 10^{20} m^{-3}$skeV. The neutron yield was the highest achieved at 4.3×10^{16} ns^{-1}. Simulation showed that if this discharge had operated with a D-T mixture (rather than solely with deuterium) 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.4 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 D-T fuel mixture [5]. Tritium neutral beams were injected into a deuterium plasma, heated by deuterium neutral beams. 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 was ~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

Fuel 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 target plates 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 [6] with the aim of establishing, in deuterium plasmas, reliable methods of plasma purity control under conditions relevant for the Next Step tokamak and to undertake preparations for the final phase of JET with deuterium-tritium plasmas. Specifically, the New Phase should demonstrate in an axisymmetric pumped divertor: control of impurities generated at the divertor targets; decrease of heat load on the targets; control of plasma density; and an exhaust capability.

First results should be available in early-1994 and the New Phase will continue to the end of 1996. Overall, the results should allow determination of the size and geometry needed to realise impurity control in a Next Step tokamak such as ITER; allow a choice of suitable plasma facing components; and demonstrate the operational domain for such a device.

5. TOWARDS A TOKAMAK REACTOR

In spite of recent scientific advances, the extrapolation from present large tokamaks to a demonstration reactor still needs an extensive test of concepts and technologies required for the design and construction of such a device. The Next Step in the Fusion Programme must be a major element in fulfilling these needs. It should be a full ignition, high power tokamak producing large amounts of power and this must be accommodated by its various components. The Next Step will spend most of its lifetime operating with a D-T fuel and components will be subjected to high energy neutron fluxes. Consequently, the device and its components must be as simple and robust as possible and must demonstrate highly reliable and safe operation throughout their lifetime. The overall aim of a Next Step tokamak must be to demonstrate fusion as a primary energy source for the future.

The next phase of worldwide collaboration, the ITER Engineering Design Activities (EDA), has just begun. The International Thermonuclear Experimental Reactor (ITER) Project, started in 1987, represents the first step in worldwide collaboration between the EC, Japan, Russia and USA, under the auspices of the International Atomic Energy Agency. The main objective of ITER is to demonstrate the scientific and technological feasibility of fusion [1]. The Definition and Conceptual Design Activities were completed in December 1990,and the four partners signed an agreement on the Engineering Design Activities (EDA) in July 1992.

The main challenge to ITER is to address successfully the critical issues related to a demonstration reactor, namely:

- a physics understanding of the approach to ignition, the control of ignition (burn control) and burn products (helium "ash");
- whether a reactor should be operated continuously or semi-continuously;
- a divertor configuration which achieves high power exhaust, impurity control, high helium pumping and the rapid recirculation of several grammes of tritium per second;

- the minimisation of the effect of disruptive plasma instabilities;
- the development of the technology required for advanced materials and components needed for the first wall and blanket, the mechanical structure and superconducting coils;
- the development of advanced technology for fast and reliable maintenance in active conditions;
- the integration of these different reactor issues into a coherent and cost effective design.

5.1. Operating conditions of a Tokamak fusion reactor

This issue relates to the prediction of the size, toroidal field, plasma current and operating conditions of the plasma core of a first reactor (Fig.5). Extrapolation of latest results and considerations of model predictions, taken together with the constraints of present technology, allow the size and performance of a thermonuclear reactor to be largely defined [7].

The minor radius of the reactor plasma, a, needs to be about twice the thickness of the tritium breeding blanket, which makes it approximately 3m. A practical aspect ratio, (R/a), of between 2.5 and 3 sets the plasma major radius, R, to 8 or 9m. The elongation of the plasma must be limited to a value less than two. Safe operation can be assumed for a cylindrical safety factor greater than 1.6. Plasma physics requirements can be fulfilled by operating at a toroidal magnetic field of about 6T. This defines a reactor with a current capability of about 30MA. The total magnetic flux available could be about 1000Wb. The reactor would operate with a D-T mixture, and helium "ash". Impurity control would be achieved by plasma flows in an appropriate divertor configuration. The reactor plasma would most likely be characterised by a temperature of 25keV and a density greater than $10^{20}m^{-3}$.

5.2. Continuous or semi-continuous operation of a reactor

An important issue is whether a reactor should be operated continuously or semi-continuously [8]. Continuous operation with non-inductive current drive would appear, at first sight, to be preferable since component fatigue due to thermal cycling would be reduced and continuous power output would be more acceptable for the Power Generating Utilities. However, the provision of an external current drive plant, operating full-time, would make the construction and operation of such a reactor more complex and would increase the capital cost. Redundant systems would also be needed to ensure reliable continuous operation. Furthermore, the power needed for the current drive plant (the "recirculating power fraction") would affect directly the economics of the reactor. On the other hand, a tokamak reactor operating semi-continuously would be simpler in construction and operation but component fatigue would be increased and the duty cycle would be reduced. However, with a superconducting central solenoid, the ohmic dissipation would be very small, the recirculating power would be kept to a minimum and the power would be used efficiently.

In considering an electricity network, the total power produced must follow demand and the installed capacity must be greater than the highest demand (which occurs only a few times a year). The network would require a few more fusion reactors if they operated semi-continuously (by the inverse of the duty factor of one reactor) or a power station (such as a pump storage system or a gas turbine generator) dedicated to regulating the peak demand.

Under the operating conditions foreseen at present for a reactor (Fig.6) [8], a high recirculating power fraction (Fig.7) would be needed and this would increase significantly the cost of the reactor. To overcome this disadvantage, the current drive efficiency would need to be significantly greater than presently envisaged; or ignition and impurity control would need to be demonstrated at lower density ($<5\times10^{19}m^{-3}$) and higher temperatures; or high power operation in a regime with a dominant bootstrap current would need to be demonstrated. At present, there is no conceptual solution that addresses all these issues consistently.

By comparison, there appears to be clear advantages to semi-continuous reactor operation with inductive current drive: the ohmic dissipation in a superconducting central solenoid is very small, the power is used efficiently and the recirculating power is kept to a minimum. The power in the plasma needed to drive 25MA inductively, could be 2.5MW, or less. For an efficiency from transformer to plasma of between 0.2 and 0.5, the transformer requires only 5-12.5MW of recirculating power. Furthermore, with semi-continuous operation, auxiliary power systems can be optimised for heating to ignition and the reactor really ignites with a fusion amplification factor, Q of about 1000.

Semi-continuous operation is possible with forward current or with alternating current and, providing the central solenoid exceeds a minimum size, both techniques can be utilised on the same device and with the same duty cycle. On JET, both forward current operation, and alternating current operation at 2MA have been demonstrated [9,10]. In a reactor, it would be desirable to smooth the power output, especially for burn interruption in forward current operation, and this can be achieved by external storage of thermal energy or by a 10% over-capacity distributed between several devices. Semi-continuous operation with inductive current drive offers, at present, the only viable solution for a long pulse tokamak reactor.

5.3. The divertor

A major challenge is the achievement of high power exhaust, impurity control, high helium pumping and the rapid re-circulation of several grammes of tritium per second. At present, the divertor appears to offer the greatest potential for meeting this challenge. The divertor configuration (Fig.8), with an X-point inside the vacuum vessel, channels particle and energy flows along open magnetic field lines just outside the separatrix towards a localised remote target and pumping region. With a divertor, the principal source of impurities is well-removed from the main plasma, but sputtered impurities cannot be eliminated completely and these have then to be retained in the divertor region against their natural tendency to contaminate the plasma core.

A fully coherent and robust divertor is one of the most difficult challenges that will be encountered in the ITER EDA. For a reactor producing power in the range 3-6GW, the heat load on the targets of a conventional divertor would be $\sim100MWm^{-2}$. This is too high for reliable operation and methods proposed so far to alleviate this problem (eg. "sweeping" the plasma over the divertor target plates) might even aggravate the problem by increased and faster thermal cycling of the targets. A new divertor concept is required and several possibilities must be developed and tested.

The impact of impurities from plasma facing components are minimised by:

(a) selecting a material of low ionic charge which would be re-deposited mainly in the divertor region; and

(b) eliminating the direct interaction of the plasma with the target.

Beryllium seems to be one of the best choices of material satisfying the requirement (a) in view of its good thermal-mechanical properties and the possibility of low tritium retention. An intensive research and development programme will be needed to qualify beryllium as a target material.

A possible implementation of an advanced divertor concept that satisfies requirement (b) is shown in Fig.8. Two separate areas receive the power load associated with normal operation and abnormal events (such as disruptions). During normal operation the power load is reduced and exhausted by energetic neutrals and radiation over a large area along the divertor channel. The power exhaust perpendicular to the magnetic field would be maintained well below $5MWm^{-2}$. Bumper targets would only receive substantial loads associated with abnormal events. Such a divertor involves plasma flows generated by the ionisation of the incident cold neutral flux and, in principle, a strong flow of deuterium directed towards the targets can prevent back diffusion of impurities with frictional forces overcoming thermal forces [11]. An energetic plasma in the SOL is converted into cold plasma and neutrals at a location in the divertor channel that depends on the power load. At the same time, the neutral density in the private flux zone should increase and the helium pumping requirement becomes reasonable. However, a feedback control system might be needed. This concept leads to a divertor geometry that requires about a quarter of the volume available to the plasma and is quite different from that anticipated, until recently, for ITER. The concept needs to be tested on present tokamaks.

5.4. Power accommodation during abnormal events

It is necessary to accommodate the power produced not only during normal operation but also during abnormal events such as disruptive instabilities, runaway electrons, etc. Steps must be taken to ensure that disruptions rarely occur and that their impact is minimised.

Operation must be restricted to within known limits. Beryllium could be used as a first wall material to render negligible the impurity radiation near the plasma edge and eliminate the risk of plasma disruptions. The vertical stability of the plasma can be assured by limiting the plasma elongation to a value well below two. Error fields, resulting from slight misalignment of coils, could be corrected by external saddle coils, thereby avoiding locked modes. Internal saddle coils could be used for additional active feedback control of potentially unstable modes.

Since disruptions might not be avoided completely, it will be necessary also to introduce safeguards that will minimise their impact. The resistance of the vacuum vessel should be low, so as to limit the dynamical and mechanical effects of disruptions outside the vessel. Sacrificial elements, such as bumpers and limiters, should also be introduced to take the brunt of disruptions, runaway electrons, etc., and prevent major structural damage. The first layer of the inner wall must be refreshed periodically by, for example, the redeposition of evaporated beryllium. Furthermore, the energy dissipated in the superconducting coils must not lead to a current quench.

5.4. Advanced material and component technology

A challenge of a different nature is to develop the technology related to the advanced materials and components needed for the reactor first wall and blanket, the mechanical structure and the superconducting coils. High quality, highly reliable components will need to be manufactured on an industrial production basis.

5.4.1. First Wall and Blanket

To ensure adequate cleanliness and outgassing during operation, the temperature of the first wall surface would need to be maintained above 200°C, and preferably above 300°C. A neutron power of 2-4GW corresponds to a neutron power flux on plasma facing elements of over $2MWm^{-2}$. In addition, plasma disruptions would induce large eddy currents in the first wall structure. To this extent, a metallic first wall with beryllium deposition and a suitable cooling system could be envisaged. Some local bumpers and limiters, which are electrically insulated and mechanically supported, could be introduced.

The first wall and blanket of a reactor will therefore operate in hot conditions and could be considered as an integral entity. In such a hostile environment, simplicity of design will be a key factor to ensure the success of these components. For example, the same coolant could be used to evacuate the heat generated by neutrons and radiation and it may be possible to load the blanket coolant with breeding material in a single-phase or multi-phase system. The coolant for both the first wall and blanket would need to comply with safety regulations and the blanket design would also ensure a low tritium inventory. This raises questions about the use of water as a coolant due to potential contamination by tritium. Therefore, it may require the use of a gaseous or liquid coolant, such as helium, that is not subject to magnetohydrodynamic forces when flowing across magnetic flux and which could be loaded with breeding granules when the blanket was being tested. Another possibility would be the use of a liquid metal flowing in semi-insulated structural pipes. The question of cooling the divertor system has yet to be resolved and depends largely on the divertor concept adopted.

5.4.2. Mechanical Structure and Superconducting Coils

To support stresses due to normal and abnormal operation, the mechanical structure of a tokamak reactor must be designed to distribute and minimise stresses. The toroidal magnetic field coils must resist centripetal and hoop forces. The centripetal forces created by the toroidal field coils can be supported by the central solenoid that is located outside a central bucking cylinder.

Further constraints arise from the superconducting nature of the toroidal field coils, implying the use of cryogenic steels. To avoid crack propagation due to the brittleness of steel, a concept of multi-welded plates may have to be adopted. The coil design must allow the superconductor cable to withstand losses induced in the cable at the start of the plasma pulse and during disruptions. The length of the wound cable does not permit the circulation of liquid helium to absorb the heat released during these transients. To avoid the risk of current quench in the coil, advantage could be taken of the relatively short poloidal length of the structural steel plates to use them as cooling plates. Supercritical helium flowing in the cable conduit could be used as a thermal bath to absorb short timescale heat losses.

For a toroidal magnetic field on axis of 6T, the magnetic field on the cable would approach 13T and the current flowing in a conductor would be 43kA. The reliability of the coil is essential in a reactor. To that extent, no fault could be permitted in the coils for decades. Operational margins must be incorporated in the coil design to accommodate the level of stress in the cable and the nature and thickness of the electrical insulation. The insulator could be based on an inorganic material, such as mica, which maintains its dielectric and mechanical properties in a radiation environment. The successful manufacture of these toroidal field coils requires considerable developments in superconducting coil and Nb_3Sn strand technology to the point where large industrial production could be undertaken.

5.5. An integrated design for a Next Step Tokamak

A further challenge is the integration of all these different reactor issues into a coherent and cost-effective design which eases the problem of stress, limits the cost and ensures the reliability of a Next Step tokamak. Its construction must be on the basis of what is known and it must achieve high levels of simplicity, reliability and safety, and yet provide flexibility and ease of access for inspection and maintenance. A remote maintenance capability must be ensured. This will rely on the simplicity of design and layout for access, and development of robotic and teleoperation devices of exceptional size, functional characteristics and reliability for operation in high radiation environment.

A Next Step Tokamak should: demonstrate sustained high power, semi-continuous operation; study the operating conditions of a reactor; provide a testbed for the study and validation of tritium breeding blanket modules in reactor conditions; test the first wall technology; and define the exhaust and fuelling requirements. Furthermore, the overall capital cost must be tightly controlled and the ratio of this cost to the thermal power output must be in the range relevant to other sources of energy. Basically, the Next Step must be the core of a reactor, if it is to demonstrate fusion as an energy source.

The ITER EDA is examining in detail the ITER CDA device (Fig.9) and moving towards a possible configuration which would achieve the above objectives. Currently, this is a tokamak with a plasma current of up to 24MA, a toroidal magnetic field of 6T, a major radius of about 7.7m, a minor radius of about 3.0m, and an elongation of 1.6. Energy exhaust and impurity control are addressed by high density operation in a pumped divertor configuration. The approach to ignition could utilise low power ion cyclotron resonance heating, while long pulse ignition (~1/2 hour) would be sustained in an X-point configuration at high power (well above 1GW) to allow studies of the ignition domain. With sustained ignition conditions, blanket modules could be tested under neutron power fluxes of over $1 MWm^{-2}$. Advanced divertors and concept development aimed at improved efficiency must also be pursued. In addition, the Next Step should aim to achieve a cost/unit thermal output relevant to the establishment of fusion as a potential economic energy source, should achieve a high level of safety and have minimum effect on the environment.

6. CONCLUSIONS

Magnetic confinement of plasmas, using the tokamak concept, is the most advanced approach to controlled thermonuclear fusion. Recent results from the Joint European Torus (JET) and other smaller tokamak devices, as well as model predictions, have allowed the size and operating conditions of a fusion reactor to be predicted with some confidence.

As a result, the major challenges facing construction of a reactor can be identified. These main critical issues include: a divertor configuration which achieves high power exhaust, impurity control, high helium pumping and the rapid recirculation of several grammes of tritium per second; whether a reactor should be operated continuously or semi-continuously; the development of the technology required for advanced materials and components needed for the first wall and blanket, the mechanical structure and superconducting coils; the development of advanced technology for fast and reliable maintenance in active conditions. A further challenge is the integration of these different reactor issues into a coherent and cost effective design

Worldwide collaboration is foreseen as the way to work more efficiently and to achieve effective solutions to these scientific and technological challenges. The advantage of a World Programme must be to:

- reduce scientific and technological risks;
- allow the study of new concepts;
- provide a wider and more comprehensive database;
- offer flexibility in location and time scheduling.

Several facilities, each with separate, clearly defined objectives, would appear to offer the way forward. The International Thermonuclear Experimental Reactor (ITER) Project, which involves the four parties - the European Communities, Japan, Russia and the USA - would be the first component in such a World Programme. ITER presents the great scientific and technological challenge to demonstrate the reality of fusion as a vast new energy source suited to the industrial world. With concerted effort and determined international collaboration, world resources exist to proceed towards a Demonstration Reactor.

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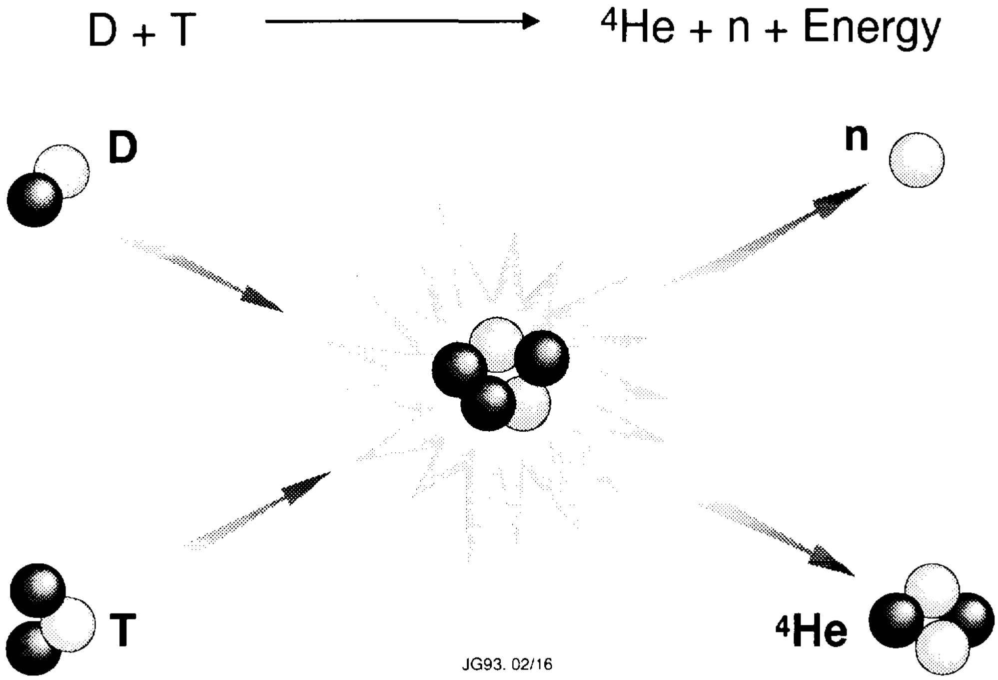


Fig.1: The Deuterium (D) - Tritium (T) fusion reaction

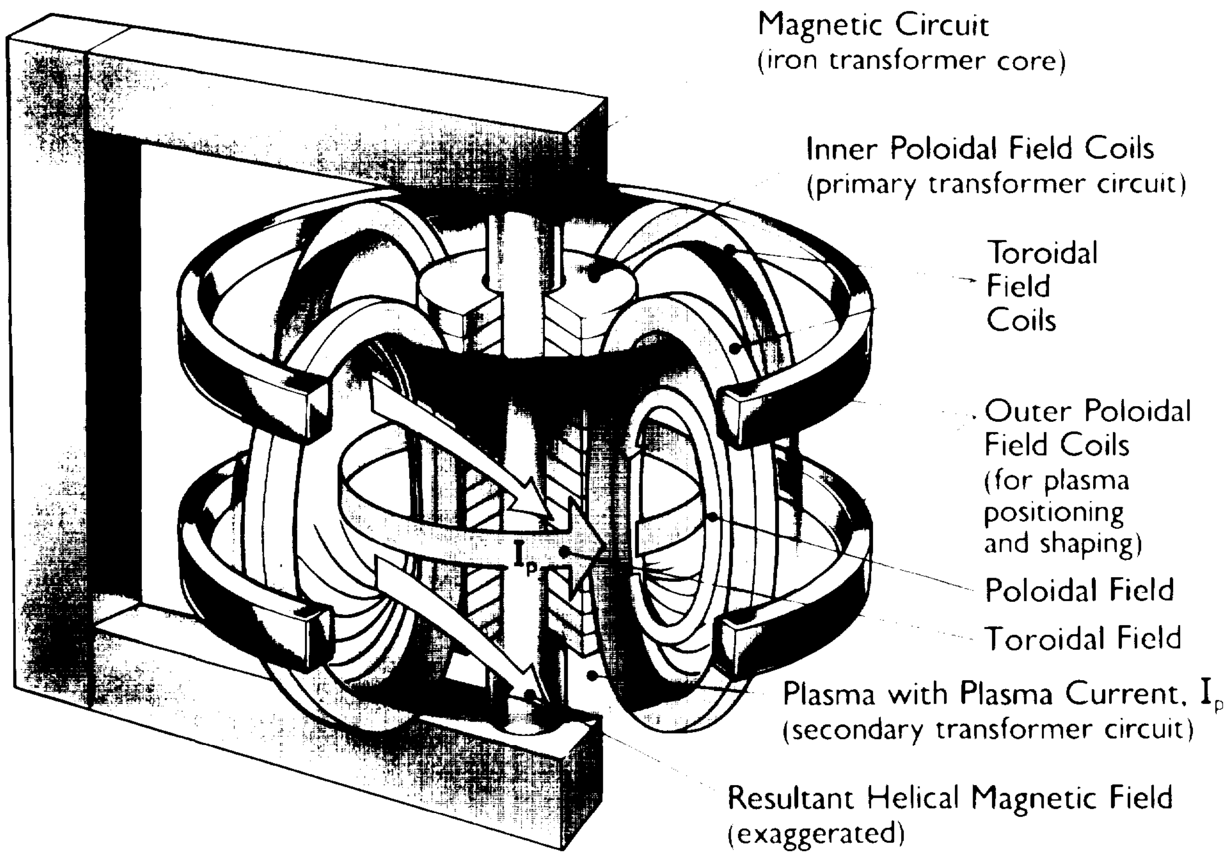


Fig.2: Schematic diagram of a Tokamak fusion device

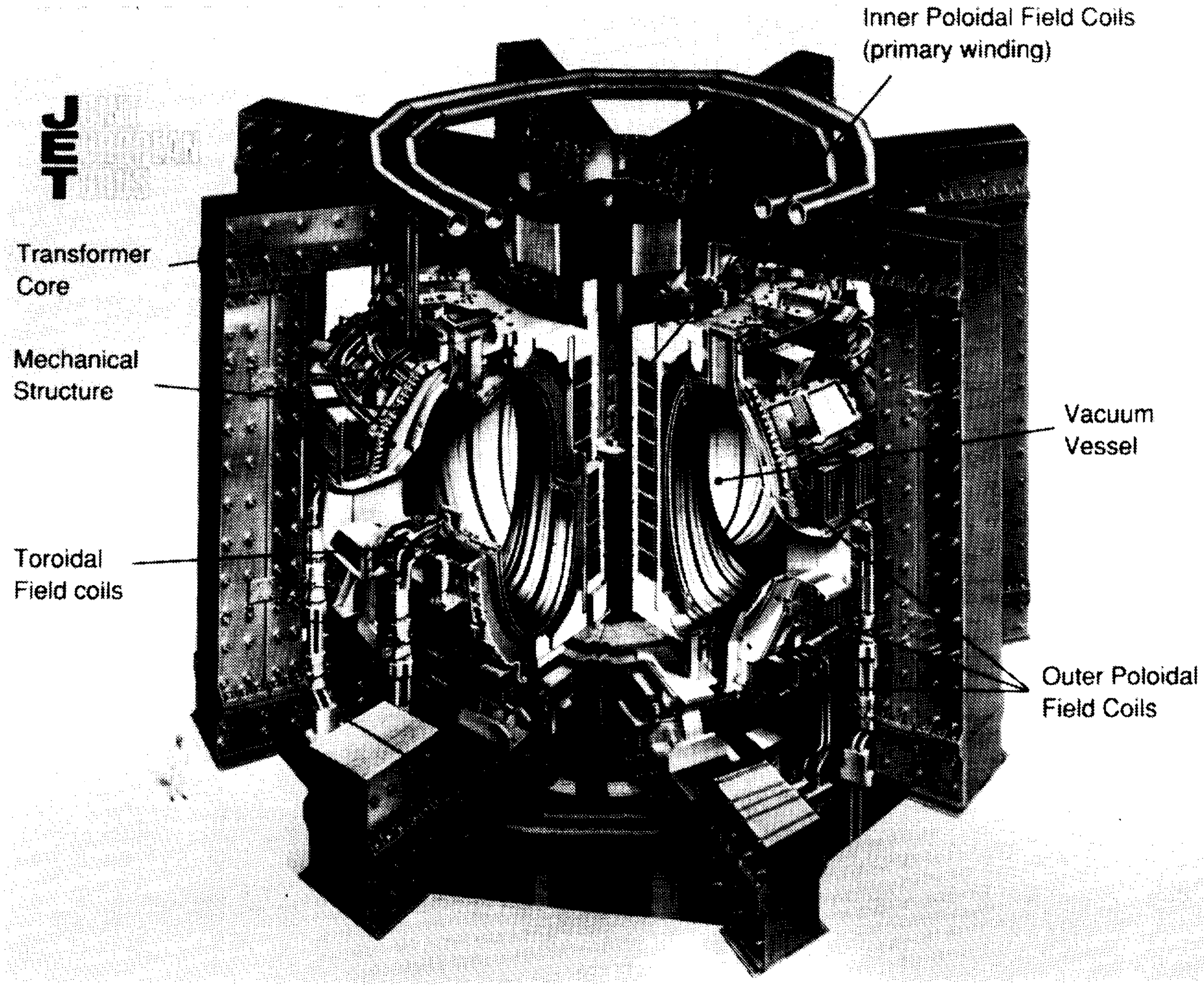


Fig.3: Diagram of the JET apparatus

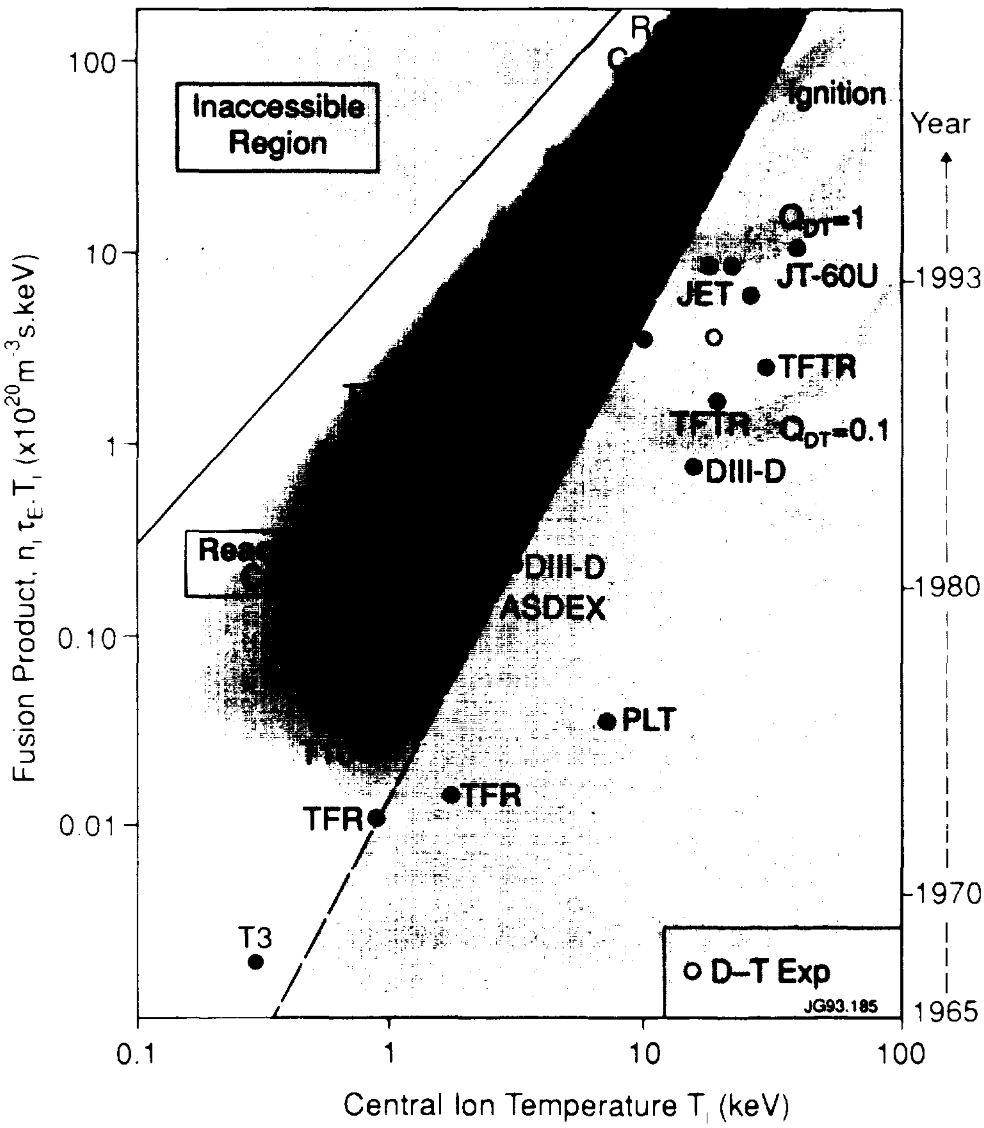


Fig.4: Fusion product ($n_D \tau_E T_i$) versus central ion temperature (T_i) for a number of machines worldwide during the period 1965-1993

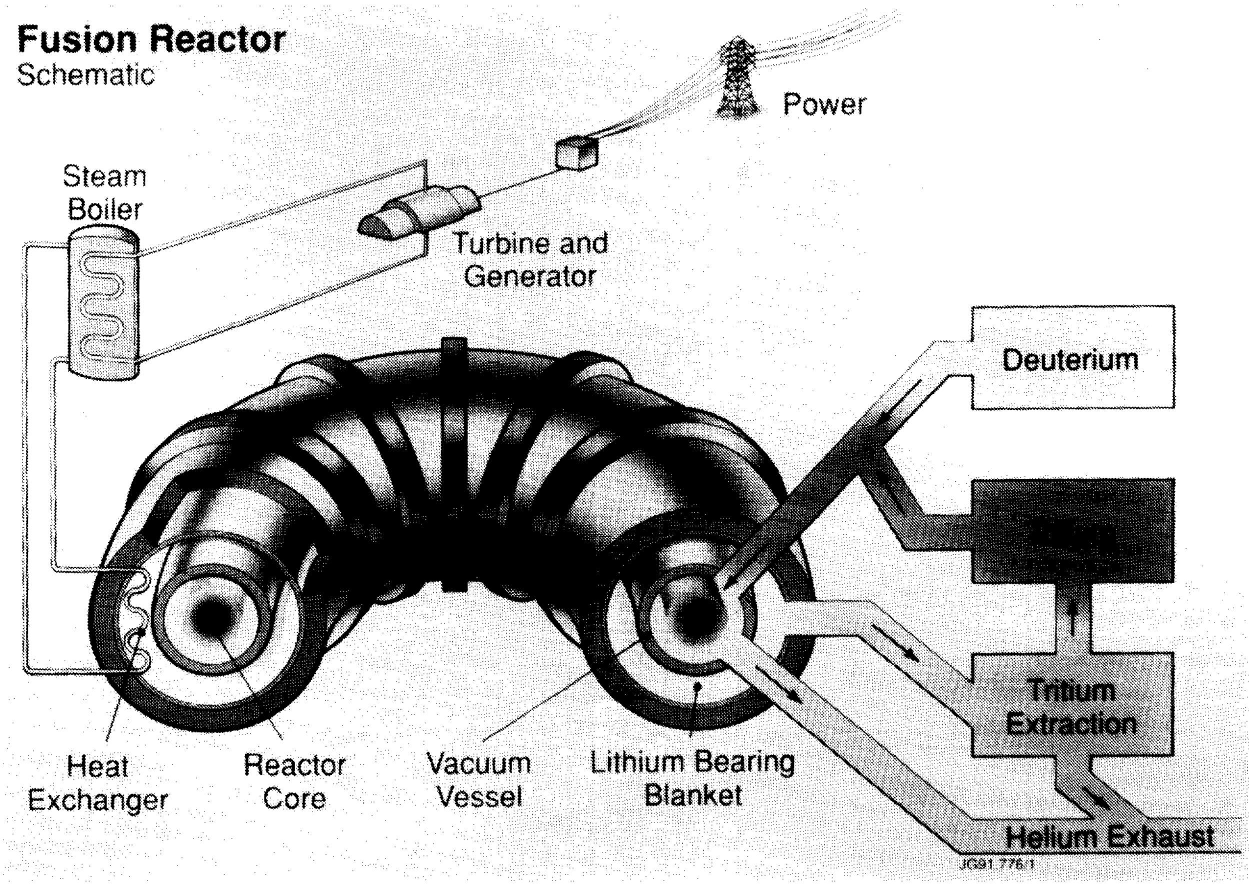


Fig.5: Schematic diagram showing the principles of a fusion reactor

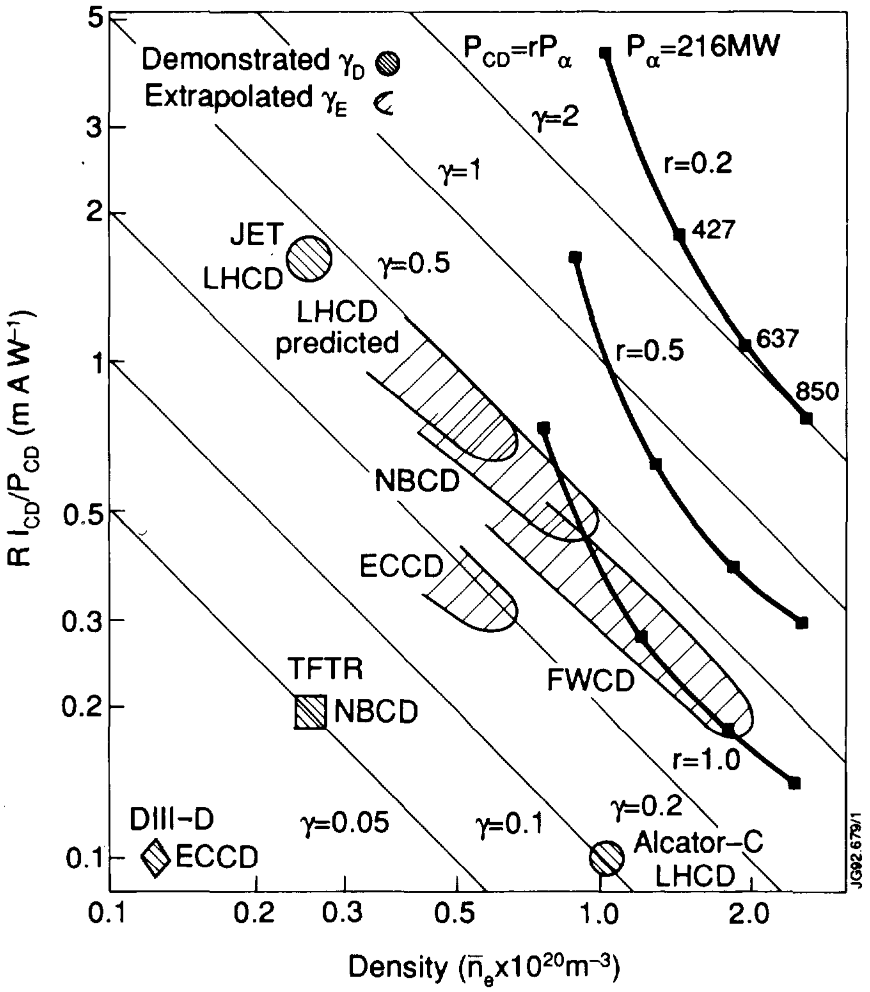


Fig.6: The operating domain for full non-inductive current drive in a tokamak reactor (with R=7.75m, a=2.8m, κ=1.6, I=25MA, B_T=6T) for various assumptions about the recirculating power fraction, r. Also shown are the current drive efficiencies, demonstrated and extrapolated, for various techniques of non-inductive current drive. On the right-hand ordinate is the launched power required for 18MA of non-inductive current drive in a tokamak with R=8m. (LHCD - Lower Hybrid Current Drive; FWCD - Fast Wave Current Drive; NBCD - Neutral Beam Current Drive; ECCD - Electron Cyclotron Current Drive). It is assumed that the bootstrap current can make up the deficit of 7MA.

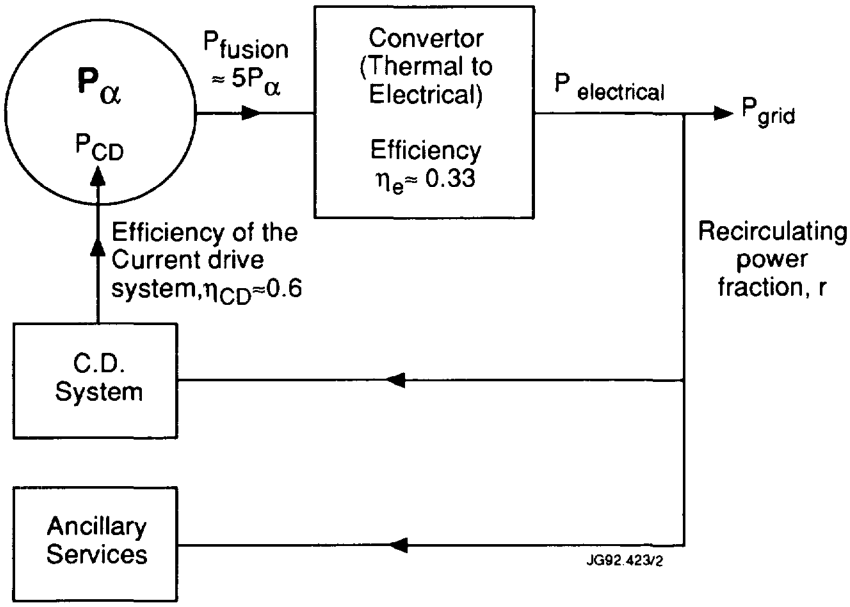


Fig.7: The power requirements for non-inductive current drive in a reactor. Here $P_{CD} = \eta_{CD}\ \eta_e\ r\ P_{fusion} = 0.6 \times 0.33 \times 5\ r\ P_\alpha = r\ P_\alpha$, where P_{CD} is the power launched into the tokamak and available for current drive.

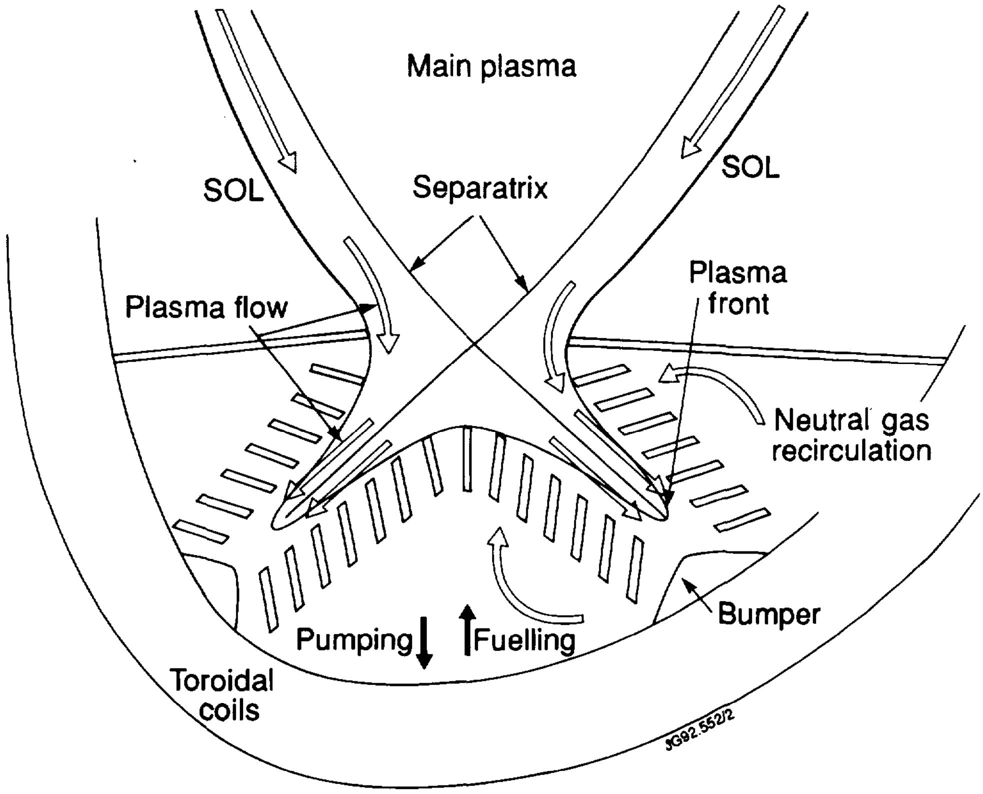
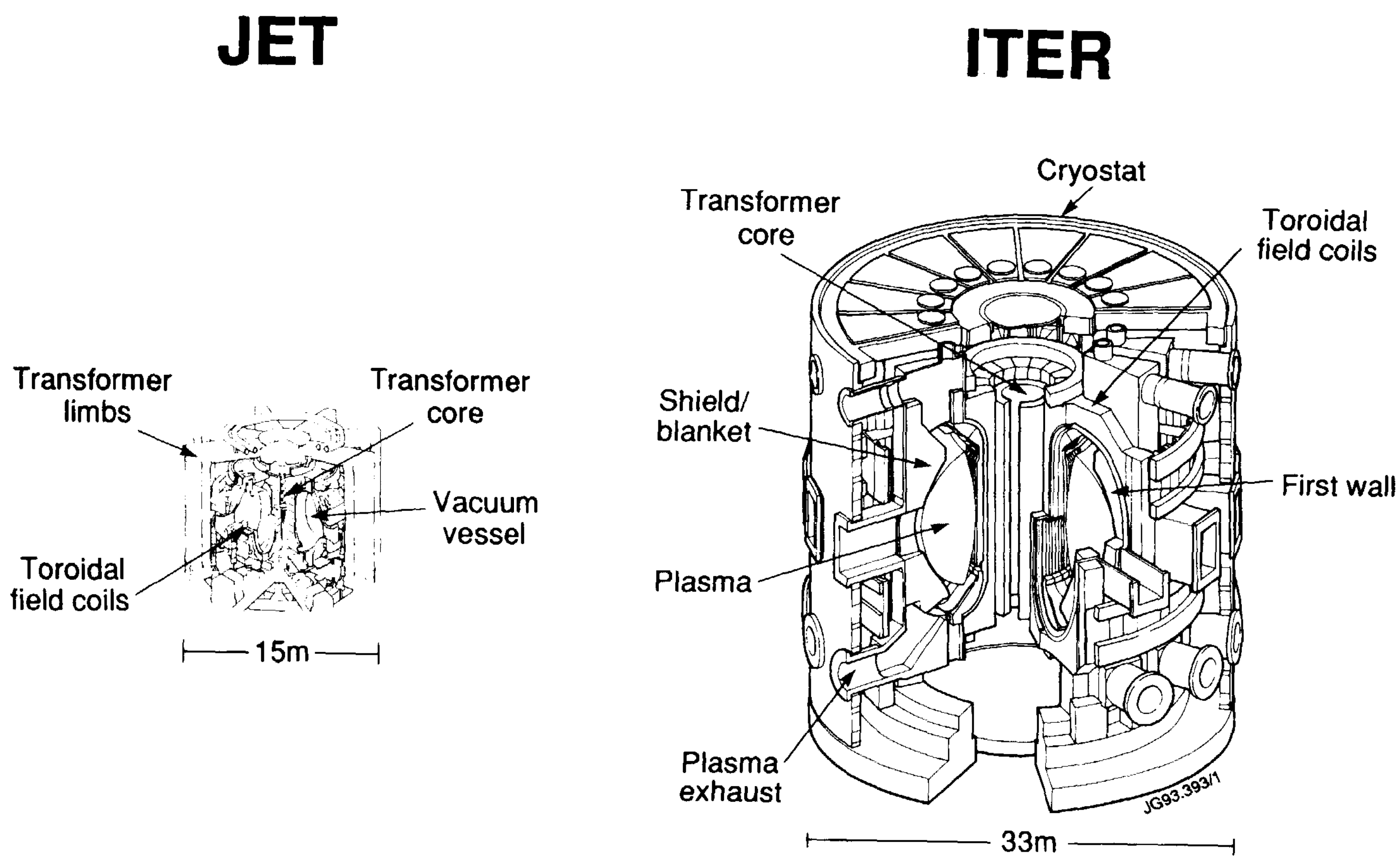


Fig.8: An advanced divertor concept.



Major radius	**3.0m**
Plasma current	**7MA**
Plasma volume	**100m³**

	CDA	(EDA)
Major radius	**6.0m**	**(7.7m)**
Plasma current	**25MA**	**(24MA)**
Plasma volume	**1900m³**	**(2200m³)**

Fig.9: A possible ITER configuration (compared with JET to show relative size).