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P.H. Rebut

Future Plans for JET and Ideas for a Post-JET Machine

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JET Joint Undertaking, Culham Science Centre, OX14 3DB, Abingdon, UK

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P.H. REBUT

JET Joint Undertaking, Abingdon, Oxon, OX14 3EA, U.K.

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INTRODUCTION

The Joint European Torus (JET), situated near Abingdon, UK, is the largest project in the co-ordinated programme of the European Atomic Energy Community (EURATOM). The EURATOM 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, the next European Torus (NET), and DEMO (a demonstration reactor), supported by medium sized specialized Tokamaks.

The objective of JET is to obtain and study a plasma in conditions and dimensions approaching those needed in a thermonuclear reactor. This involves four main areas of work:

(i) to study various methods of heating plasma up to the thermonuclear regime;
(ii) to study the scaling of plasma behaviour as parameters approach the reactor range (that is to determine how the plasma temperature, density and confinement vary with dimensions, shape, magnetic field, plasma current etc., so that we can accurately define the parameters for a reactor);
(iii) to study the interaction of plasma with the vessel walls and how to continuously fuel and exhaust the plasma;
(iv) to study the production of alpha-particles generated in the fusion of deuterium and tritium atoms and the consequent heating of plasma by these alpha-particles.

In a fusion reactor, the values of temperature, (T_i) density (n_i) and energy confinement time (τ_E) must be such that the product of the plasma parameters ($n_i.\tau_E.T_i$) exceeds $5x10^{21}m^{-3}skeV$. Typical values for the parameters that must be attained simultaneously for a reactor are:

- Central ion temperature, T_i — 10-20keV
- Central ion density, n_i — $2.5x10^{20}m^{-3}$
- Global energy confinement time, τ_E — 1-2s

Therefore, the fusion product ($n_i\ \tau_E\ T_i$) represents a figure of merit to judge the closeness of a device to a self sustaining reactor.

JET: PRESENT STATUS

The plasma in JET is heated and contained in a very large toroidal or ring-shaped vessel know as a torus [1]. The plasma is confined away from the walls of the vessel by a complex set of magnetic fields. The detailed shape of the magnetic field is described as a Tokamak, a name used by the Russians who pioneered this particular form of magnetic device for high temperature plasma. There are many tokamaks in the world, principally in Europe, the United States, the Soviet Union and Japan, but JET is by far the largest and most powerful. The main dimensions of the machine are given in Table I, and details of the construction of JET have been given previously [2].

Plasma is confined and heated in JET by a very large electric current - up to 7 Million amperes (MA). In addition, there are two other additional heating methods. These are radio frequency (RF) heating (up to 20MW), and neutral beam (NB) injection heating (up to 21MW) which involves injecting beams of energetic atoms into the plasma. When the plasma is sufficiently hot and well confined abundant fusion reactions will take place converting the deuterium and tritium nuclei into alpha-particles and neutrons. The alpha-particles remain in the magnetic confinement region and their high energy continuous to heat new plasma to keep the reactions going.

Although this will not take place in JET, when sufficient reactions are taking place in a reactor the external heating systems can be turned off as the plasma will continue to heat itself (the ignition condition). The neutrons will escape from the plasma and, in a reactor, will be slowed down in a surrounding blanket or moderator causing the blanket to heat up to a few hundred degrees Celsius. This heat will be removed to raise steam to drive turbines and generate electricity in the conventional way. By making the blanket of a lithium compound, the neutrons will also combine with lithium to produce tritium for fuelling the plasma. A basic objective of JET is to study the self heating of plasmas by the alpha-particles, but the production of tritium and heating by neutrons will be the major objectives of the Next-Step experimental device.

JET is now about midway through it programme and results achieved in JET so far are most impressive [3]. The technical design specifications of JET have been achieved in all parameters and exceeded in several cases (see Table I). The plasma current of 7MA and the current duration of up to 30 seconds are world records and are over twice the values achieved in any other fusion experiment. The neutral beam injection system has been brought up to full power (21MW) exceeding the design value. In addition, the ICRF heating system has taken advantage of improvements in technology to increase its power level to ~18MW in the plasma. In combination, these heating systems have provided 35MW power to the plasma, and this is likely to increase in the near future.

So far, plasma temperatures up to 250M°C (23keV) have been reached and the plasma densities (up to $\sim 1.8 \times 10^{20} m^{-3}$) and energy confinement times (up to 1.5s) are within the range required in a reactor (see Table II). Although these values have been achieved in individual experiments, they have not all been reached simultaneously. There are two regimes of energy confinement observed in a special magnetic configuration (the X-point or magnetic limiter configuration). One of these, a higher confinement regime (called the H-mode) has energy confinement times about twice the lower values (called the L-mode). In both regimes, confinement degradation occurs in that the plasma thermal energy does not increase proportionally to the heating power. Therefore considerably more power is needed to increase the plasma temperature and energy. This problem is presently being investigated.

Fig.1 presents a plot of the fusion product ($n_i.\tau_E.T_i$) versus central ion temperature T_i for a series of experimental devices developed over the last 20 years. It is seen that considerable progress has been achieved in 20 years and JET has now reached a fusion product value of $2.5x10^{20}m^{-3}$skeV, which is only about a factor of 20 below the value of $5x10^{21}m^{-3}$skeV required in a reactor. In particular, JET has shown that plasmas of thermonuclear grade may be contained in a controlled way in a terrestrial device. These latest results put JET on the top of the fusion league table - a position it is likely to maintain for the rest of its operating life.

These experiments are currently being carried out in hydrogen or deuterium plasmas and plans for using tritium are scheduled to come into operation in 1991. The forward programme up to 1992 includes a number of these enhancements and innovations intended to enable JET to create abundant fusion reactions and thereby achieve its objective of producing "plasmas of dimensions and parameters close to those in a fusion reactor".

In summary, the main JET technical and scientific achievements have been the following:

In quasi steady-state:

- a plasma current of 7MA has been obtained for 2s in the plasma;
- JET has operated routinely with plasma current above 5MA, and a current of 6MA has been maintained for 8s;
- ion and electron temperatures, T_i and T_e, in excess of 5keV have been sustained for over 20s at a plasma current of 3MA;
- total heating powers up to 35MW has been delivered to a 5MA plasma producing a total stored energy exceeding 10MJ;
- both electron and ion temperatures simultaneously in excess of 10keV for 2s were observed at a plasma density of $2x10^{19}m^{-3}$;
- confinement times exceeding 1s have been observed;
- neutron yields have reached $1.2x10^{16}n.s^{-1}$ with 20MW of deuterium neutral beams.

In a more transient situation:

- JET has routinely operated with a magnetic limiter (or X-point) configuration at I_p=4.5MA. H-mode plasmas were regularly observed during neutral beam heating;
- A record H-mode plasma at 5MA has been obtained. Transiently, the fusion product ($n_i.T_i.\tau_E$) reached $2.5x10^{20}m^{-3}$.keV.s at temperatures exceeding 5keV;
- the maximum neutron yield exceeded $10^{16}n.s^{-1}$ produced by D-D fusion reactions during an H-mode at 4.5MA; the plasma was heated by 12MW of deuterium neutral beams at an energy of 80keV;
- the total plasma energy content has transiently exceeded 11MJ in X-point operation with 27MW of input power;
- peaked density plasmas were obtained by using pellet injection ($n(0)>10^{20}m^{-3}$).

CONSEQUENCES OF JET RESULTS FOR A REACTOR

The scientific and technical advances achieved on JET have been most impressive. Plasmas of thermonuclear quality have been produced and no adverse effects on confinement have been observed when both electron and ion temperatures reached thermonuclear reactor values. Even so, the record values of neutron yield, pressure and total energy have been obtained while the plasma was in a non-steady-state situation. To predict the performance of a future machine, it is appropriate to start from results of truly steady-state. Transient improvements might prove to be useful to reach ignition but should not be relied upon when working quasi-continuously at full fusion power required routinely in a reactor.

The main consequences of the JET results for a future thermonuclear reactor may be considered as follows:

Energy Confinement and Fusion Product

Degradation of energy confinement with additional power is now a well-known phenomenon. Experiments on JET have extended such observations to high input power, and Fig.2 shows the degradation with power observed in both the H- and L-modes of operation. At high power, the improvement with increased current is obvious but a detailed examination of an individual scan shows a more complex pattern. The gain due to the current saturates in JET when the safety factor at the plasma boundary q_a decreases below 4.

Here the safety factor q_a is defined by the current, I_p, through the relationship:

$$q_a = \frac{AB_t}{\mu_0 I_p R} \qquad (1)$$

where B_t is the toroidal field, A is the cross-sectional area and R the major radius of the plasma.

The degradation of confinement time with the input power is considered a major threat to the success of a future tokamak reactor. The difficulty in improving the fusion parameter ($n_iT_i\tau_E$) (and so the ignition margin of a given machine) by using only additional power is illustrated in Fig.1. The major gains observed in JET result either from transient changes in confinement or from increasing the magnetic field and/or plasma current.

Particle Transport

Particle and impurity transport in JET have been studied under different operating conditions [4,5]. In most cases, particle confinement, like energy confinement, is anomalous. With particle confinement ~5-10 times higher than the energy confinement. The anomaly in particle transport prevents impurity accumulation in the discharge centre. Combined with wall-carbonization, this has kept a low metallic impurity content in JET ($\approx 10^{-4} n_e$) and Z_{eff} has been achieved with large additional power. For most of JET discharges, the mean value of Z_{eff} ranges between 2 and 4 with a radial profile which tends to peak on axis. Radiation losses in the plasma core are marginal, as long as the dominant impurities are of low atomic number. Under conditions of improved energy confinement, such as H-modes or peak density profiles, impurities are also better confined. In the latter case, especially, the medium and low-Z impurities accumulate near the plasma axis. This results in an increased deuterium dilution in the plasma centre and increased central radiation losses.

The particle confinement time being several times the energy confinement time has consequences for the reactivity of the plasma core: it will result in a relatively high concentration of impurities and helium. For high Z impurities, the radiation losses may prevent attainment of the required reactor temperature. For low Z impurities, in addition to helium produced by nuclear reactions, dilution of reacting ions will significantly reduce α-particle power.

Operational Limits

There is a maximum value of density which can be contained with a given value of plasma current. If this density is exceeded, a plasma disruption occurs, in which the plasma confinement is suddenly destroyed and the plasma current falls to zero in a short period of time. Under these conditions, very large mechanical and thermal stresses are produced on the machine structure. Disruptions are thought to be caused by instabilities mostly developing on the plasma surface where the safety factor, q, has an integral value.

Major disruptions occur when the power radiated by the periphery of the plasma, around the $q_\psi = 2$ surface, exceeds the input power in this area or when $q_\psi = 2$ at the plasma boundary. Therefore, they occur preferentially when attempting to increase the plasma density above a limit, which depends on the input power and on the cleanliness of the plasma or when attempting to work at a too low q_a value. The ignition domain of a reactor must be large enough to avoid operational limits experienced in present days tokamaks. A major disruption at full current cannot be completely excluded and the machine must be able to support the resulting stresses, but repetitive disruptions must be avoided. This means that the required performance should be achieved at $q_a \geq 2.5$ and with a low enough heat load on the wall. On the other hand, it can reasonably be expected that the ohmic density limit will be overcome in the presence of strong of α-particle heating.

Impurities

A variety of materials have been used for wall protection and high heat flux components. JET initially operated with metallic walls, but the inner surface of the vessel ($\approx 200m^2$) is now more than 50% covered with graphite tiles. The remaining area is carbonised and the wall temperature is maintained at ~300°C. If previously conditioned by running discharges in helium, the carbon wall has proved to be a very efficient pump for deuterium during plasma discharges.

The dominant impurities in JET plasmas are carbon and oxygen. Their total amount is controlled mainly by the interaction of the plasma at the material limiter. During a discharge, erosion of the limiter material is observed at the point of contact with the plasma and redeposited slightly further outside. Major plasma disruptions are most efficient in transporting materials from the first wall to the limiter.

The use of low Z material for the plasma facing components seems still to be the best option. Graphite, used so far on JET, behaves generally well but problem areas have been identified such as its role as an impurity source, its high chemical reactivity with hydrogen and its high retention of hydrogen leading to problems with density control and with tritium inventory. Combined use of beryllium carbide and of carbon fibre reinforced graphite is a possible option but this is clearly an area where further research is required.

Fuelling and Exhaust

A multi-pellet injector has been used to increase the central density well above normal levels by injecting solid hydrogen pellets into the plasma. To achieve deep density refuelling, pellet speeds in excess of $5kms^{-1}$ will be needed to reach the centre of a high temperature plasma. The increase in central density must be obtained without increasing that at the plasma edge. A particle exhaust system is therefore required to control the edge density.

Achieving control of the impurity influx into the plasma is a prerequisite for building a tokamak reactor. Under present conditions, the lifetime of the plasma-facing components would be severely limited. Impurity control has a direct consequence on the size of a next step device. The means of controlling impurities to be tested in JET are described later in this paper.

IMPACT ON A NEXT STEP DEVICE

The heat and particle transport observed in JET, as well as in other tokamaks, clearly shows anomalous behaviour: the various observed scalings also depend on operating mode. In order to be able to predict the requirements of an ignition experiment or a reactor, it is necessary to obtain a plasma description in terms of physics parameters which fit present experiments and allow calculation of the various temperature and density profiles and their time evolution.

A critical temperature gradient model generally successfully describes most JET results in terms of energy transport. In this model, anomalous transport occurs above a critical temperature gradient [6]. This specific model gives a general scaling, for asymptotic behaviour (at high power) of the form:

$$\tau_E \propto I_p a R^{1/2}$$

$$(nT\tau_E) \propto \varepsilon I^2{}_p B_t R^{1/2}$$

Using a 1-D transport code, good simulations have been achieved not only of JET results in various conditions, but also of other tokamaks.

This model has been used to predict the size of an ignition device and its conditions of operation. The starting point is that reactor studies show that the thickness of the breeding blanket and shielding inside the field coils should exceed 1m in thickness. To use the magnetic field efficiently, a minimum size of the plasma minor radius, a is 3m. With a practical toroidal magnetic field of 4-5T and plasma elongation of 2, the scaling of JET results, using the critical temperature model, shows that a reactor should be able to ignite in the L-mode without problems of confinement with a plasma current capability of about 30MA. The parameters of such a device, in comparison with those of JET are shown in Table III. To define the pumping, fuelling and exhaust specifications, a better knowledge of particle transport and of phenomena controlling plasma wall interactions is required.

One conclusion of the calculations is that it is extremely difficult to vary the electron temperature (and consequently the ion temperature) in a tokamak, without involving heating power larger than the α-particle heating resulting from fusion reactions in the device. This means that ignition should almost be reached without additional heating; only a modest level would be required in a machine of the size predicted for a reactor. The increase in thermonuclear power could be realised by increasing the density at almost constant temperature.

THE JET FORWARD PROGRAMME

Objectives

At present, the approved forward JET programme is shown in Fig.3, which foresees the introduction of tritium into the machine in early 1992 and a conclusion of the Project at the end of 1992.

Present achievements show that the first three objectives of JET are being actively addressed and substantial progress has been made. The programme can now be summarised as a strategy to optimise the fusion product ($n_i.\tau_E.T_i$). For the energy confinement time, τ_E, this involves maintaining, with full additional heating, the values that have already been reached with ohmic heating alone, which means diminishing the effects of energy confinement degradation. For the density and ion temperature, it means increasing their central values to such an extent that operation with deuterium and tritium would produce α-particles in sufficient quantity to be able to analyse their effects on the plasma.

In addition, the JET Project, as a central part of the European Fusion Programme, is directed towards the objectives of that programme, agreed by the Council of Ministers in the following terms:

> *'The main objectives of the programme are ... to establish the physics and technology basis necessary for the detailed design of NET: in the field of physics and plasma engineering this implies the full exploitation of JET and of several medium-sized specialized tokamaks in existence or in construction ....'*

The EURATOM 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, NET and DEMO, supported by medium-sized specialized tokamaks.

Means

The ways of optimizing the fusion product in JET are the following:

- Decoupling the temperature profile from the current density profile;
- Increasing the density of deuterium and tritium ions in the central region to between 1 and $2\times10^{20}m^{-3}$;
- Reducing the edge density by pumping;
- Achieving high central temperatures 12-15keV;
- Increasing the plasma current in two main configurations

  (a) with a magnetic separatrix (X-point operation) up to 6MA;
  (b) with low-Z material limiters at higher currents up to 7MA.

- Introducing tritium into the machine to produce α-particles and study their confinement and heating effects on the main plasma.

The methods to meet these aims are the following:

Control of Current Density Profile

In an ohmically produced (inductively driven) plasma, there is a strong link between the temperature profile and the current density profile in the discharge. This strong correlation can be removed by using non-inductive current drive mechanisms such as neutral beams, ion-cyclotron resonance frequency (ICRF) waves and lower hybrid wave current drive techniques. This would keep the safety factor q above unity everywhere in the plasma, and thus avoid (or considerably reduce) the sawtooth (or internal disruption) phenomena. JET would then benefit from higher core reactivity by sustaining peaked profiles of plasma density and temperature. These methods would help to flatten the current profiles. In addition, this would modify local values of the current gradient and improve energy confinement at the plasma centre.

In JET, all three methods of current drive will be carried out, but the main method for controlling the current profile will be driving 1-5MA current, (depending upon density) in the outer half radius of the plasma using 12MW of lower hybrid current drive (LHCD) power at 3.7GHz. A prototype launcher will be installed in mid-1989 and the final system is planned for introduction in mid-1990.

Control of Density and Impurities

A high speed multi-pellet injector will be used to increase the central density well above present levels by injecting solid hydrogen pellets into the plasma. To achieve deep density refuelling, pellet speeds in excess of $5kms^{-1}$ will be needed to reach the centre of the plasma. A prototype single pellet high speed ($>3kms^{-1}$) gun has been developed and will be installed in mid-1989. The final multiple pellet high speed gun ($>5kms^{-1}$) will be introduced in 1991. In addition, further plasma fuelling will be ensured by inection of neutral beams at 140keV.

Control of Disruptions

In tokamaks, major plasma disruptions impose limits on the plasma density. This not only strongly affects the overall performance of the plasma but also determines the operational limits of the device, as the mechanical and thermal stresses reach their peak values during a disruption. It is proposed to control disruptions in JET by:

- Minimising the radiation cooling at the plasma boundary by using low-Z materials for the first wall facing the plasma. Carbon, which has already been used, is now being replaced with beryllium from early 1989, which has an additional advantage of gettering oxygen;
- Stabilising the magnetic oscillations present at the onset of a disruption using feedback control of magnetic perturbations produced from a set of internal saddle coils.

Increasing the Central Plasma Temperature

High central plasma temperatures (~12-15keV) should be achieved by a combination of additional heating from:

- On-axis ion cyclotron resonance frequency (ICRF) heating using upgraded radio frequency (RF) generators with eight antennae in the torus, which should produce at least 24MW power in the plasma;

- Two neutral beam (NB) injectors providing at least 16MW of 140keV neutral particles in the plasma.

In addition, 10MW of lower hybrid current drive power at 3.7GHz will be dissipated in the outer plasma region, which should assist in raising the whole temperature profile.

Increasing the Plasma Current

To assist improvements in confinement time, the plasma current is being increased in two main modes of operation: with material limiter, the conditions at currents up to 7MA will be optimized; and in the magnetic limiter mode (X-point configuration) the current will be gradually increased up to 6MA.

These experiments, which correspond to a considerable extension to the original design parameters, will make full use of the inbuilt capability of the JET machine and of the new reinforced supports for the vacuum vessel.

Phases of the Programme

The future JET programme has been divided into phases governed by the availability of new equipment and fitting within the accepted lifetime of the Project (up to the end of 1992) (see Fig.3). The Project is proceeding along the paths of:

- Optimisation of plasma parameters and device performance;

preparation for tritium operation.

These are due to converge in the final Tritium Operation Phase of the Project.

In terms of plasma performance the strategy is to raise the value of the fusion product ($n_i.\tau_E.T_i$) towards values of reactor relevance. To achieve this, and especially in view of the current understanding on plasma density and purity control, all elements of the existing JET Programme play specific and crucial roles during the coming operation periods.

In parallel, preparations for tritium operations are proceeding at full speed to ensure that the necessary systems for gas processing, remote handling, radiological protection, active handling and operational waste management are fully commissioned and operating satisfactorily in good time before the introduction of tritium into JET.

On the JET programme, Phase I, the Ohmic Heating Phase, was completed in September 1984, and Phase II (Additional Heating Studies) was completed in October 1988. The present Phase IIIA (Full Power Optimization Studies) has just started and future phases are as follows:

Phase IIIA (End 1988 - Mid 1990)

The main aim of this phase will be to control the plasma density and improve the plasma purity, by the use of beryllium as a first wall material. In addition, work will continue on consolidating the operation of the machine at full additional heating power and to explore further the use of X-point operation as a means of improving confinement. The effect on confinement of the current and density profiles using pellet injection and current drive by ICRH, NBI and LHCD in quasi-stationary states will be established. Emphasis will be given to controlling the plasma density and improving the plasma purity, including, the use of beryllium.

Phase IIIB (Mid 1990 - End 1991)

The scientific aims of Phase IIIB will be to reach maximum performance with high reliability in deuterium plasmas and to control the development of disruptions (through feed-back to stabilize magnetic perturbations) and sawteeth (through current-drive effects).

During this phase, the machine will be upgraded to the status compatible with full radioactive operation (i.e. remote handling systems tested, tritium compatibility of systems completed, shielding requirements implemented, tritium plant commissioned, neutral beam and pellet injectors upgraded for tritium beam and pellet injection).

Phase IV (D-T Phase, Early 1992 - End of 1992)

D-T operation will begin when the overall reliability of all systems operating simultaneously is acceptable. D-T operation is scheduled to last eight months and should provide essential information on the confinement properties and behaviour of hot plasmas close to those needed in a thermonuclear reactor.

The main aim of this phase is to operate JET with D-T plasmas. In the light of present knowledge, all of the currently planned new equipment will be needed to bring the performance to a level justifying the introduction of tritium in the torus. The phase will develop along two main directions:

(i) Establishment of Tritium Operation: The characteristics of D-T plasmas will be studied, including their confinement properties and impurity content. An important element will be control of the composition of the core plasma using tritium neutral beam injection. New scenarios will be explored leading to the optimisation of ICRF and NBI for D-T plasmas;

(ii) High Fusion Yields and the Detection of alpha-Particle Heating: The study and optimisation of intensely heated D-T plasmas will be required both for the maximisation of the fusion yield and for the database for future devices. It is anticipated that current and density profile control should play important roles.

FUTHER POSSIBLE TASKS FOR JET

In a programme to the end of 1992, it would not be possible to achieve the degree of impurity control necessary to prevent the plasma dilution effect from strongly limiting α-particle power. Consequently, taking into consideration the beam power and penetration achievable at that time, the study of heating at high levels of α-particle power could be pursued only after 1992. The experience of impurity control has led progressively to wall protection moving from metal to graphite to beryllium, and the role of the edge effects in establishing good confinement needs to be understood both in order to enable JET to meet its objectives and for the benefit of the design of a Next Step device.

Based largely on JET results, the present studies to define a next step tokamak clearly emphasize the need for obtaining additional information not only on impurity control and plasma-wall interaction but also on modes of operation, such as those avoiding plasma disruptions and on enhanced confinement regimes.

Cessation of JET Programme operations at the end 1992 would mean that the Project would gain only limited, albeit valuable, experience of operation in tritium. However, α-particle heating might be limited, since, within the short time available for investigating plasma control, it is unlikely that the Project would have been able to establish satisfactorily that a solution exists to the problem of impurity control. This is now one of the most crucial aspects of reactor design.

JET is in a unique position to address these problems, related to a next-step device, in the following areas:

Impurity Control

Achieving control of the impurity influx into the plasma is a prerequisite for building a Tokamak reactor. In the case of high Z impurities, radiation losses may prevent attainment of the temperature required for ignition. The presence of low Z impurities, in addition to helium produced by nuclear reactions, dilutes the concentration of reacting ions and therefore reduces the α-particle power. Under present conditions, the lifetime of the plasma facing components would be severely limited. The degree of impurity control achieved has a direct consequence on the size of the next step.

The means of controlling impurities which could be tested are:

- passive, - by reducing the production of impurities at source, i.e. by a proper choice of target plates;
- active, - by creating an outward flow of deuterium from the bulk of the plasma towards target plates with pumps located nearby, therefore preventing impurities from back-scattering into the main plasma.

To be directly applicable to the next-step, the experiments should be performed in conditions so that:

- exhaust is ensured by the combined use of an X-point configuration and a pumped divertor;
- plasma is fuelled by repetitive injection of high speed pellets;
- plasma facing components must reach the equilibrium temperature of the walls of a reactor in the presence of relevant heat and particle loading (i.e. long pulse duration at full additional power);
- the radial distribution of particles in the plasma must reach a steady state.

In JET, the exhaust capability could be improved by installing an axi-symmetric pumped divertor. Figs.4(a) and (b) shows the results of a preliminary design study, in which a single-null system should keep a plasma current capability of up to 6MA and still provide the required information for the next step.

Plasma fuelling would be ensured by the high speed multi-pellet injector presently under development and by a 140kV neutral beam injection system. Tasks would be carried out on plasma facing components, with the objective of defining wall protection and X-point tiles for a next step device. The present line of approach on JET is directed towards low-Z materials such as beryllium or beryllium carbides, but alternative routes might need to be considered and tested.

Another advantage in JET is its long pulse capability, which allows the plasma particle distribution to reach a steady state.

Operational Domain of the Next Step

Regimes of enhanced confinement, such as H-modes, peaked density profiles or stabilized sawtooth would be advantageous in permitting a reduction in reactor size, only if these regimes could be maintained for the duration of the thermonuclear burn. If not, they might still facilitate reaching ignition in the first phase of the burn. As enhanced particle confinement is concomitant with improved energy confinement, helium exhaust in a reactor may justify interrupting the periods of better confinement, i.e. controlling them.

Enhanced regimes of confinement in JET appear to result from either,

(i) better insulation at the plasma edge in the H-mode; key parameters and means of control are:

- the existence of a magnetic separatrix and the distance separating the X-point from the wall;
- the plasma density controlled by the pumped divertor;
- the plasma temperature which depends on additional heating power and on radiative losses near the plasma edge; therefore, the use of beryllium tiles and impurity control should be beneficial;

(ii) reduced heat transport or suppression of instabilities in the plasma core for peaked density profiles. Means of control are:

- deep fuelling of the plasma by either repetitive injection of high speed pellets or high energy beams to increase the central density;
- strong pumping at the edge to maintain the density gradient in steady-state conditions;
- production of fast ions in the plasma core by either ICRF heating or neutral beam injection to produce sawtooth stabilization and control of the current profile by LHCD to prevent the safety factor on-axis decreasing to a too low value.

The design of the next step Tokamak is also strongly dependent on the frequency and severity of major plasma disruptions expected during its life time. Ability to control the plasma in conditions close to disruptive limits would also influence the operating domain of the next-step. Studying these limits and controlling the plasma by use of feedback stabilization is already part of the JET programme.

Fusion Technology Tests

Technology tests must be performed and systems must be assessed for the next step, for which the necessary expertise, necessary equipment available on JET and these tests could be performed independent of machine operations.

Such studies could include:

- exhaust and pumping, especially related to helium pumping in a tokamak;
- fuel purification, by assessing the validity of the JET active gas handling plant relevant to the next step requirements. New components developed for the next step could be incorporated in the JET plant and tested as part of a full system;
- remote handling and viewing, where experience gained on JET in a real tokamak environment is unique. Specific components could be tested and assessed;
- testing of negative ion injectors, by using the existing neutral injection testbed and power supplies;
- LHCD current drive, by assessing and testing components such as grill design, klystrons and windows.

Presently achieved plasma performance in JET has allowed the prediction of the plasma current capability of a Tokamak aimed at achieving ignition. As a step towards an energy producing reactor, the objective of a next-step device is to sustain an ignited state for a period of up to one hour. The control of impurities and the exhaust of helium is an essential condition for maintaining ignition. Techniques of achieving this, together with limits of operation must be experimentally tested prior to construction of a next step. This could be undertaken on JET and the required data could be available by the end of 1994. Information on tritium operation and α-particle heating could be obtained by the end of 1996.

By virtue of its size, its already demonstrated plasma performance and its long pulse capability, JET is in the best position to address these problems in the basic geometry considered for the next step. Furthermore, the expertise of the JET team in these areas is a major asset ensuring the best use of the machine. Such studies are the original *raison d'être* of JET and it is important that JET contributes this necessary information in furtherance of the EURATOM Fusion Programme.

POST JET DEVICES

Next-Step Requirements

A next-step device should be aimed at producing a full reactor plasma and at demonstrating that solutions for the first wall and smooth plasma operations can be found. Such a plasma would require a very long burn time (~1 hour) and a high duty cycle (semi-continuous operation). It should also be possible to test different blanket concepts for a demonstration reactor (DEMO). A specific choice allows a large ignition domain depending on safety factor q and plasma density: the power produced could vary in the range 500-4000MW. This should demonstrate the potential of fusion as an energy source. Such a device is referred to as a "Conceptual Thermonuclear Furnace" and a possible choice of parameters (JIT) is given in Table IV.

One of the questions which has an overall impact on the choice of solution and on the very concept of a reactor is that of the necessity of continuous operation. In a reactor at temperatures required for ignition and in the presence of bootstrap currents, the voltage per turn should be extremely low (~0.05V or less). Flux consumption of 100V.s during the flat-top would ensure a burn time of more than 30 mins which could even exceed 1 hour in the presence of bootstrap currents. A high duty cycle could be obtained by reversing the plasma current when the transformer has reached saturation.

Continuous operation of a tokamak is more complex in terms of physics and equipment and present techniques to achieve this do not seem compatible with the concept of ignition. High energy neutral beams or radio-frequency techniques are required to drive the current. However, the efficiency of this non-inductive current drive is quite low compared to the use of an inductive transformer and the gain in duty cycle might not be significant. In addition to cost and complexity of equipment needed for current drive, the recirculating energy would also increase the overall cost of electricity produced by an amount which could become a dominant factor and condemn such a concept.

On the other hand, semi-continuous operation requires a moderate intermediate thermal storage or a set of tokamak-reactors working together. The main advantage of continuous operation would be to reduce the thermal fatigue of components inside the reactor but as long as the burn time is longer than 1 hour this may not be important.

To maintain simplicity and considering the duty cycle and tritium consumption in the next step machine, a breeding blanket seems an unnecessary complication for an apparatus which is unlikely to test blanket material at high neutron fluence. For the next-step device, although some test modules of breeding blankets could be introduced, only neutron shielding is an essential requirement.

In summary, the next generation of tokamaks must demonstrate that an ignited and burning plasma at high power with semi-continuous operation can be realised.

The aims of such an experiment should be:

- to study the ignition domain;
- to test wall technology;
- to test some breeding blanket modules for a demonstration reactor (DEMO);
- to demonstrate the potential of fusion as an energy source.

A Conceptual Thermonuclear Furnace (JIT)

The main design characteristics of a conceptual thermonuclear furnace (JIT) which would meet these objectives are presented. The main priorities are simplicity and sturdiness of concept [7,8]. Basically, the plasma size is scaled up from JET by a linear factor of 2.5 and the concept uses extensively the experience gained in the design and the construction of JET.

To minimize technical risk, size of the coils, shielding and overall cost, water cooled copper magnets are proposed but superconducting magnets could be envisaged, if certain advantages compensated for the extra complexity. Low current density is used in the water-cooled copper coils to allow continuous operation. It must be noted that a superconducting version would almost meet the requirements of a demonstration reactor if equipped with blankets. A large flux swing ($\gtrsim$400V.s) would also provide the necessary drive to maintain the flat-top currents for periods up to one hour. A overall view of the apparatus is given in Fig.5 and a cross-sectional view is shown in Fig.6.

The device concept is based on a highly integrated and modular construction, where all elements would be manufactured on-site, as transport of such sized components of this size would not be possible. The machine basically consists of 20 sectors supported radially by a cylinder formed by the ohmic transformer. Each of these sectors integrates the toroidal field coil, the vacuum vessel and the mechanical structure (see Fig.7). These different sectors are assembled together by welding flexible lips. The torque induced by the vertical field is taken in shear by the light structure which encloses the toroidal field coil. The coils are made of single pancakes of water-cooled copper corrugated to transmit the shear forces, while the insulation remains purely in compression.

The magnetic configuration contains a single-null X-point where the X-point is located in the lower position to allow the building foundations to take the extra forces resulting from the presence of the separatrix and providing greater access at the top of the machine. This position also decreases the risk of broken tiles falling into the plasma. The presence of the X-point would allow transient H-mode operation during the transition to ignition, but its main purpose is to provide plasma pumping at relatively high pressure.

The shielding is made of several elements which could be disassembled remotely, without dismantling the main apparatus. Each of these elements is made of a single box structure filled with pebbles of metal directly cooled by water. It should be noted that the magnetic circuit is also used as a radiation shield around the machine. One of the main problems in maintaining shielding efficiency is the presence of gaps to allow remote handling and thermal expansion of the different elements. These gaps reduce the shielding locally and could induce a high neutron flux in the coils. The overall layout includes a series of thicker elements which could easily be replaced by reactor relevant breeding-blanket test modules (see Fig.8) without modification of the initial shielding and of the plasma parameters.

The first wall itself is formed by the surface of the shielding box facing the plasma with only local protection by carbon tiles which radiate energy received at a temperature ~1500°K. A continuous deposition of beryllium could ensure that the first wall seen by the plasma is of beryllium or its carbide to limit sputtering and outgassing of tiles.

The divertor presents the main technical challenge in the design of inner wall components. This challenge can be reduced to a manageable level if the region of high power deposition is expanded over a larger area. This can be achieved by moving the X-point vertically or horizontally using a set of internal saddle coils.

Proposed Next-Step Devices

There are several teams working on designs for a Next Step. These include NET (Next European Torus) and ITER (International Thermonuclear Experimental Reactor), involving groups from USA, USSR, Japan and Europe. The results from JET are very important from a reactor point of view. In fact, both design teams are adopting the same design philosophy as JET, that is a large non-circular cross-section and a large plasma current. The main parameters of NET and ITER are included in Table III.

Further details on these devices may be found in previous articles [9,10].

Summary

JET data shows that a tokamak with a plasma current capability around 30MA is required to produce ignition of a D-T plasma in a practical domain. With lower values of current, the risk of not achieving ignition is large, if sawtooth oscillations, impurity radiation, dilution, etc., are taken into account. In a reactor, the size would be defined by the blanket and this leads to a tokamak with a current capability greater than 30MA, when optimized. Confinement would no longer be the dominant problem. However, uncertainties remain in the areas of plasma-wall interaction, fuelling, exhaust and impurity control.

Technical considerations of stress level, wall loading and economy result in a design of a Conceptual Thermonuclear Furnace (JIT) producing a thermal output of several GW in semi-continuous mode. This device is aimed at demonstrating the potential of a Tokamak reactor and to test wall technologies and breeding blankets. For such a device, priority must be given to simplicity and reduction of technical and scientific risks.

If progress in fusion is to continue, it is important to build on JET achievements. To meet these objectives a next-step device must be conceived which clearly establishes the potential of fusion as a major source of power. Such a device must give high priority to simplicity. A Conceptual Thermonuclear Furnace of several GW is such a device, which seems compatible with present technical and financial capabilities. It could prove that fusion is potentially a major source of future world energy.

CONCLUSIONS

The main conclusions from present JET results and possible options for the future may be summarized, as follows:

- JET is a successful example of European collaboration involving fourteen countries;

- This advanced technology machine was constructed on time and broadly to budget;

- On the technical side, JET has met all its design parameters and in many cases, has substantially exceeded these values. In particular, it has reached a record plasma current of 7MA;

- On the scientific side, JET has achieved plasmas with ion temperatures of 23keV and simultaneously ion and electron temperatures have exceeded 10keV. In addition, JET has reached record plasma energy confinement times in excess of 1s;

- Individually, the parameters required for a fusion reactor have been achieved, and simultaneously, the fusion product is within a factor of 20 of the reactor value. JET's overall performance is closer than any other machine to required reactor conditions and has successfully achieved and contained plasmas of thermonuclear grade;

- JET technical and scientific achievements give confidence that a fully ignited experimental reactor could be built, as soon as control of particles is achieved;

- JET results show that a Tokamak with a plasma current of 30MA in a machine of 2-3 times the size of JET is required to produce and maintain ignition;

- Energy confinement would no longer be the dominant problem. However, scientific difficulties remain in the areas of:

  - plasma wall interactions and impurities
  - plasma fuelling and exhaust
  - quasi-continuous operation

- JET is the largest and most powerful fusion experiment in the world. By virtue of its size, its already demonstrated plasma performance and its long pulse capability, JET is in the best position to address these problems in the basic geometry considered for a Next step, and providing vital information required in planning and designing the next-step device.

In conclusion, the message obtained from JET experience is that the plasma of the next-step tokamak should be similar in size and performance to that of an energy producing reactor. The aims of such a machine should be to study burning plasma, test wall technology, provide a test-bed for breeding blankets and most importantly to demonstrate the potential and viability of fusion as an energy source.

The main design characteristics of a Conceptual Thermonuclear Furnace (JIT) dedicated to these objectives have been proposed. Basically, the plasma parameters are scaled up from JET by a factor of 2.5. Watercooled copper magnets are used to benefit from proven technology. A single-null divertor configuration ensures helium exhaust and possibly benefits from an H-mode to reach the ignition domain. By changing operating density, the thermonuclear power could be varied from 0.5 to 4GW(th), according to power loading and tritium consumption.

This device, which seems compatible with present technical and financial capabilities, could prove that fusion is a potentially major source of future world energy.

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TABLE I: JET PARAMETERS

PARAMETER	DESIGN VALUES	ACHIEVED VALUES
Plasma Major Radius (R_o)	2.96m	2.5 - 3.4m
Plasma Minor Radius (hor.) (a)	1.25m	0.8 - 1.2m
Toroidal Field at R_0	3.45T	3.45T
Plasma Current	4.8MA	7.0MA
Neutral Beam Power	20MW	21MW
ICRF Heating Power	15MW	18MW

TABLE II: JET RESULTS

	Best Achieved	Achieved Simultaneously	Reactor Values
Temperature, T			
(M°C)	250	70	120 - 240
(keV)	23	6	10 - 20
Energy Confinement Time, τ_E(s)	1.5	0.9	1 - 2
Density, n, ($x10^{20}m^{-3}$)	1.8	0.5	2 - 3

TABLE III: MAIN PARAMETERS OF VARIOUS DEVICES

	R	a	k	B_t	Add. Power	I_p
	m	m		T	MW	MA
JET	3	1.2	1.7	3.4	40	7
ITER	5.8	2	2.2	5.1	100	20
JIT	7.5	3	2	4.5	50	30
IGNITOR	1.16	0.43	1.8	14	10	12
NET II	6	2.2	2.2	5.4	50	28

TABLE IV: MAIN PARAMETERS OF A CONCEPTUAL THERMONUCLEAR FURNACE (JIT)

Plasma minor radius (horizontal)	(m)	3
Plasma minor radius (vertical)	(m)	6
Plasma major radius	(m)	7.5
Plasma aspect ratio		2 - 2.5
Flat top pulse length	(s)	1000 - 4000
Toroidal field	(T)	4.5
Plasma current	(MA)	30
Volt seconds	(Vs)	425
Additional heating	(MW)	50
Fusion power	(MW)	500 - 4000

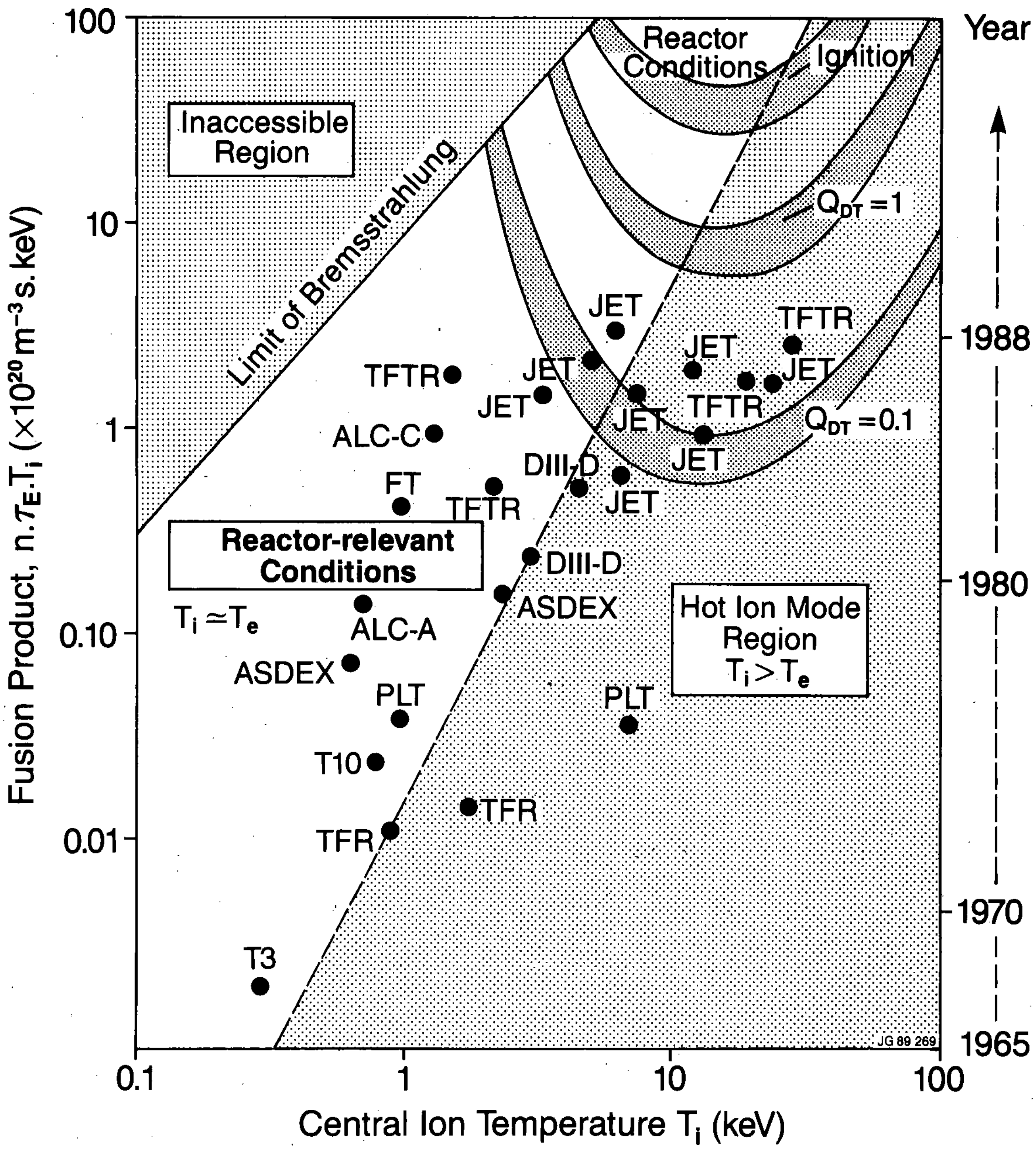


Fig. 1 Fusion product as a function of central ion temperature for various fusion devices. The mode of operation relevant for a reactor is where the electron and ion temperatures are nearly equal at values between 15 and 50 keV. The high density/low temperature region is forbidden due to radiation losses.

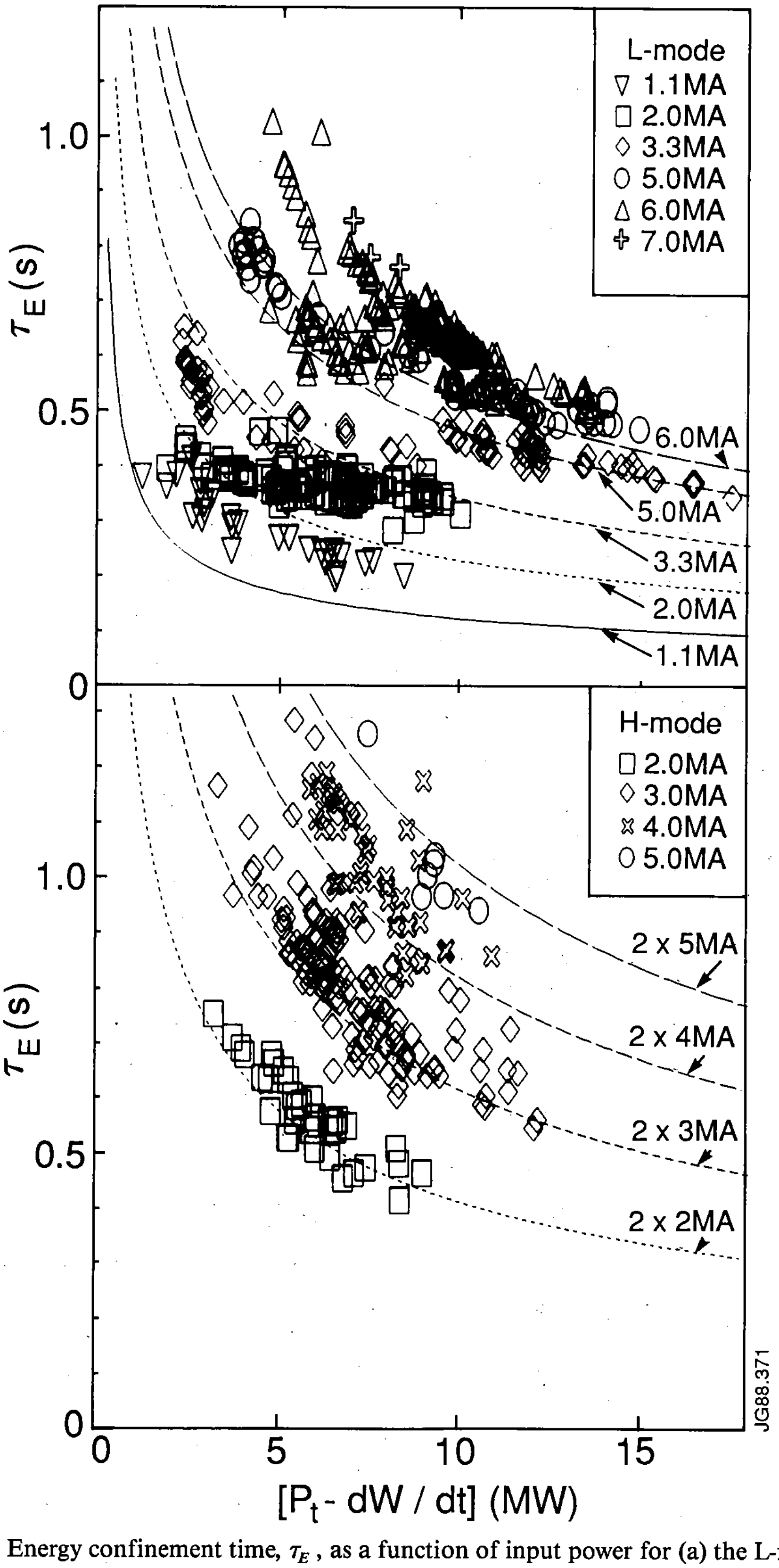
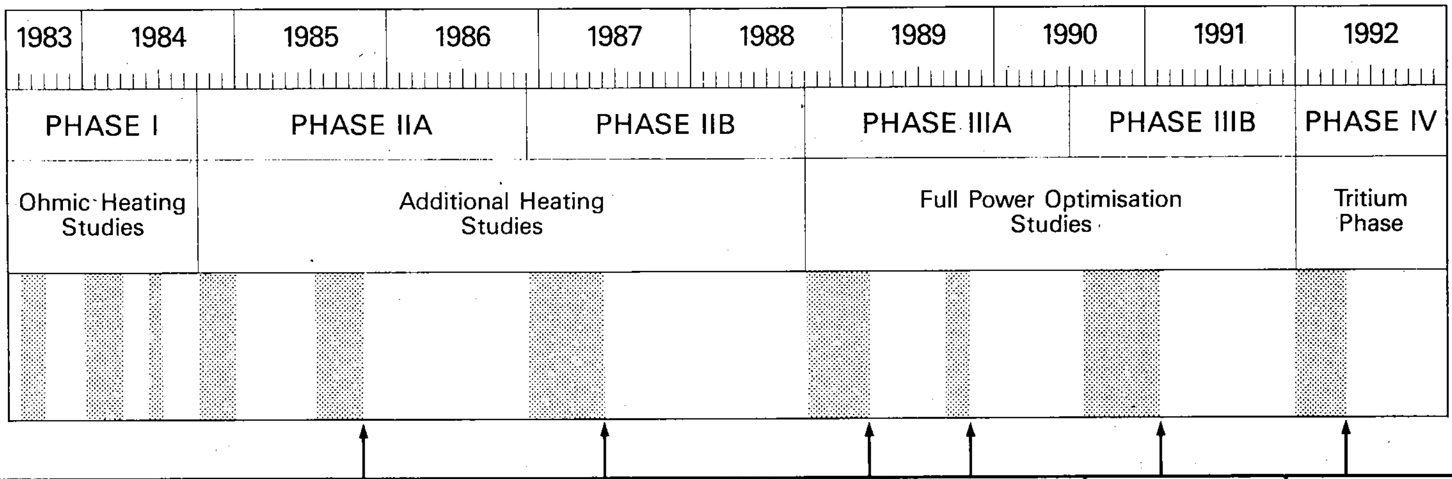


Fig. 2 Energy confinement time, τ_E, as a function of input power for (a) the L-mode and (b) the H-mode of operation.

JET PROGRAMME



Ohmic Systems	5 MA	Vessel restraints and improved volt-seconds for 7 MA operation	Vessel reinforcements			
Separatrix		Additional P1 Coils	Separatrix dump plate supports		Cooled separatrix dump plates	
Limiters	Eight carbon mid-plane limiters	Carbon belt limiters	Beryllium belt limiters			
Pellets	Single pellet injector	ORNL multiple pellet injector (1.5 km s^{-1})		Prototype high speed pellet injector (>3 km s^{-1})	Multiple high speed pellet injector	
NBI	First NBI line (80 kV)	Second NBI line (2×80 kV)	One line modified to 140 kV D		Second line modified to 140 kV D	One line modified to 160 kV T
ICRH	Three A_0 antennae	Eight A_1 antennae		Be antennae Screens		
LHCD			Install Vacuum Chamber	Prototype system	Full system	
Disruption control					Saddle coils	
Tritium and Remote handling					Tritium plant and main RH modifications	Final modifications

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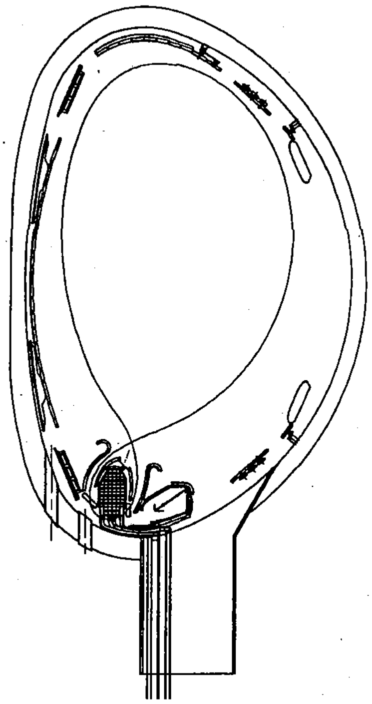
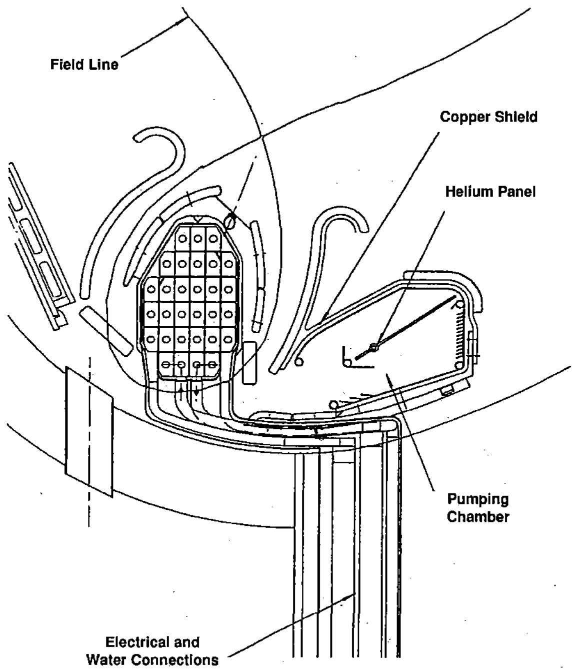
Fig. 3 The JET Forward Programme.



Pumped Divertor



Details of the Pumped Divertor

Fig. 4 (a) Preliminary design study for an axi-symmetric pumped divertor in JET; (b) detail of X-point region showing magnetic coil and pumping chamber.

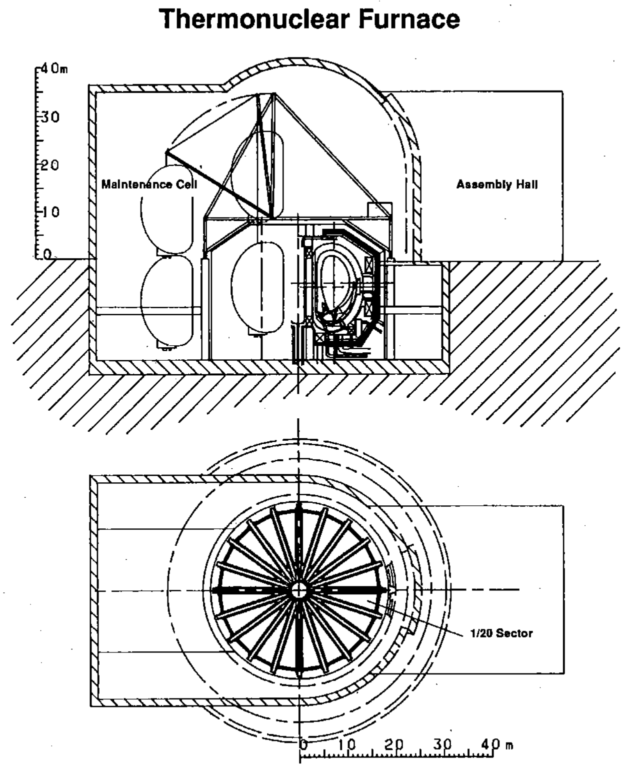


Fig. 5 An overall view of a conceptual Thermonuclear Furnace.

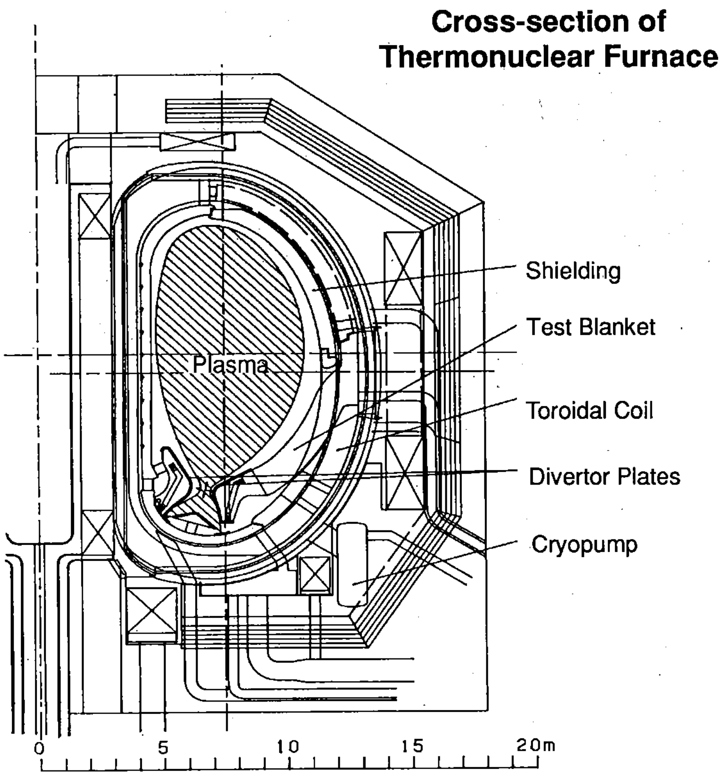


Fig. 6 A cross-section view of a conceptual Thermonuclear Furnace.

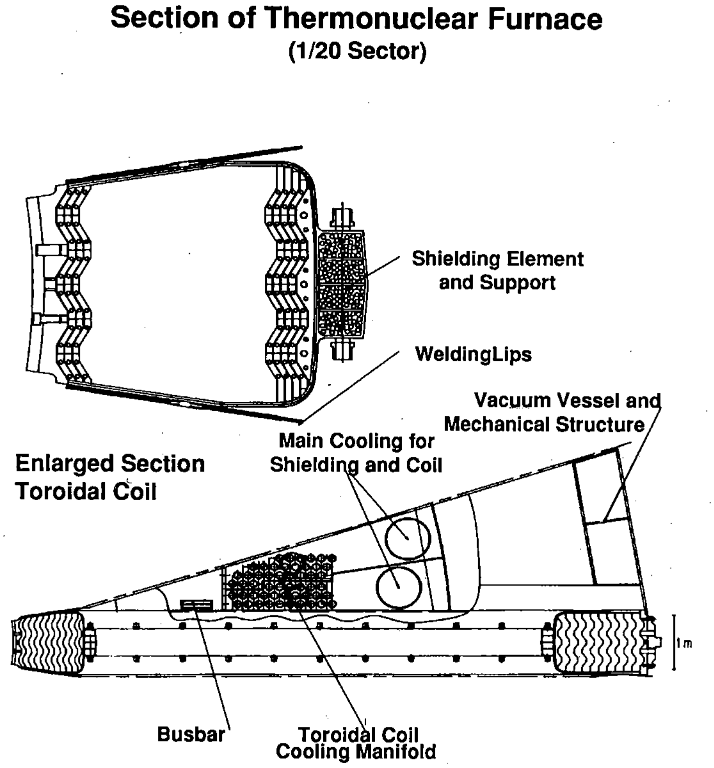


Fig. 7 A sector of a conceptual Thermonuclear Reactor formed as an integrated element including the toroidal coil, the vacuum vessel, and the mechanical structure. An enlarged section of the toroidal field coils are also shown.

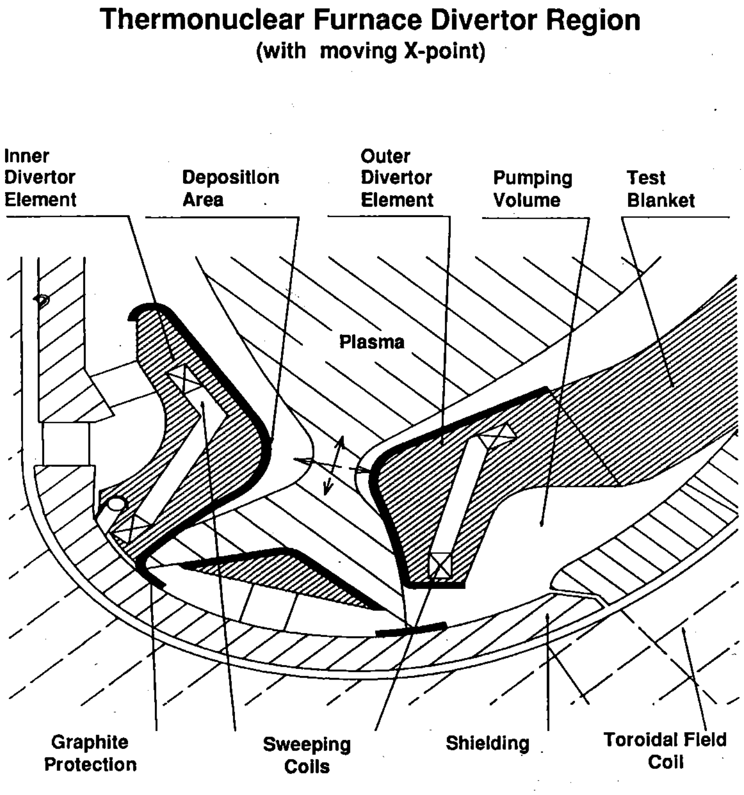


Fig. 8 Divertor region of a conceptual Thermonuclear Furnace showing a set of saddle coils which allow sweeping of the thermal load on the divertor plates covered with graphite. Pumping is carried out through a slot at the base of the divertor.