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I. INTRODUCTION

Considerable advances have been made in nuclear fusion research over many years and the stage was reached in 1988 for international collaboration between the European Communities, Japan, the USSR and the USA, under the auspices of the International Atomic Energy Agency. The aims of the collaboration are to develop the concept of an experimental fusion reactor to demonstrate the scientific and technological feasibility of fusion power. The Conceptual Design Activity (CDA) of the International Thermonuclear Experimental Reactor (ITER) started in early 1988 and was completed by the end of 1990 [1]. During the ITER-CDA the major physics and technological issues related to an ITER design were recognised and the work carried out on these issues provides the basis for the definition of the Next Step.

Since completion of the ITER-CDA, the largest operating tokamak, the JET European Torus (JET) [2], has made further advances in which the plasma density, temperature and confinement time required for a reactor have been achieved in separate discharges. The fusion triple product of these parameters obtained at the same time has reached breakeven conditions, within a factor of about five of that required in a fusion reactor, but only transiently. In November 1991, JET undertook the first tokamak experiments using a deuterium-tritium (D-T) fuel mixture [3]. Approximately 10% tritium was introduced into a deuterium plasma. The peak fusion power generated was 1.7MW in a high power pulse lasting 2s, giving a total energy release of 2MJ. This experiment provided a firm basis for predicting accurately the performance of D-T plasmas from experiments in deuterium. In addition, it demonstrated the technology related to tritium useage. This was clearly a new step forward in the development of fusion as a source of energy and marks the turning point away from plasma physics and towards reactor research.

In spite of these advances, the extrapolation from JET to a fusion reactor still needs an extensive test of the concepts and the technologies required for the design and construction of a reactor. The Next Step in the Fusion Programme must be a major element in fulfilling these needs. It will be a full ignition, high power tokamak producing large amounts of power which must be accommodated. 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 demonstrate highly reliable and safe operation throughout their lifetime. A close and extensive involvement with industry is therefore required.

The next phase of worldwide collaboration, the ITER Engineering Design Activities (EDA), has just begun. The ITER challenge is to address successfully the critical issues related to a reactor and these are considered in the present paper. The critical issues of particular importance relate to:

- a physics understanding of the approach to ignition, the control of ignition (burn control) and burn products (helium "ash"). This requires the development of a plasma model which can be used with confidence to define the operating conditions of a reactor;
- whether a reactor could 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 plasma disruptions;
- 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.

To obtain solutions to these scientific and technological challenges, world collaboration is the way forward. Even so, such collaboration is likely to bring its own special managerial and political challenges.

II. THE OPERATING CONDITIONS OF A TOKAMAK FUSION REACTOR

The issue considered first in this paper relates to the prediction of the size, toroidal field, plasma current and operating conditions of the plasma core of a first reactor. 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 [4]. To limit the uncertainty of the extrapolation, ITER parameters must remain as close as possible to the domain of present experience. In particular, the basic configuration, aspect ratio and elongation of a first reactor should be similar to that of operating tokamaks, such as JET and DIII-D. To define the operating conditions of a reactor, a fully coherent, time-dependent transport model has to be developed. For example, the approach to ignition and the control of ignition (burn control) and burn products (helium "ash") requires a model which addresses all aspects of energy and particle transport, together with a description of "sawteeth", β-limit instabilities and the edge plasma (including the scrape-off layer of open magnetic field lines outside the separatrix in an X-point divertor configuration). One such model, which is consistent with available experimental results and statistical scaling laws (such as ITER89-P [1] for L-mode energy confinement) in their domain of validity, is the critical electron temperature gradient model of energy and particle transport [4,5,6]. This is discussed in the present section and used to define the operating conditions of the plasma core of a first reactor.

A. Definition of a Tokamak Fusion Reactor

A reactor is a full ignition, high power device, producing power in the range of 1-2GW (electrical) or 3-6GW (thermal). It would include: superconducting coils; a divertor with high power handling capability and low erosion, which is likely to require high density operation; an exhaust system for impurities and helium "ash" products; and a D-T fuelling system, which is an important part of burn control. A hot blanket to breed tritium and exhaust heat will surround the plasma with a first wall that is highly resilient to 14MeV neutrons. Activation and tritium inventory must also be minimised. Low power auxiliary heating will be required for the start-up of the reactor which will operate either continuously with non-inductive current drive or semi-continuously with long pulses (in the range of 1-2 hours). Above all, a reactor is a large and complex device which must achieve high reliability, a high level of safety and must be economically viable.

The parameters of a **first reactor** are defined by technology and physics predictions. The minor radius of the reactor plasma needs to be about twice the thickness of the tritium breeding blanket, which makes it approximately 3m. A practical aspect ratio of between 2.5 and 3 sets the plasma major radius 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 will operate with a D-T mixture, and helium "ash". Impurity control will be achieved by plasma flows in an appropriate divertor configuration. Sawteeth will be beneficial in ejecting helium from the central plasma. The reactor plasma will most likely be characterised by a temperature of 25keV and a density greater than $10^{20}m^{-3}$.

B. A Plasma Model

Any model used to predict the performance of a tokamak reactor must be consistent with experimental data from different devices and with physics constraints. Experimental observations support a model for anomalous transport based on a single phenomenon and MHD limits. This **Critical Electron Temperature Gradient** model [4,5,6] of anomalous heat and particle transport features electrons which determine the degree of confinement degradation; ion anomalous transport with heat diffusivity χ_i linked to electron heat diffusivity χ_e; anomalous particle diffusivities, D, for ions and electrons, proportional to χ; and an anomalous particle "pinch", V, related to the profile of the safety factor, q [6].

Specifically, above a critical threshold, $(\nabla T_e)_c$, in the electron temperature gradient, the transport is anomalous and greater than the underlying neoclassical transport. The electrons are primarily responsible for the anomalous transport, but ion heat and particle transport are also anomalous. The general expressions for the anomalous conductive heat fluxes are:

$$Q_e \equiv -n_e \chi_e \nabla T_e = -n_e \chi_{an,e} (\nabla T_e - (\nabla T_e)_c) H(\nabla q)$$
$$Q_i \equiv -n_i \chi_i \nabla T_i$$
$$\chi_i = 2 \chi_e \sqrt{T_e/T_i} \times \{Z_i/\sqrt{(1+Z_{eff})}\}$$

The anomalous coefficients for particle transport are:

$$D = 0.5\,\chi$$
$$V = -D\,(\nabla q)/2q$$

The critical electron temperature gradient model [4,5,6] specifies possible dependencies for $\chi_{an,e}$ and $(\nabla T_e)_c$. To complete the plasma model requires a description of sawteeth, β-limit instabilities and the edge plasma (the separatrix, scrape-off layer and divertor) for which rudimentary models are also included.

The model exhibits the following features which are in accord with experiment: consistency with physics constraints, global scaling laws and statistical analysis; a limitation in the electron temperature; no intrinsic degradation of ion confinement with ion heating power; no dependence of confinement on mass; similar behaviour of particle and heat transport. The model, which has no free parameters, reproduces plasma profiles for a wide variety of discharges in ohmic, L-mode and H-mode (high confinement) regimes in various tokamaks. In particular, the simulation of off-axis heating [7] is almost a direct confirmation of a $(\nabla T_e)_c$, while electron heat pulse propagation studies [8] show a diffusivity, $\chi_{HP} \sim \chi_{an,e} > \chi_e$. The existence of the hot-ion mode is consistent with a critical gradient associated with the electron temperature and current ramp experiments are consistent with the effects of magnetic shear modifying the dependence of confinement on poloidal magnetic field. It should be noted that in the plasma interior, the same model applies to the L-mode and H-mode regimes and particle and energy confinement improve together. However, at the very edge of an H-mode, a transport barrier forms and the transport might be classical over a short distance (~few cm). In fact, the H-mode may be the natural consequence of the transport model, since $\chi_{an,e}$ depends on shear, and reduces towards zero near the separatrix. Furthermore, MHD activity reduces on making the transition from the L-mode to the H-mode. This may imply the stabilisation of some other instability at the edge, where the effect of impurity radiation and neutral influxes on MHD might be important in destroying, at least partially, the edge confinement barrier. This instability is apparently easier to suppress in an X-point configuration with high edge magnetic or rotational shear. However, the spontaneous improvement in edge confinement has yet to be modelled.

C. Modelling Reactor Plasmas

Typical configurations for JET and a **reactor core** are shown in Fig.1(a). The full energy and particle transport model is solved in the simulation geometry (major radius, R=7.75m, horizontal minor radius, a=2.8m, toroidal magnetic field at R, B_T=6T, plasma current, I=25MA, and plasma elongation, κ=1.6) shown in Fig. 1(b). A D-T fuel mixture is assumed and the transport of helium ash, created during the D-T fusion process, is modelled. 1% of the total recirculation (helium and D-T fuel) in the divertor is pumped. Impurity control is assumed to limit the concentration of beryllium impurities to 1%.

Simulations using the model described in Section II.B show that the reactor core operates well in L-mode and at high power. Low power operation is possible in a clean plasma, but high helium concentrations (helium "poisoning") precludes such operation. Contrary to the Goldston scaling law [9], which suggests that the fusion triple product, $n_i\tau_E T_i$ is constant when τ_E degrades as $P^{-1/2}$, the degradation of confinement with the transport model of Section II.B saturates at high power, P. Thereafter, $n_i\tau_E T_i$ increases with power until the β-limit [10] is reached.

In these simulations, ignition is achieved with 10MW of ion cyclotron resonance heating. Burn control at various power levels is achieved with fuel injection controlled by feedback on the power produced. The operating density is then determined by the conservation equations for energy and density. With a feedback system, ignition can be sustained for a wide range of powers (Fig. 2) above a minimum α-power, P_α of approximately 0.2GW. The corresponding minimum density is high (about $10^{20}m^{-3}$) and is compatible with impurity control concepts foreseen at present to rely on energy removal by neutrals and radiation in a divertor (Section IV and reference [11]). Higher power ignition is achieved at even higher density and stored energy but, generally, at lower temperature. The confinement time decreases from 4s when P_α=216MW to 2s when P_α=850MW and the global Troyon factor [10] increases from 1.4 at 216MW to 3 at 850MW (Table I). In all cases the density profile is slightly peaked (Fig. 3) with edge fuelling being sufficient to fuel the centre. Steady ignition conditions are achieved with a relatively high helium concentration (~20-25%): without sufficiently high transport and adequate pumping, helium poisoning can quench the ignition. In fact, while the H-mode might have short term benefits for approaching ignition, long term deficiencies due to helium poisoning can arise (Fig.4). Furthermore, the compatibility of the H-mode and high density operation has yet to be established.

Ignition is achieved in this reactor core with a total current of up to 25MA. When the plasma pressure, p is determined by the Goldston scaling law [9], the bootstrap current, $I_{BS}\propto(a/R)^{0.5}\beta_p I$ [12] (where the poloidal beta, $\beta_p\propto pa^2/I^2$) is limited and dependent only on the input power. In fact, $I_{BS}\propto P^{0.5}$. For the range of conditions considered, the bootstrap current increases from 2.7MA at 216MW to 7.1MA at 850MW and for relatively flat density profiles the bootstrap current tends to occur near the plasma edge. Furthermore, the loop voltage is similar in all cases (~0.1V), the resistive flux consumption is quite low and a magnetic flux of 360Wb is sufficient for one hour current flat-top. Increasing the radius of the central solenoid of the reactor core by 0.8m would make available a further 360Wb and provide an extra hour of steady operation.

III. CONTINUOUS OR SEMI-CONTINUOUS OPERATION OF A REACTOR

Another important issue is whether the reactor should be operated continuously or semi-continuously. 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 external systems for current drive 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, since the ohmic dissipation in a superconducting central solenoid is very small, the recirculating power is kept to a minimum and the power is 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 will require a few more fusion reactors if they operate 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.

A. The Efficiency of Non-inductive Current Drive

Ignition is achieved in the reactor core studied in Section II.C for a total plasma current up to 25MA and the bootstrap current is in the range 3-7MA. Thus, it would be necessary to provide about 15-20MA of non-inductive current drive for continuous operation. In principle, this could be provided by the injection of beams of high energy neutral particles or radio frequency waves at various frequencies, including fast waves, lower hybrid waves and electron cyclotron waves [13]. The current, I_{CD}, that can be driven non-inductively is usually determined from the efficiency of the current drive technique, defined as $\gamma = I_{CD} R \langle n_e \rangle / P_{CD}$ [in units of A/m^2W], where P_{CD} is the power launched into the tokamak and available for current drive. For each technique, the presently demonstrated efficiency, γ_D and the extrapolated efficiency, γ_E (based on extrapolations to an average electron temperature of 20keV in an ITER CDA tokamak [14]) are plotted in Fig. 5, together with the lines of constant efficiency, γ. It will be noted that lower hybrid current drive alone has demonstrated an efficiency as high as the extrapolated efficiency, but only at low density. At the higher density required in a reactor core, only fast wave current drive offers potential, but so far this technique has not been demonstrated.

B. Power Requirements for External Current Drive Systems

Also shown on the right-hand ordinate of Fig. 5 is the power required for 18MA of non-inductive current in the reactor core of Section II.C. It should be noted that for densities above $10^{20}m^{-3}$ and an efficiency $\gamma = 0.5 \times 10^{20} A/m^2W$, more than 300MW is needed. The power available from the reactor for current drive is, however, limited. Consider a reactor which produces a fusion power, $P_{fusion} = 5P_\alpha$ (Fig. 6). The efficiency, η_e, of converting thermal power to electrical power is about 0.33. Most of the electrical power will be supplied to the grid, but a fraction, r - the recirculating power fraction - may be used to power ancillary services, and any systems needed for non-inductive current drive. Given an optimistic efficiency for the current drive system, $\eta_{CD} \cong 0.6$, the launched power, P_{CD} is approximately given by:

$$P_{CD} = \eta_{CD}\ \eta_e\ r\ P_{fusion} = 0.6 \times 0.33 \times 5\ r\ P_\alpha = r\ P_\alpha$$

It is possible to superimpose the operating domain of the reactor core modelled in Section II.C on to Fig. 5. The simulations now include the auxiliary power needed to drive a non-inductive current equal to the difference between the total plasma current of 25MA and the bootstrap current. The radial profiles of the auxiliary power and the total current are assumed to be similar. With various assumptions on the recirculating power fraction, r, it is seen (Fig. 7) that the most optimistic extrapolations with γ=0.5x10^{20}A/m^{2}W require all the electrical power produced by the reactor (r=1.0). Even then, over most of the operating domain, only a fast wave system (which has yet to be demonstrated experimentally) could provide central current drive. With a reasonable assumption of r=0.2, a current drive efficiency of 2x10^{20}A/m^{2}W would be required. Even reactor concepts relying on a high bootstrap current [15] have difficulty in meeting the requirement of a reasonable level of recirculating power [16].

C. The Potential of Non-inductive Current Drive for Continuous Operation of a Reactor

Non-inductive current drive requires the full-time operation of a current drive plant which must include redundant systems to ensure reliable, continuous operation. Under the conditions foreseen at present for a reactor, a high recirculating power fraction 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 (<5x10^{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.

D. The Potential of Inductive Current Drive for Semi-continuous Operation of a Reactor

By comparison, there appear 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, with a plasma loop voltage of 0.1V (Table I), 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, provided 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, with reduced ohmic dissipation and extended flat-top by heating, and alternating current operation at 2MA have been demonstrated [17,18]. 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.

IV. 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, 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 a fusion power in the range 3-6GW, the heat load on the targets of a conventional divertor would be in the region of $100MWm^{-2}$, or more. 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. One solution for an advanced high density divertor is presented in this section: energy is lost by energetic neutrals (created by ion-neutral collisions such as charge-exchange) and radiation and the scrape-off layer plasma is extinguished before a material target is reached.

A. Principles of Energy Exhaust in a Divertor

The principles of energy exhaust in the divertor are contained in the conservation laws for particles, momentum and energy for one-dimensional flow along the open magnetic field lines in the scrape-off layer (SOL). If a cold neutral flow is injected perpendicular to the magnetic field in the SOL, the SOL plasma can be extinguished before the target plates are reached provided the divertor region is sufficiently large and the density is relatively high [11]. Energy is transported to the targets by energetic neutrals and radiation and can be distributed more evenly over the target than with an ion flux. Even though the neutrals will not be accelerated by the sheath potential, the energy with which these neutrals leave the SOL is still sufficiently high to sputter target-plate impurities and can be reduced only by radiation and rapid equipartition between electrons and ions, in a cold plasma target, or by neutral-neutral collisions, in a gas target.

B. Principles of Impurity and Particle Control in a Divertor

It is fundamental to control impurity ingression into the plasma core and avoid dilution from both plasma facing components and helium ash. 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 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.

Under the divertor conditions proposed in this section and reference [11], plasma flows are generated by the ionisation of the incident cold neutral flux and, in principle, a strong flow of deuterium directed towards the targets can prevent the back diffusion of impurities with frictional forces overcoming thermal forces [19]. At the same time, the neutral density in the private flux zone should become high. The helium pumping requirement is then reasonable, but a feedback control system might be needed.

C. A Possible Implementation of this Advanced Divertor Concept

A possible implementation of this advanced divertor concept is shown in Fig. 8. Two seperate areas recieve the power loadassociated with normaloperation and abnormal events (such as disruptions). During normal operation the power load is reduced and exhausted by energetic neutrals 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 recieve substantial loads associated with abnormal events. Such a divertor converts an energetic plasma in the SOL into cold plasma and neutrals in the divertor channel, its position depending on the power load. The 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 so far for ITER. The concept needs to be tested on present tokamaks.

V. DISRUPTIONS

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

First, operation must be restricted to within known limits. Beryllium can be used as a first wall material to render negligible the impurity radiation near the q=2 surface and eliminate the risk of density limit disruptions. The vertical stability of the plasma can also be assured by limiting the 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.

Second, 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.

VI. 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.

A. 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 (which could be possible) 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 operate in hot conditions and may 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.

B. Mechanical Structure and Superconducting Coils

To support stresses due to normal and abnormal reactor operation, the mechanical structure must be designed to distribute and minimise stresses. A possible solution, which is integrated into the design considered in Section VII, is discussed. The toroidal magnetic field coils must resist centripetal and hoop forces. The centripetal forces created by the toroidal field coils are supported by the central solenoid that is located outside a central bucking cylinder. This configuration, used in JET, minimises the material required to resist the forces. In the proposed concept (Fig.9(a)) the coils are in equilibrium, the hoop forces being resisted by internal steel plates, contained in the coil winding, and by a steel belt running along the outboard portion of the coil. This belt is connected by a combination of bearing edges and hinged connections to the central bucking cylinder. A low friction surface could allow some vertical slip at the interface between the toroidal field coils and the central solenoid. This design ensures a tight contact between the toroidal field coils and the central solenoid at all times. Finally, a combination of shear keys could resist the overturning moment due to the interaction of the poloidal and toroidal magnetic fields (Fig. 9(b)). Therefore, the bucking cylinder supports the central solenoid against the centripetal force and a part of the hoop tensile forces of the toroidal field coil.

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.

VII. AN INTEGRATED DESIGN FOR A NEXT STEP TOKAMAK

The final challenge addressed in this paper 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 the development of robotic and teleoperation devices of exceptional size, functional characteristics and reliability for operation in high radiation environment. ITER should: demonstrate sustained high power, semi-continuous operation (eg. more than 1GW for 1000s); 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.

A possible configuration which would achieve these objectives is a tokamak with a plasma current of up to 25MA, a toroidal magnetic field of 6T, a major radius of about 7.5m, a minor radius of about 3m, and an elongation of 1.6 (Fig.10). 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 with X-point, L-mode confinement at a power of several GW. With sustained ignition conditions, blanket modules could be tested under neutron power fluxes of over $1MWm^{-2}$.

VIII. CONCLUSIONS

Recent results and model predictions allow the size and operating conditions for a fusion reactor to be predicted with some confidence. Simulations show that the reactor core of Section II would operate at high power with X-point, L-mode confinement. The advantage of better H-mode energy confinement can be offset by increased levels of helium "ash". Ignition can be maintained for a wide range of α-powers, but above a minimum, approximately equal to 0.2GW. The corresponding minimum plasma density is about $10^{20}m^{-3}$ and central plasma temperatures are about 25keV. However, before constructing such a reactor core there are certain critical issues which need to be resolved.

At present, the basis for a reactor that operates continuously is not apparent. Such a continuously operating reactor would require a convincing demonstration of:

(a) a current drive efficiency exceeding $1x10^{20}A/m^2W$ for a density above $10^{20}m^{-3}$; or

(b) ignition and adequate impurity control at a density of $5x10^{19}m^{-3}$ and a current drive efficiency of $0.5x10^{20}A/m^2W$; or

(c) high power operation in a regime with a dominant bootstrap current.

Semi-continuous operation with inductive current drive offers, at present, the only viable solution for a long pulse tokamak reactor. Such a reactor would be simpler in construction, would use recirculating power more efficiently and is likely to be more reliable in operation. The Next Step tokamak must be based on inductive semi-continuous operation.

A fully coherent and robust divertor for power exhaust and impurity control is one of the most difficult challenges that will be encountered in the ITER EDA. For a reactor core producing fusion power in the range of several GW, the heat load on the targets of a conventional divertor would be too high for reliable operation. Further development of the divertor concept is required and several possibilities must be tested. An acceptable solution is most likely to be achieved with a high density divertor that utilises: energy loss by energetic neutrals (created by ion-neutral collisions such as charge-exchange); a radiative or cold plasma or gas target; and high plasma flows for impurity retention.

It is also necessary to accommodate the power exhausted during abnormal events, such as disruptions. Limitations on operation, control coils and sacrificial elements will be needed to help avoid disruptions and limit their impact.

The technology related to the advanced materials and components needed for the reactor first wall and blanket, the mechanical structure and the superconducting coils must develop. High quality components with sufficient reliability will need to be manufactured and Industry must be involved at an early stage to enable fabrication on a production basis.

The final 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 which is, in fact, a reactor core.

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, are required. ITER, which involves the four parties - the European Communities, Japan, the Russian Federation and the USA - is the first component in such a World Programme. ITER is a great scientific and technological challenge: it should demonstrate the reality of fusion as a source of energy. With determined worldwide collaboration, the support of the ITER Parties and the Fusion Community, ITER will be built and these challenges will be met. Even so, such a collaboration is likely to bring its own special managerial and political challenges.

IX. REFERENCES

[1] Tomabechi, K. and the ITER Team, Nucl. Fusion, **31(6)** (1991) 1135

[2] The JET Team, The JET Project - Design Proposal, Report EUR-JET-R5, CEC, Brussels (1975).

[3] JET Team, Nucl. Fusion, **32(2)** (1992) 187.

[4] Rebut, P-H., Watkins, M.L., Gambier, D.J. and Boucher, D., Phys. Fluids (B), **3(8)** (1991) 2209.

[5] Rebut, P-H., Watkins, M.L. and Lallia, P.P., in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 12th Int. Conf., Nice, France, 1988) **Vol 2**, IAEA, Vienna (1989) 191.

[6] Boucher, D., Rebut, P-H. and Watkins, M.L., Comptes Rendus de l'Academie des Sciences, **315(II)** (1992) 273.

[7] Balet, B., et al., Europhysics Conference Abstracts (Proc. 19th European Conference on Controlled Fusion and Plasma Physics), EPS, Innsbruck, Austria, **Vol. 16C (Part I)** (1992) 59.

[8] Tubbing, B.J.D., Lopes Cardoza, N.J. and van der Wiel, M.J., Nucl. Fusion, **27** (1987) 1843.

[9] Goldston, R.J., Pl. Physics and Cont. Fusion, **26(1A)** (1984) 87.

[10] Troyon, F., Gruber, R., et al., Pl. Physics and Cont. Fusion, **26(1A)** (1984) 209.

[11] Watkins, M.L. and Rebut, P-H., Europhysics Conference Abstracts (Proc. 19th European Conference on Controlled Fusion and Plasma Physics), EPS, Innsbruck, Austria, **Vol. 16C (Part II)** (1992) 731.

[12] Cordey, J.G., Challis, C.D. and Stubberfield, P.M., Pl. Physics and Cont. Fusion, **30(11)** (1988) 1625.

[13] Fisch, N.J., Rev. Mod. Physics, **59(1)** (1987) 175.

[14] Nevins, W.M., et al., in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 13th Int. Conf., Washington, USA, 1990) **Vol 3**, IAEA, Vienna (1991) 331.

[15] Seki, Y., et al., in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 13th Int. Conf., Washington, USA, 1990) **Vol 3**, IAEA, Vienna (1991) 473.

[16] Rebut, P-H., et al., Proc. Europhysics Topical Conf. on RF Heating and Current Drive of Fusion Devices, Brussels, Belgium, 1992, To be published in Plasma Physics and Cont. Fusion.

[17] Brusati, M., et al., Europhysics Conference Abstracts (Proc. 19th European Conference on Controlled Fusion and Plasma Physics), EPS, Innsbruck, Austria, **Vol. 16C (Part I)** (1992) 307.

[18] Tubbing, B.J.D., et al., Nucl. Fusion, **32(6)** (1992) 967.

[19] Keilhacker, M., et al., Nucl. Fusion, **31(3)** (1991) 535.

Table I

Parameter	Case 1	Case 2	Case 3	Case 4
P_α(MW)	216	427	637	850
$T_i(0)$(keV)	19	23	21	16
$<n_e>(10^{19}m^{-3})$	11	16	21	26
τ_E(s)	4.0	3.4	2.8	2.1
I_{boot}(MA)	2.7	4.8	6.6	7.1
n_{He}/n_e (%)	19	24	24	20
V_{loop}(V)	0.15	0.10	0.10	0.11
g_{Troyon}	1.40	2.34	2.86	2.97

These values were obtained using the full energy and particle transport model for electrons, D-T ions, helium and a specified concentration of beryllium in an X-point, L-mode tokamak configuration.

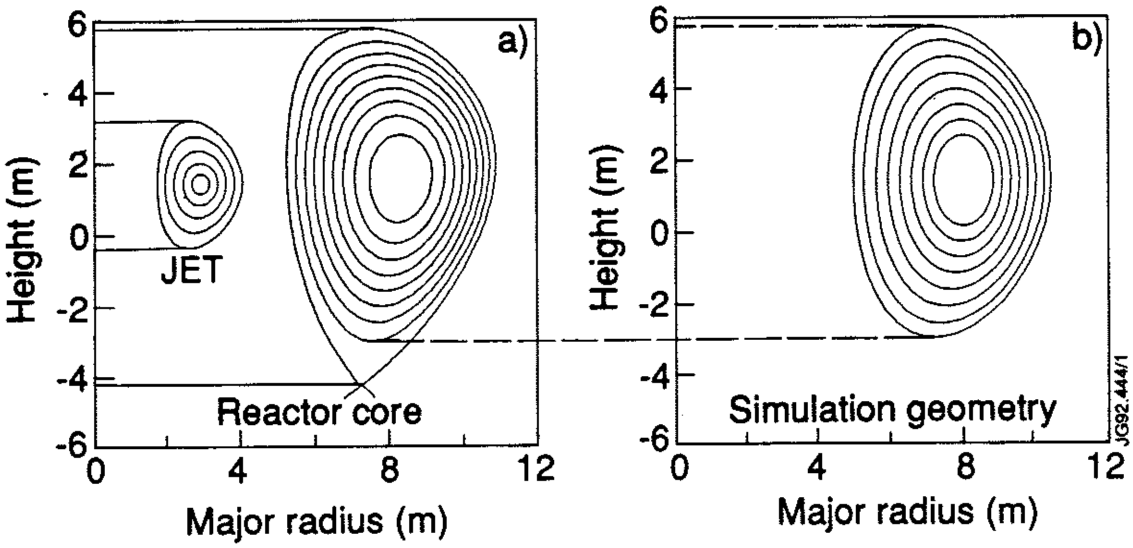

Fig. 1: (a) Typical configurations for JET and a reactor core and (b) the simulation geometry used to solve the full energy and particle transport model.

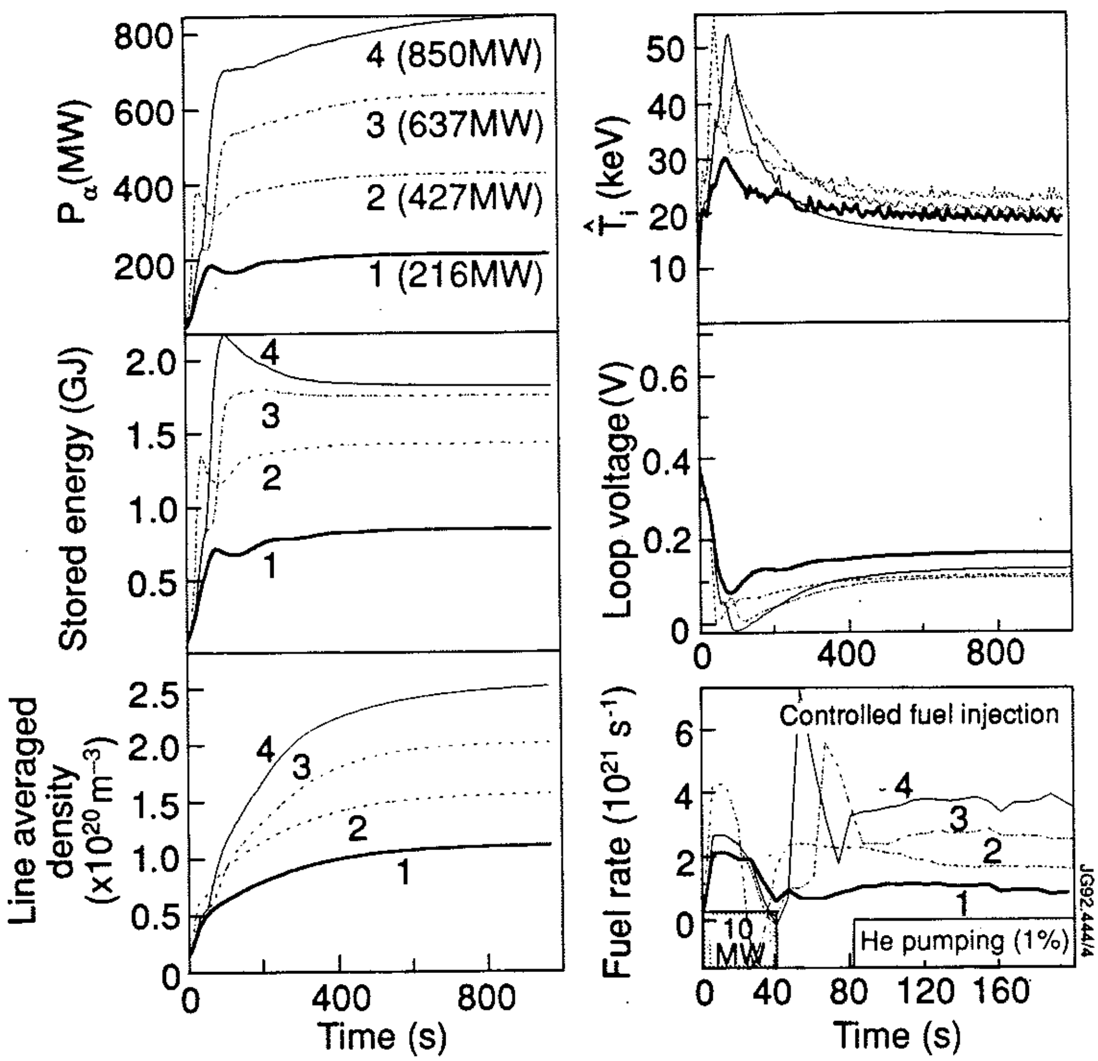

Fig. 2: Simulations of the start-up and burn control of a reactor core at various power levels. Shown are the temporal evolution of α-power, stored energy, line-averaged density, central ion temperature and loop voltage. The RF heating needed for ignition, the fuel injection (controlled by feedback on the power produced) and the helium pumping are also shown.

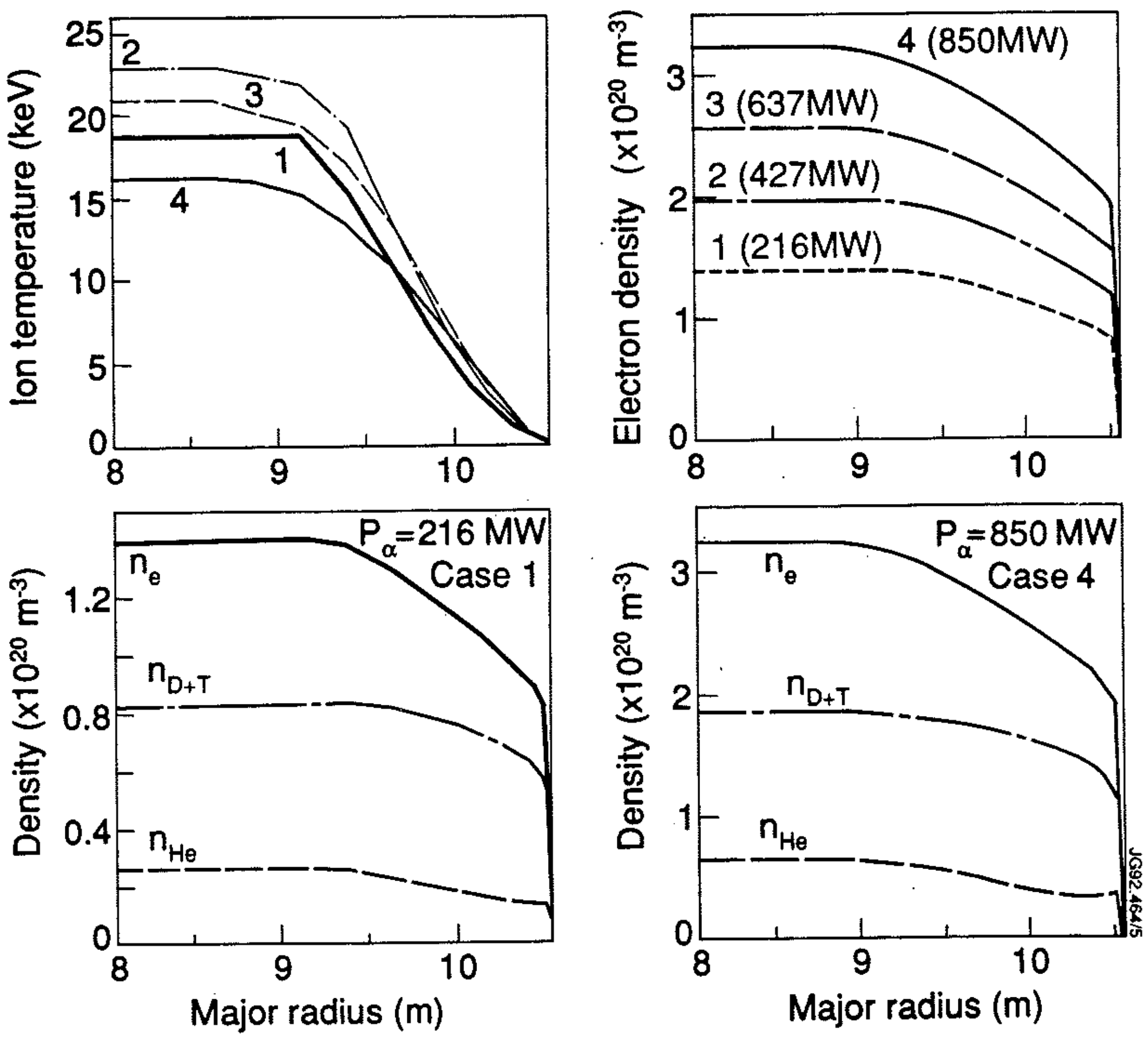

Fig. 3: Steady-state profiles of ion temperature and electron density calculated for a reactor core at various power levels. Shown also are the density profiles at nominal α-powers of 0.2GW (Case 1) and 0.8GW (Case4).

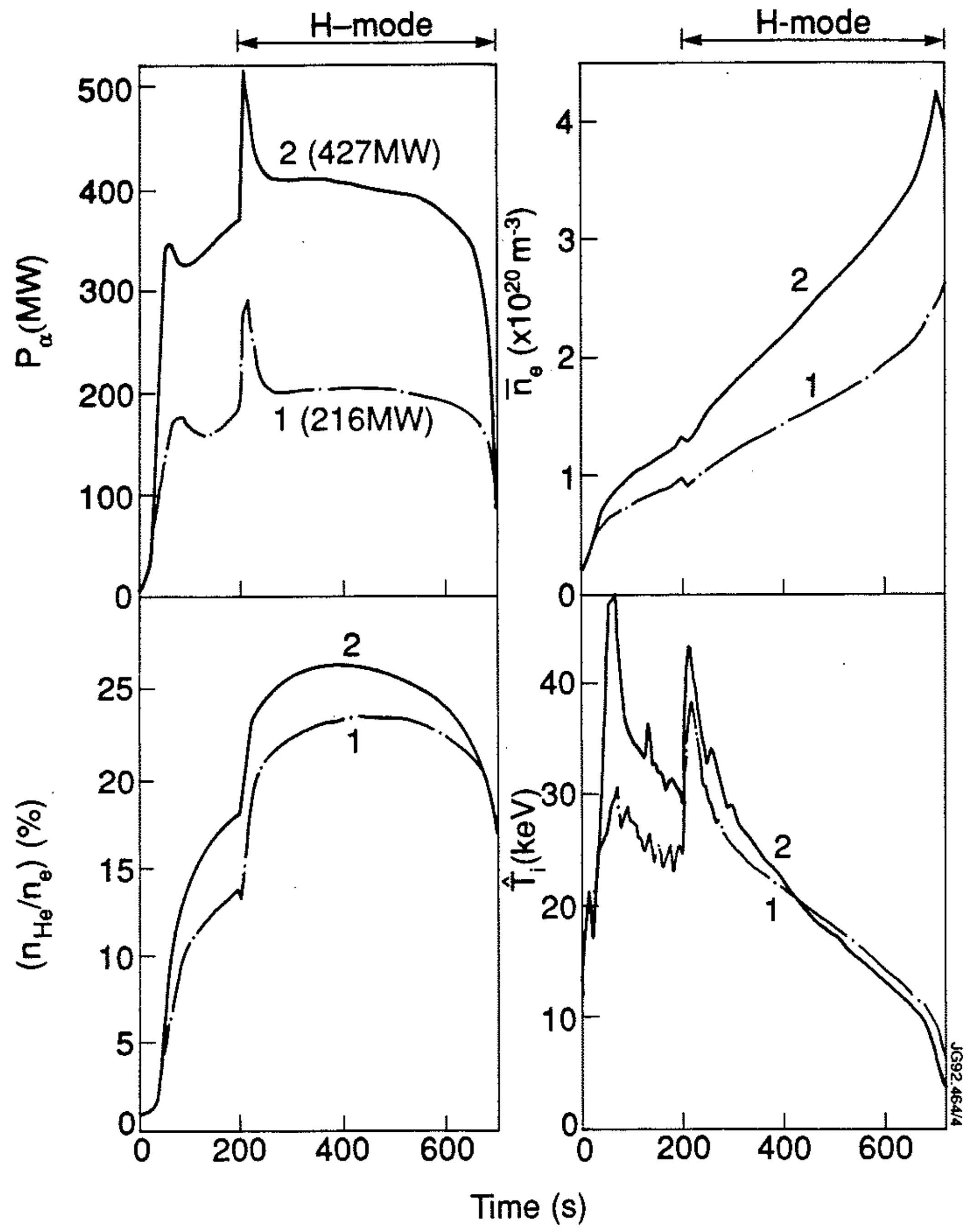

Fig. 4: The long term deficiencies of helium poisoning are shown by modelling the H-mode during the ignited phase of a reactor core. The nominal α-powers are approximately 0.2GW (Case1) and 0.4GW (Case2).

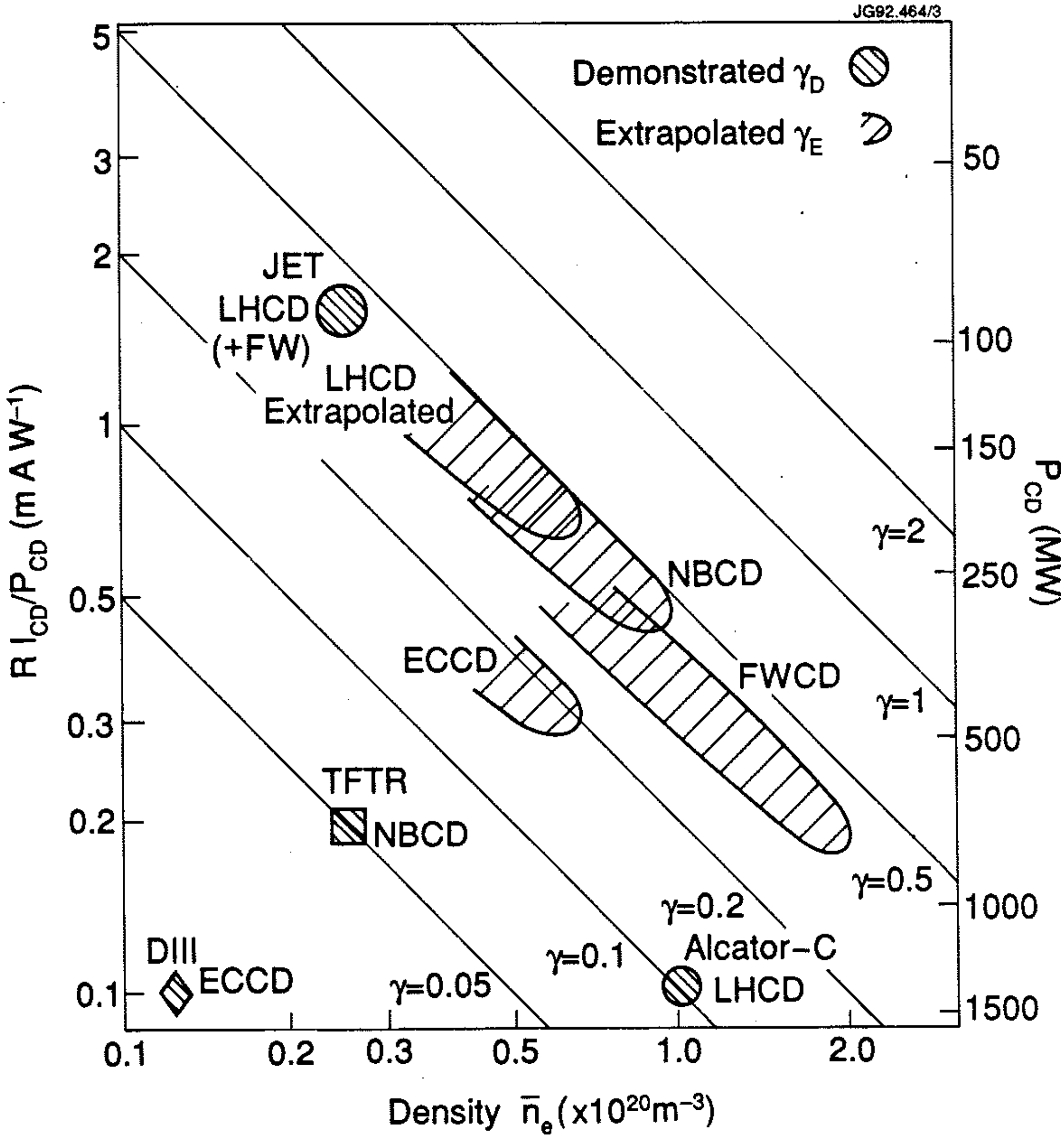

Fig. 5: Current drive efficiencies, demonstrated and extrapolated, for various techniques of non-inductive current drive. Also shown on the right-hand ordinate is the launched power required for 18MA of non-inductive current drive in a tokamak with R=8m. (LHCD - Lower Hybrid Current Drive; FWCD - Fast Wave Current Drive; NBCD - Neutral Beam Current Drive; ECCD - Electron Cyclotron Current Drive).

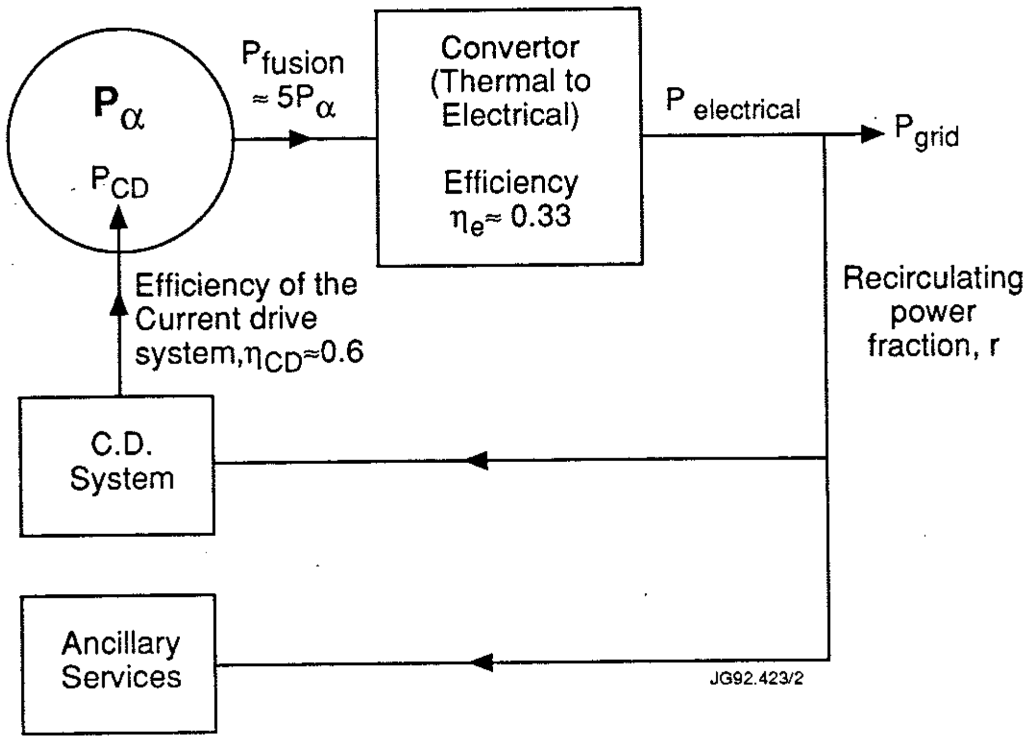

Fig. 6: The power requirements for full non-inductive current drive in a reactor.

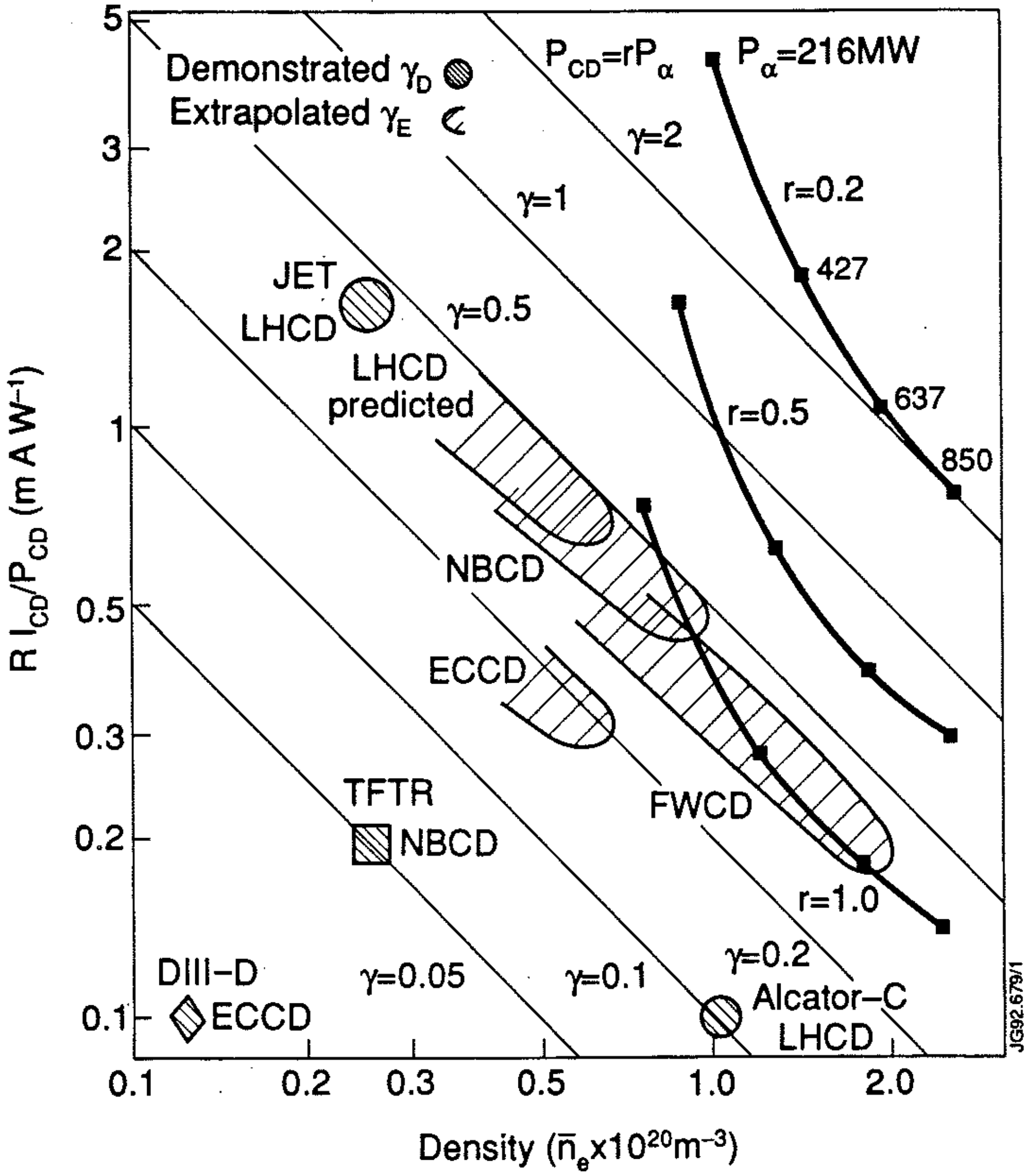

Fig. 7: The operating domain (heavy black lines) for full non-inductive current drive in a tokamak reactor with R=7.75m, a=2.8m, B_T=6T, I=25MA, κ=1.6 for various assumptions about the recirculating power fraction, r. (LHCD - Lower Hybrid Current Drive; FWCD - Fast Wave Current Drive; NBCD - Neutral Beam Current Drive; ECCD - Electron Cyclotron Current Drive).

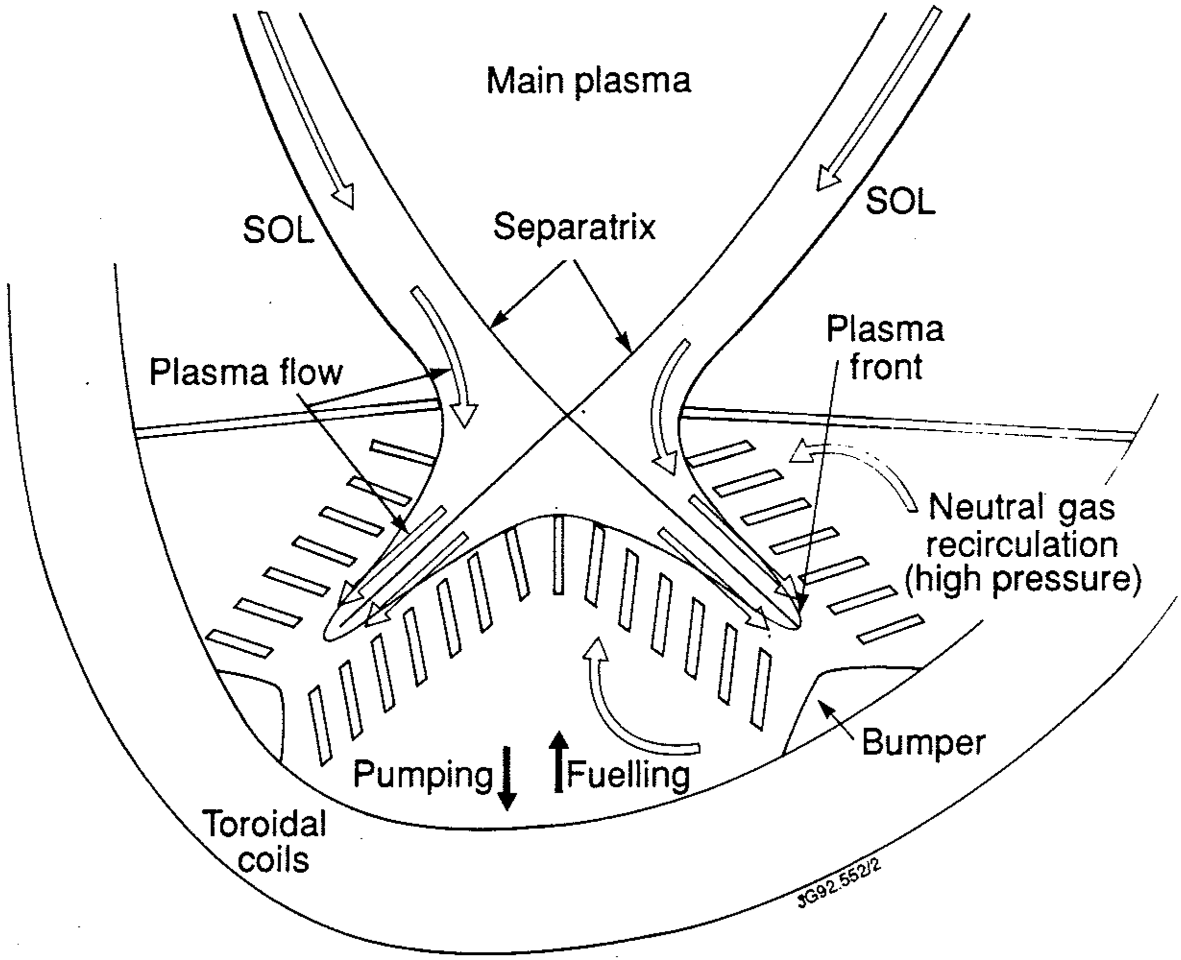

Fig. 8: An advanced divertor concept.

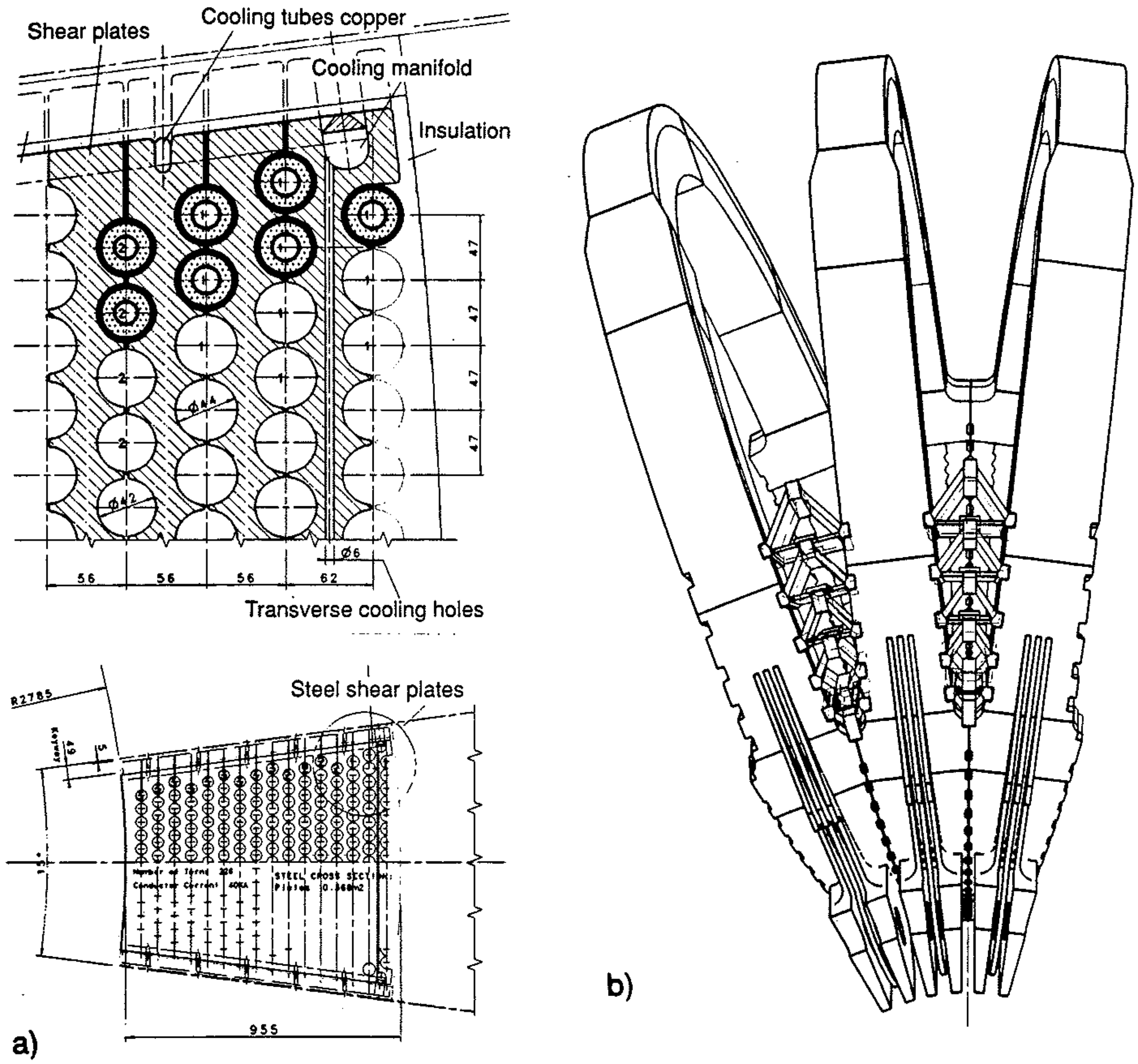

Fig. 9: A concept for the mechanical structure.

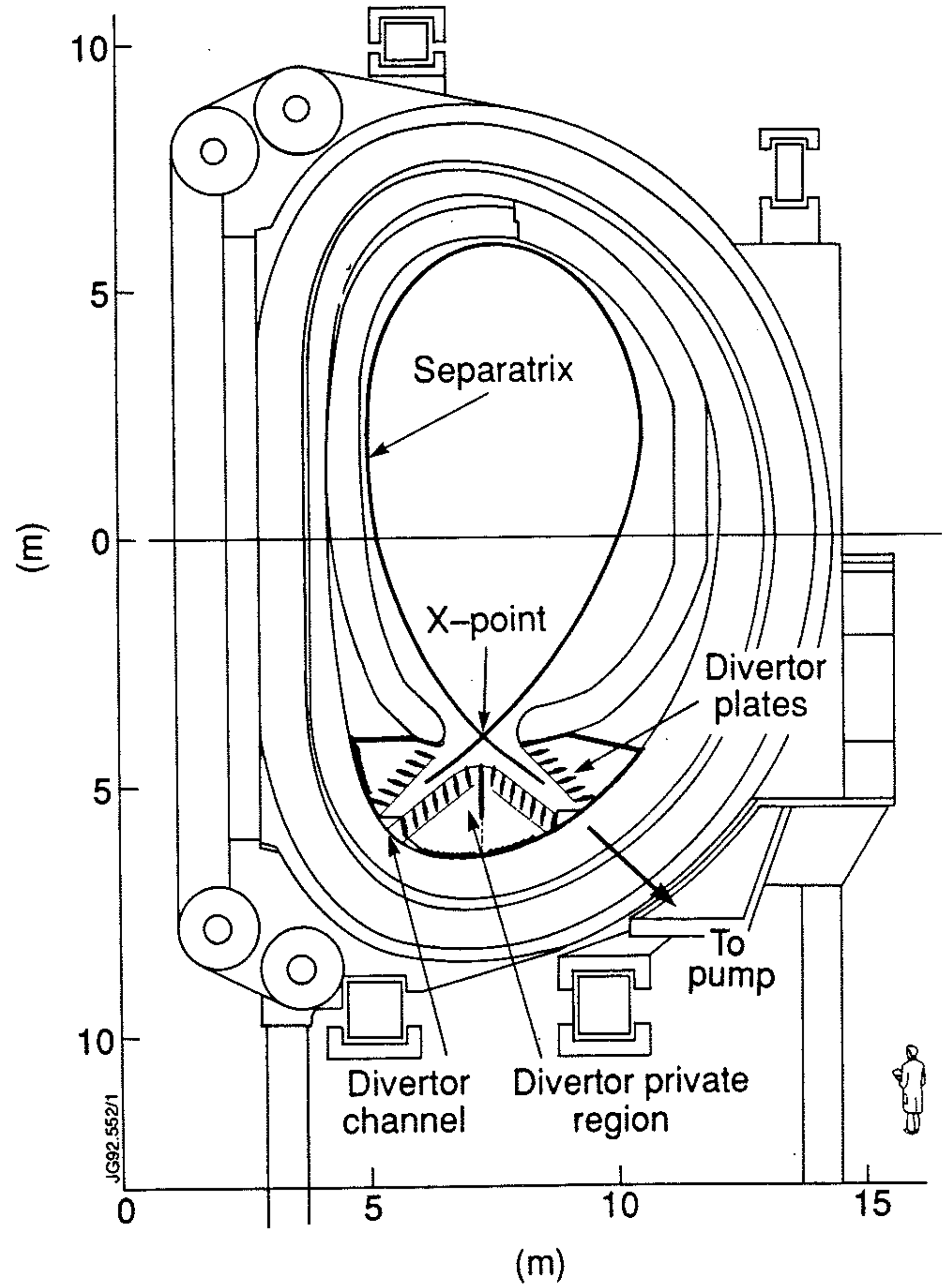

Fig. 10: A possible Next Step configuration.

Appendix I

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