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P.H. Rebut, K.J. Dietz, P.P. Lallia
and JET Team

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

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EXPERIENCE WITH WALL MATERIALS IN JET AND IMPLICATIONS FOR A FUTURE IGNITED TOKAMAK

P.H. Rebut, K.J. Diëtz and P.P. Lallia

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

Abstract

A variety of materials have been used in JET for wall protection and high heat flux components. The machine initially operated with metallic walls, but the inner surface of the vessel ($\sim$200m²) is now covered to more than 50% with fine grain and carbon fibre reinforced graphite tiles. The remaining wall area is carbonized. This paper presents the materials behaviour in the presence of plasma; their influence on plasma properties; the conditioning methods employed; a discussion of future enhancements of inner wall components and the planned use of beryllium as an alternative to the present concept of an all-graphite machine. It is essential for the further development of fusion that the experience gained in JET is transferred to the next machine, which should produce a burning plasma on a scale comparable to a reactor. Such a proposed machine is a single null divertor tokamak with the following parameters: 3m plasma minor radius, 7.5m major radius, elongation $\sim$2, aspect ratio $\sim$2.5, toroidal field 4.5T, pulse duration 2000s and fusion power up to 5GW. The underlying physics for the choice of these parameters and the basic design is presented. Based on this concept and the experience with materials in JET, the lay-out of inner wall components, as well as possibilities for plasma exhaust and refuelling are discussed. As a consequence of this assessment, open questions with respect to the physics of the plasma edge and materials properties are highlighted.

Introduction

The study of plasma facing components requires a reference design with data on machine configuration, operation modes, plasma edge parameters in normal and abnormal operating conditions, materials data and power levels.

Tokamaks may employ limiters and/or divertors in a variety of configurations and the plasma facing components must be designed within boundary conditions imposed by the envisaged machine configuration. Compromises will be required to take into account access restrictions for installation, required compatibility with remote maintenance, possible interference with diagnostics, and the need to install sufficiently large surface areas to reduce thermal loads to acceptable levels.

Requirements concerning power handling capabilities and resistance against eddy current forces result not only from the different plasma configurations (X-point, limiter, inner wall discharges) but also from the various operation modes. For example, JET employs a variety of plasma currents, toroidal fields, elongations and heating powers. The inner wall components must cope with all of these. Knowledge of edge parameters is essential for the design of elements in direct contact with the plasma. For a maxiumum tolerable heat load on inner wall components, the scrape-off thickness and the power flux define the shape of the surface intercepting the plasma. Particle fluxes and energies at the edge influence impurity release rates and determine the lifetime of high-heat-flux components.

A variety of materials have so far been used for plasma facing components in JET. They range from nickel based alloys for the wall protection used initially [1], to carbon fibre composites (CFC) [2] for areas of the vessel which are exposed to shine-through from the neutral beams. In future, it is planned to employ annealed pyrolytic graphite for the leading edge of a pump limiter [3] and the use of beryllium [4] has been proposed as an alternative to graphite for limiters and wall protection.

Wall Components

In contrast to beam line components of the JET Neutral Injectors [5], which are actively cooled, all wall components in JET are inertially cooled during discharges. Consequently, the possible pulse duration is limited by the surface temperature, which is determined by the thermal diffusivity of the plasma facing material. Material in contact with the plasma (so far, graphite) is cooled by radiation in the 10 to 20 minute intervals between discharges. This limits the possible duration of the interaction with the plasma to times of about 10s at full power.

From the start of JET operation, fine grain high-purity graphite has been employed for the limiters. Based on good experience with graphite 5890 PT in TFR [6], this material was chosen initially. However, difficulties were experienced with the availability of the required six tonnes of graphite. Therefore, a study was initiated [7] and EK 986 was identified as an alternative. Data for the various materials used, and for others of possible future interest have been published previously [8]. During the ohmic phase, four discrete limiters [9] were employed, and later, eight were installed when additional heating reached levels exceeding 10MW. Extensive operational experience exists [10], and no failures occurred. Macroscopic surface damage was observed once only over an operating period of four years [11].

Belt limiters [12] are now in use, whose power handling capabilities exceed 40 MW for ten seconds. Again, fine grain graphite was selected as limiter material. Operational experience is good, but so far the belt limiters have received only ∿25% of the designed heat load.

For the walls and wall protection [1], Nicrofer 7216 (equivalent to Inconel 600) was used initially. After a short operational period, melting of the surface of the inboard wall resulted from the interaction with runaway electrons which were created during disruptions [13]. This was avoided by changing to graphite as a low density low-Z material for wall protection. The energy deposition which was localized near the surface of the high-Z material, is now distributed over a larger volume. For typical runaway energies of 20-40MeV, the electron penetration depth is now about 15cm.

Damage was observed again when the wall protection was used as a limiter. The protection does not form a perfectly smooth boundary and the octant-joints [2], especially, protrude a few mm. Overloading occurred in these areas at the midplane of the inboard wall, and as a consequence, the tile material was changed to a carbon fibre composite (DMS 678). Operational experience is good, so far; in cases of runaway impact, the graphite is eroded but remains structurally intact. In addition, carbon fibre reinforced graphite is used in areas where the shine-through from neutral beams is intercepted by the inner wall. Power loads in excess of $30 MWm^{-2}$ can be sustained for a few seconds. Fine grain graphite would fail in fracture, under these conditions.

In addition, damage to the top and bottom of the vacuum vessel was observed due to vertical instabilities. Eight poloidal rings (consisting of graphite) were installed to protect the vessel. These initially covered the octant-joints [14] and were used as well as energy dumps for X-point operation. The available surface area was not large enough and therefore 40 poloidal rings are now installed. These cover the octant-joints and bellows protection plates [1]. The power handling capability is still limited to $\sim$40MJ in 2s. Improved water cooled dump plates, covering the top and bottom of the vessel, will be available for X-point operation in 1989. Fig.1 shows the inside of the JET vessel in its present state.

Wall Conditioning Techniques and Impurity Behaviour

Typical procedures, following opening the vessel to air and prolonged personnel access, are to rinse the inside of the vessel with demineralized water followed by a twenty-four hours bake-out under vacuum at a temperature of 300°C. Then, R.F. assisted glow discharge [15] in hydrogen (deuterium) is carried out for about seventy hours. This operates typically at current densities of $1\mu Acm^{-2}$ at a cathode fall of 300V, vessel temperatures of 300°C, neutral pressures in the lower 10^{-3} mbar range, and with pumping speeds of $6200ls^{-1}$ for hydrogen.

To reduce the metal contamination, plasma activated deposition of carbon from a deuterium-methane mixture was used (carbonization) and the vessel was covered by carbon films [16,17] of thicknesses up to $\sim 1\mu m$.

With increasing amounts of solid graphite in the vessel, it became apparent that the glow discharge conditioning with hydrogen (deuterium) became less effective and the oxygen concentration in the plasma increased to values of 1-2%. Residual gas analysis during conditioning showed that neither water nor carbon oxides were significant components in reaction products removed. Thes consisted mainly of hydrocarbons (acethylene $\sim$50%, methane 30%, ethylene balance).

Conditioning the vessel with tokamak discharges in helium resulted in decreased oxygen levels, and therefore, following a recommendation [18] of TEXTOR, glow discharge cleaning in helium was introduced. With this method, oxygen is removed in the form of carbon-oxides and average removal rates of up to 10mbarl were observed for glow discharge times of ten hours. There are indications that improved plasma performance can be achieved using this method.

Carbonisation was frequently carried out to reduce the concentration of high-Z impurities at the plasma centre. During a transient phase (20-200 discharges), depending on the thickness of the deposited layer [19], metals are suppressed to concentrations below 10^{-4} and oxygen and carbon become the dominant impurities even at low densities [20].

For initial operations of JET, when there was only a negligible amount of graphite in the vessel, typical impurity concentrations in the plasma were 2-3% carbon, 1-2% oxygen and $\sim$0.2% metals. With increasing amounts of graphite inside the vessel, following helium conditioning and carbonisation, typical concentrations of metals (10^{-6} – 10^{-4}) and oxygen ($\sim$0.5%) decreased whereas carbon increased (typically 5%).

The presence of carbon in the plasma was expected and cannot be avoided, whereas the source of oxygen contamination could not be identified unambiguously. However, it appeared to be related to the amount of graphite in the vessel and was therefore attributed to the absorption of water in the graphite and its release during discharges.

The values of the effective charge, Z_{eff}, are close to two at the density limit, which indicate a dilution of about 20% in the core of the plasma [21]. This situation can be improved by pellet injection, in which Z_{eff} values close to unity can be obtained [22], where the mechanism is dilution of impurities by increasing the hydrogen (deuterium) concentration. With helium glow discharge conditioning, Z_{eff} values less than two appear to be possible even in 5 MA discharges with additional heating.

The density limit of ohmic discharges was increased during neutral beam injection by about a factor of two, but initially it was not possible to achieve a similar increase with RF-heating. Following heavy carbonisation this behaviour changed and it is now possible, with a reduction of high-Z impurities, to run RF-discharges at densities close to those possible with neutral injection. The main source of high-Z impurities are the antenna screens which release nickel into the plasma. Carbonisation suppresses this metal source temporarily, but it has draw-backs in problems with density control and recovery from disruptions. Therefore, it is proposed to replace the nickel screens with beryllium, and due to their higher electrical and thermal conductivity, active screen cooling is no longer required.

Abnormal Operation Conditions

It is generally observed in limiter tokamaks that during a disruption the plasma thermal energy is dumped on the limiter and about 50% of the magnetic energy of the plasma current are dissipated in the vessel walls. The time-scale ranges from hundreds of microseconds for the thermal dump to tens of milliseconds for the deposition of the magnetic energy. In addition, runaway electrons are produced in the decaying plasma by the voltage spike accompanying the current decay. The power loading of inner wall components is rather localised during such an event and can only be accommodated by sacrificing parts of the surface which evaporate. In JET, a deposition of up to 500 MJm^{-2} was observed [7]. The main reasons for disruptions are: due to exceeding the density limit (power radiated outside the $q = 2$ surface not balanced by power input) creating locked modes ($m = 2$, $n = 1$); or by objects falling through or being injected into the plasma. The latter can be parts of the wall protection tiles, flakes of redeposited material, plasma probes, or molybdenum during neutral beam injection originating from flaking molybdenum coatings on beam dumps. During high current radial disruptions (normal disruption) (up to 7MA) or vertical instabilities, the forces acting or the vacuum vessel are calculated to reach values of 900-1200 tonnes. Without additional stiffening of the vessel by internal rings and outer supports, the material of the vacuum vessel (Nicrofer 7216 LC) can yield. In this case, the vessel lifetime would be reduced by low cycle fatigue to about 1000 disruptions at full current. There are several possibilities to minimize the occurrence of disruptions (e.g. not to operate close to the density limit, to stabilize the $m = 2$, $n = 1$ modes before they lock and to have strict quality control for all in-vessel components).

Wall Pumping

With the introduction of graphite as wall protection, density pump-out occurred during discharges when the wall protection was used as a limiter [14,21]. This phenomenon could be used to terminate beam-heated discharges with densities above the ohmic density limit. Furthermore, helium conditioning of the inner wall [23] increased the pumping capability and made it possible to operate at low densities and high heating power in the hot ion mode. Wall pumping was not only observed with discharges at the inner wall. For divertor discharges in JET, when the bellows protection was used as plasma facing component, strong density pump-out was also found.

Mechanisms governing wall pumping are not yet fully understood. A variety of models exist ranging from assuming transient effects [24,25] similar to hydrogen pumping by metal walls [25,26], to the co-deposition of hydrogen and carbon in form of saturated H-C films [24,27]. To date, it is not clear what is the dominant process. Experimental evidence exists from JET, derived from gas balance measurements in ohmic discharges, which supports the co-deposition model. Only 30-40% of gas input during discharges can be found in the exhaust gas even after waiting for times up to 24 hours. To date, it is concluded that most of the deuterium introduced into the JET vessel remains in the form of a deposited layer of hydrocarbons.

Experiments on particle balance of heated and pellet fuelled discharges are being carried out. First results indicate a higher retention, as in ohmic discharges. If this can be further corroborated by experiments, there will be serious implications for the tritium inventory of machines with graphite walls.

Future Developments

During the 1988/89 shutdown starting at the end of 1988, major new components will be installed in the vessel. To stabilize the m=2, n=1 intstability mode, eight saddle coils will be installed each with an area of $\sim 6m^2$ covering one and a half octants. The coils each have three windings and will be driven with voltages up to 5kV, currents up to 5kA and frequencies from DC to 10kHz. They are bakeable to 773K and will be protected from the plasma by graphite. X-point dump plates will be installed at the top and the bottom of the machine, whose surface in contact with the plasma will be $\sim 26m^2$. These consist of a water-cooled support structure (Nicrofer 7612) covered with 25mm thick CFC graphite tiles. These tiles are pressure contacted to the backplate, and will be inertially cooled.

Existing wall protection will be modified to blend with the saddle coils and the dump plates. To prevent runaway damage and to allow for higher power loads during normal operation, more CFC graphite will be used and protruding edges will be removed from the inboard wall.

In early 1990, two pump limiters will be installed. For the leading edges, inertia cooled annealed pyrolytic graphite will be used to sustain the expected power loads of 30 to $50MWm^{-2}$ for times of 3-10s. The surface temperature will not exceed 2200K in these conditions. Bonding of the pyrolytic graphite to the watercooled support structure is a problem still to be solved. Fatigue lifetime and degeneration of the material under high loads needs to be tested. The pump limiters will operate as stand alone limiters for a few seconds with a particle removal rate of 9-12% of the particles reaching them, in load sharing with the inner wall with removal rates of 4-6%, together with the belt limiter at a removal rate of 1-3% and with L-mode plasmas. Removal rates of $\sim 1\%$ are sufficient to balance the density rise due to neutral injection (20MW at 140keV).

The Use of Beryllium

Preparations have been made to use beryllium as an alternative to graphite, in case major difficulties are experienced with graphite under high power loads in discharges of long duration. Problems might arise from impurity production, dilution, density limits, density control and tritium inventory, which could be less severe with the use of beryllium. High power long pulse operation in JET has not yet started and it is too early to arrive at firm conclusions about the future use of graphite.

The antenna screens which have provided a continuous source of high-Z impurities may be exchanged for beryllium which then will alleviate the need for carbonisation of thicknesses up to 1.5μm. This process is unsatisfactory as the density and isotope control become extremely difficult and, in addition, is detrimental to the operation of neutral beam injection. The carbon deposited on the copper duct scrapers must be removed manually after each carbonisation, otherwise it is transported by the beams into the plasma and leads to disruptions.

Beryllium could be used at any time from the beginning of 1988 onwards. The graphite belt limiter and antenna tiles would be exchanged for beryllium and all other areas of the vessel in contact with the plasma could be covered by a beryllium layer of ∿10μm thickness. Alternatively, only evaporation could be used. Four evaporators capable of evaporating 1 kg beryllium each are being tested and will be available for that purpose. For a few tens of discharges, the vacuum vessel would have a surface consisting mainly of beryllium. Details of the planned beryllium operation have been reported previously [4]. The use of beryllium as a getter material has been strongly supported by gettering experiments with boron in TEXTOR [18], where the supression of oxygen impurity has proven to be beneficial for the plasma.

Extrapolation into the Future

Experience gained in JET can only be projected into the future if the results obtained are relevant to the requirements of a next generation machine. It is clear that the problems already encountered in JET, will further increase when the plasma and wall parameters of a fusion reactor are encountered. The next step tokamak will have to tackle fully those problems, since it should be relevant in size to a reactor and provide an ignited plasma for a duration of at least days in semi-continuous operation. The scientific and technical aims of such a machine should be to study a burning plasma, to test wall technology, to provide a test-bed for breeding blankets and above all to demonstrate the viability of fusion and its potential as an energy source.

The physics results obtained so far in JET have shown or confirmed several features of tokamaks, specially relating to heat and particle transport:

- With ohmic heating only, electron energy transport is anomalously high in comparison with expectations of neoclassical theory and shows a different scaling with plasma parameters. The ion energy and particle transport are also anomalous in JET;
- The confinement properties degrade further when additional heating power is applied (so far almost independently of heating method). With ICRF heating, plasma heating occurs mostly through very energetic ions, which exceed 1 MeV in energy at high power input, and which supports the idea that confinement degradation will also result from α-particle heating;
- The electron temperature profile is resilient to changes in the input power, while the density profile can be changed more readily;
- The improved confinement observed in JET H-mode plasmas exhibit the same degradation with increasing additional heating power as L-modes. The consequences are that an H-mode may help to go through the "ignition pass", if the required additional power is moderate, but also that the benefit of the H-mode would vanish when the reactor delivers its full output power;
- Global confinement in JET seems better described by an offset linear dependence between the energy and the power [28]. The linear energy increment is proportional to the current as long as the volume in which internal disruptions occur is not too large (i.e. when $q_{cyl} \geq 3$);
- The maximum density without disruption depends on the cleanliness of the

plasma. This can be increased by pellet injection and by increasing the input power. The prospects of increasing the density above the normal Murakami limit in the presence of powerful α-particle heating looks reasonably good.

An interpretation of JET confinement results, based on the existence of chaos in the plasma, has led to local transport laws which can describe JET results rather well [29,30]. Expressed in terms of global scaling laws, the agreement is also reasonable as the fit found with the JET data points applies also to other Tokamaks.

The relationship between confinement laws and the plasma beta (ratio of kinetic to magnetic energy) is difficult to identify, especially in the heat diffusivity coefficient. When the power input is very large compared to the ohmic case (i.e. the relevant case for a power reactor), the confinement time according to this interpretation should scale as:

$$\tau_E \propto \beta^{-\alpha}\ I\ a\ R^{1/2}$$

where I is the plasma current, and R and a the major and minor radius, respectively. The exponent α was taken to be zero in the JET simulations and values above unity can be reasonably excluded. At the Troyon β limit, $\beta = g\,\frac{I}{aB}$ [31], the ignition product becomes:

$$nT\ \tau_E \propto g^{\ (1-\alpha)}\ (q\frac{R}{a})^{\alpha}\ I^2 B R^{1/2}$$

where q is the safety factor and g is a constant.

As the product $nT\tau_E$ is a good measure of the power amplification of a thermonuclear D-T plasma (as long as the temperature is larger than 7keV), this expression shows the importance of the current capability of the machine.

Based on this analysis of results obtained in JET, it should be difficult to achieve ignition in most of the tokamaks so far being discussed or designed. To demonstrate the viability of thermonuclear fusion and its potential as an energy source, a tokamak is proposed 2-3 times larger than JET in linear dimensions, at a magnetic field of $\sim$4.5T, with a current capability reaching 30MA. A single null divertor configuration should be used to ensure helium exhaust and to possibly benefit from an H-mode in enlarging the ignition domain. Confinement degradation for α-particle heating and L-mode behaviour is taken into account in the design. The additional heating power requirements are relatively low ($\leq$ 50MW). The maximum thermal output of such a machine depends on the maximum density which can be achieved and on the ability to limit dilution of the plasma by helium and impurities. With Z_{eff} < 2 and a Murakami parameter $M = \bar{n}\frac{R}{B} = 15$, the maximum fusion power should be around 4GW, with still a safety margin on the limit in β.

The proposed machine is a true "Thermonuclear Furnace" because it has all the elements in contact with the burning plasma inside the vacuum vessel which will be required for a reactor. On the basis of the proposed design, problem areas related to wall components can be assessed and, based on JET experience, possible solutions can be discussed.

The Thermonuclear Furnace

The design should present only minimal technical risks. It uses conventional or proven technology as far as possible. The tokamak cross-section is shown in Fig. 2. The machine consists of twenty identical sectors, of which each one incorporates a toroidal field coil, the mechanical structure, part of the vacuum vessel and one integrated unit. Due to the box-like shape, the sectors are strong enough to withstand forces during a disruption at full current. The sectors would be joined by welding lips and so form the vacuum enclosure. The material of the wall would be stainless steel (AINSI SS 316 or equivalent) of about 3 mm thickness. The wall would be watercooled, and at the same time, the water forms the blanket of thickness 40-60 cm.

The plasma is in a single null divertor configuration. The divertor plates are exchangeable and it is possible to incorporate blanket test modules into the divertor supports. The plasma occupies about 50% of the toroidal field volume. Thus, a drastic relative reduction in costs could be achieved. In this proposal, copper coils are used to minimize the technical complexity and risk by reducing the number of concentric vessels and structures and by decreasing the neutron shielding requirements. Table I lists the main parameters of the furnace.

Wall Concept

The design of the proposed machine aims at a plasma size comparable to a reactor. This results in modest requirements for the power handling capabilities, so that there is no need to develop new techniques for high heat flux components. The wall concept is based on graphite components covered with a renewable thin beryllium layer in order to block the chemical reactivity of graphite. Radiation cooling for inner wall components is envisaged as in JET and active cooling for divertor components. Heating by neutrons (nuclear heating) will produce the main thermal load for the inner wall components and contribute significantly to the divertor loads.

The power output is rated at 0.5-4GW depending on the achievable central density. For further evaluation, a power production of 2.5GW is assumed. 2GW is emitted as neutrons, and the rest as α-particles. Power losses from the plasma are by radiation and conduction; convection and charge exchange neutrals are neglected. The radiation from the plasma volume is mainly due to bremsstrahlung and it contributes $\sim$20% to the power balance. It is assumed that low-Z radiation from the plasma edge contributes another 20% and a further 20% would be radiated from the X-point during L-mode operation. Half of the latter would be intercepted by the divertor plates which as well would receive 40% of the total power by conduction.

Table II sets out the basic data for loads on wall components. These are averaged assuming equal power distribution on the respective components. Following experience on JET, all wall and divertor components would be low-Z materials (i.e. carbon, beryllium, boron and lithium or combinations).

Wall Protection

Fig.3 shows a diagram of the inboard wall. The wall protection consist of graphite tiles, which is radiation cooled. Pressure contacts are not considered to be sufficiently reliable. There would be two types of protection. The first forms a poloidal ring of 300mm width, in each sector, in a configuration reminiscent of the belt limiter in JET. The tiles radiate to water cooled fins which form part of the blanket. The thickness of each graphite tile would be 30mm and the depth 50 to 100mm. The cooling fins are recessed by 20mm, and therefore, 20mm of graphite can act as a sacrificial layer during disruptions. The second purpose of the tiles is to shield the toroidal field coils which are partly situated behind a gap in the blanket. Tile temperatures are mainly determined by nuclear heating and should reach $\sim$1500K at the front surface. Situated between the rings is further protection over a height of 5m in the form of tiles similar to the wall protection in JET. These are well radiation cooled and have a thickness of 20mm. Their front temperature will also be 1500K. The rest of the inside of the vessel would be carbonized. The material for the tiles would be carbon fibre reinforced graphite of a type similar to that presently used in JET. It is not anticipated that the inner wall protection presents any major problems during normal operation conditions. Fig.4 shows a three-dimensional view of the inboard wall.

Divertor

The divertor presents the main technical challenge in the design of inner wall components. This challenge is reduced to a level which can be handled 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 the existing coil set (or preferably sets of internal coils), with minimal effects on the overall plasma geometry. Alternatively, the movement can be achieved by oscillating the divertor components horizontally. The main advantages of this scheme are:

- no need for absolute positioning of the divertor components with an accuracy of millimetres as in JET. Feedback control could be used;
- in spite of high local thermal loads the average loads remain small;
- heat transfer to the cooling medium corresponds to the average loads;
- lifetime increases according to the ratio of peak load to average load.;
- to compensate for erosion, material (beryllium, hydrocarbons) can be injected which would be deposited in areas with low loads;
- deposited and redeposited layers are burnt off before they can flake and before large amounts of tritium could accumulate.

For the movement, the limiting factor is the sweep frequency which must be low enough so that additional coils are not required in the vacuum vessel, and must be high enough to minimize thermal stresses in the divertor material. As an alternative to this movement, use of liquid metals (lithium or beryllium) could be envisaged to act as a divertor target producing a high density ionized shield (plasma plugging, if possible). In this paper only the relative movement between plasma and divertor will be discussed.

Fig.5 shows a cross-section through the divertor region. The divertor plates extend above and below the X-point. In the down-stream region, additional wall protection is installed (cone between the divertor plates) to avoid direct line-of-sight between the plasma and the blanket and to protect the blanket against damage by loss of plasma control during a vertical instability. The divertor receives only a moderate average power level of 130 Wcm^{-2} from radiation and conduction. The X-point can be moved horizontally and vertically to distribute the power load evenly over the whole divertor surface. As H-mode operation is not envisaged, the outer and inner part of the divertor plates can be used to this purpose.

The material proposed for the divertor plates is carbon fibre graphite of 40mm thickness. It is brazed in blocks of 10 x $10mm^2$ surface area to water cooled copper plates of 10mm thickness. The average heat flux (conduction plus nuclear heating) is $\sim 1.8MWm^{-2}$. This can be removed without having to resort to two-phase cooling. In the proposed design, the peak loads on the surface during the sweeping of the X-point should be about x5 higher than the average load ($\sim 10MWm^{-2}$). For the calculation of thermal response, the high thermal conductivity of the fibre graphite was not taken into account as it degrades at low neutron doses ($\sim 10^{20}cm^{-2}$) to that of fine grain graphite [32]. For a sweep frequency of 1Hz, the thermal stresses at the surface will be about 25MPa in compression (limit 100MPa) and the peak temperature will be $\sim$1400K. The material thickness is mainly limited by the average load and the resulting front temperatures. An alternative set of parameters would be: 60mm material thickness, sweep rate 3Hz, surface temperature 1600K and thermal stress 55MPa. More detailed calculations should take into account the annealing of radiation damage and a varying thermal conductivity inside the material.

It is proposed that the graphite divertor should be covered with a thin layer of beryllium to employ the good properties of both materials. A beryllium carbide layer would be formed [33] at temperatures above 700K. It is envisaged that such a scheme should be validated in JET.

The lifetime of the divertor plates is determined by the erosion due to particle impact. A detailed modelling of the plasma edge and the divertor region is required to assess the particle fluxes and energies, but is presently not available. Therefore, a worst case assessment is made. The energy (∿300eV) and the temperature (∿750K) of the maximum sputtering yield is taken for the sputtering of carbon by deuterium. For 300eV ions and an energy transmission factor of 17 [34], the particle outflow from the plasma is 7×10^{23} particles per second.

With a sputtering yield of 0.2 [35], this is equivalent to a removal of about 35 monolayers per second. Neglecting any increase in lifetime due to lower particle energy, possible supression of chemical activity and redeposition, the lifetime would be at least two months of continuous operation at a level of 2.5GW. It should be mentioned that H-mode operation might immediately halve the divertor lifetime, as the outer part might not be allowed in contact with the plasma and the loads would be doubled on the rest.

It appears that a fast movement (sweep frequency >10Hz) of the X-point is difficult to achieve and therefore peak loads up to a factor five higher than the average loads can be expected for times of at least 0.1s. These should be taken by the thermal inertia of the divertor components and then be transferred to the cooling water at the moderate heat transfer discussed above. Lifetimes will not be affected as average loads do not change. Experience in JET shows that graphite fibre material is well suited to operate in such a way.

For 2.5GW thermal output, the neutron flux density is $5.3 \times 10^{13} cm^{-2} s^{-1}$. To reach a dose corresponding to one displacement per annum, the plasma should burn for 216 days continuously ($1dpa = 10^{21} cm^{-2}$). This is in excess of the predicted lifetime of the divertor components in the above assessment. In the long term, it would be necessary to validate schemes like the high recycling divertor or to develop liquid or plasma or gaseous energy dumps to arrive at lifetimes of several years.

In this proposal, Beryllium as a metal cannot be used as wall protection because the temperatures required for radiation cooling would be too high. In addition, it can not be envisaged for the divertor plates. The stresses induced during sweeping of the X-point ($10Wm^{-2}$, 235MPa in compression after 0.03s) requires execution of a cycle in a fraction of this time to avoid plastic deformation. Sweep frequencies higher than 10Hz are not compatible with internal coils (300V/turn required at 10Hz) and, therefore, the use of massive beryllium is not proposed.

The design of the inner wall components and the divertor plates for the heat loads discussed above does not present major technical risks. However, the impact of the design on plasma behaviour was not assessed and the impurity behaviour in such a configuration must be studied.

Disruptions

Problems arise if longer burn times than 200-400 days are required or when abnormal operation conditions (disruptions) are taken into account. During a disruption, the thermal energy in the plasma is dumped onto areas in direct contact with the plasma and $\sim$50% of the magnetic energy are deposited on the wall protection. The time-scales will be many x100 μs for the divertor plates and many x10 ms for the inner wall. The energies considered are $\sim$1.8GJ for the inner wall and $\sim$1GJ for the divertor. The deposition area may be rather localized. It is clear that for the times mentioned, there exists no material which can take the resulting loads without melting or in the case of graphite, without vapourizing. At the inner wall; 30kg of graphite will evaporate for each disruption. In the case that one ionization step is taken into account during the interaction (plasma shielding), the amount of material lost would be only $\sim$10kg. The respective value for the divertor would be about 7kg. If such an interaction was not localized, several thousand disruptions could be sustained.

It will never be possible to avoid disruptions completely. Therefore, each machine must have the capability of withstanding hundreds of disruptions without being destroyed. Unfortunately, the interaction with wall components is not yet fully understood and effort is required to study in divertor machines the pattern of power deposition and deposition times during disruptions, to model the interaction of materials with extremely high heat loads and, if possible, outside a tokamak, to verify the models by experiments. In addition, disruptions, as the source of this problem, should be controlled, and, therefore, schemes should be devised and tested to minimize their occurrence.

Exhaust and Refuelling

Pumping of the proposed machine can be achieved with existing techniques. Here again, the physical size is beneficial because it allows installation of pumping ducts of sufficient cross-section. Each machine sector has two pumping ducts with an area of $\sim 0.8m^2$ each. The ducts with a length of 10m end outside the biological shielding. The conductance for deuterium is $\sim 3x10^4 ls^{-1}$. A cryopump of $\sim 1.5x10^5 ls^{-1}$ pumping speed would be installed at the end of each duct. The installed pumping speed at the divertor would therefore be $\sim 10^6 ls^{-1}$ for deuterium and by using charcoal at 4K a similar pumping speed for helium could be achieved. The vacuum time constant would be about 3s. If during a detailed study, these values turned out to be too low, thermomechanical pumps [36] could be installed which would increase the pumping speeds by x100.

Without a detailed study of particle transport for hydrogen and helium in the plasma, reliable predictions cannot be made for recycling fluxes and the pumped fraction. Models for α-particle transport cannot yet be verified experimentally. JET might contribute information as soon as a sufficient number of α-particles are produced.

Refuelling can either be implemented by gas or by pellet fuelling or a combination of both. From observations in JET [22], pellet fuelling can produce densities far above the Murakami limit, reduce the impurity concentration and increase energy confinement times. To obtain these favourable features, it seems necessary to inject the pellets at such speeds that they penetrate at least well beyond the $q = 2$ surface before ablation. For the proposed machine, speeds $\geq 10 kms^{-1}$ would be required with the necessary development of pellet injection.

Summary and Conclusions

Only low-Z materials can be employed inside the JET vessel. Carbon in a variety of forms is extensively used for plasma facing components either as fine grain graphite, carbon fibre reinforced graphite or as a carbon film. Operational experience is generally good. However, the following problem-areas have been identified:

- Wall conditioning, isotope and density control require wall temperatures above 250°C;
- Graphite seems to be the main source of oxygen;
- Frequent conditioning with helium is required;
- Carbon films on exposed areas have a limited lifetime;
- Carbon retains hydrogen introduced into the vessel in the form of hydrogen-rich carbon films deposited during discharges, which leads to difficulties in density control, recovery from disruptions and tritium inventory;
- graphite is the main impurity at typically 3%, and leads to increased Z_{eff} values and plasma dilution;
- Fine grain graphite, if overloaded thermally, is prone to fail in fracture. Carbon fibre reinforced graphite is extremely forgiving when thermally overloaded and should replace the fine grain graphite for plasma facing component, even if the thermal erosion may be higher.

Extrapolation of results from JET show that high plasma currents are needed for a reactor which would be of such a size that confinement will no longer be the dominant issue. The main problems would be impurity production and their control, fuelling and exhaust and resistance against disruptions.

Technical considerations such as stress levels in coils, wall loadings and economy suggest the design of a "Thermonuclear Furnace" with several GW thermal output. Its aims should be to study an ignited plasma with semi-continuous operation, to test wall technology and breeding blankets and to show the prospect of fusion as a reliable energy source. The most promising solution for the divertor is based on the movement of the X-point relative to the divertor plates, using graphite fibre material and to block the chemical activity by a beryllium layer. However, studies are required to validate this scheme.

The technology to build such a machine exists but there are still uncertainties due to unavailable data. JET can contribute in filling this gap by providing information on:

- Particle fluxes and energies in the divertor region; - α-particle transport;
- Stabilisation of disruptions;
- Behaviour of beryllium as an alternative low-Z material;
- Deep fuelling of the plasma with high speed pellets;
- Tritium retention.

Additional effort is required in the areas of:

- Erosion-deposition processes and their application to a reactor
- Interaction of solids with plasmas at GJ energies during periods of hundreds of microseconds;
- Properties of liquid/gaseous/plasma targets for the divertor;
- Plasma exhaust and refuelling.

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References

[1] G. Duesing and K.J. Dietz, J. Vac. Sci. Technol. A3 (1985) 1151.

[2] M. Pick et al., Proc. 12th SOFE, Monterey, USA, 1987.

[3] K. Sonnenberg, J. Nucl. Mat. this conference.

[4] P.H. Rebut et al., JET-R (85) 03.

[5] R. Haange, Proc. 9th Symp. Eng. Problems of Fusion Research IEEE, 1981, p.1352

[6] TFR Group, Proc. Int. Conf. Plasma Wall Interaction, Jülich 1976 (Pergamon Press 1977), p.3.

[7] W. Delle et al., Jül-Spez-401, May 1987.

[8] J. Dietz, J. Nucl. Mat., 154-156 (1987) in print.

[9] J. Dietz et al., J. Nucl. Mat., 128+129 (1984) 10.

[10] M. Huguet et al., Fusion Technology 11 (1987) 43.

[11] M. Huguet et al., Proc. 11th Symp. Fusion Eng., Austin, 1986 (IEE publication, 1986) p.1238.

[12] G. Celentano et al., Proc. 14th SOFT, Avignon, 1986 (Pergamon Press, 1986) p.581.

[13] K.J. Dietz et al., in: Tokamak Start-up, Ed. Heinz Knoepfel (Plenum Press, 1986) p.317.

[14] K. Sonnenberg et al., Proc. 14th SOFT, Avignon, 1986 (Pergamon Press, 1986) p.273.

[15] E. Usselmann et al., Proc. 13th Symp. Fusion Technol. (1984) 150.

[16] P. Coad et al., J. Nucl. Mat. 145-147 (1987) 747.

[17] J. Winter, J. Vac. Sci. Techn. A5(4) (1987) 2286.

[18] F. Waelbroeck and J. Winter, private communication.

[19] M. Stamp et al., Proc. 12th Europ. Conf. Contr. Fusion and Plasma Physics, Budapest, 1985 (European Physical Society, 1085) p.539.

[20] K. Behringer, JET Progress Report 1986, (EUR-JET-PR4, March 1987), p.41.

[21] L. de Kock et al., J. Nucl. Mat. 145-147 (1987).

[22] A. Gondhalekar et al., 11th Int. Conf. Plasma Phys. Contr. Nucl. Fusion Research, Kyoto, 1986, paper IAEA-CN-47/I-I-6.

[23] R.J. Bickerton, Plasma Physics and Contr. Fusion, Vol. 29-10A, p.1219 (1987)

[24] J. Winter, J. Vac. Sci. Techn. A (1987), in press.

[25] F. Waelbroeck et al., J. Nucl. Mat. 85+86 (1979) 345.

[26] M.A. Pick and K. Sonnenberg, J. Nucl. Mat. 131 (1985) 208.

[27] W.L. Hsu and R.A. Causey, J. Vac. Sci. Tech. A (1987), in press.

[28] J.G. Cordey et al., Proc. 11th IAEA Conference, Kyoto, Japan, 1986, Vol. I, pp. 99-110

[29] P.H. Rebut et al., Proc. 11th IAEA Conference, Kyoto, Japan, 1986, Vol. 2, pp. 187-196

[30] P.P. Lallia et al., 2nd Eur. Fusion Theory Meeting, Varenna (1987) and JET Preprint JET-P(88)05

[31] F. Troyon, Phil. Trans. R. Soc. London A322, 163-171 (1987)

[32] B.T. Kelly, Physics of Graphite, Applied Science Publishers, London and New Jersey, 1981

[33] T.G. Nich, etal., Scripta Metallurgica 20 (1986) 87

[34] P. Mioduszewski, J. Nucl. Mat. 145-147 (1987) 210

[35] J. Roth, J. Nucl. Mat. 145-147 (1987) 87

[36] H. Hemmerich, JET-P (87) 03

TABLE 1

Main Parameters of the Furnace

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)	600-2000
Toroidal field (plasma centre)	(T)	4.5
Plasma current	(MA)	30
Voltseconds	(Vs)	425
Additional heating	(MW)	50
Fusion power	(MW)	500-4000

TABLE 2

Loads on Wall Components

Surfaces		
Wall	(m^2)	1660
Divertor	(m^2)	190
Plasma	(m^3)	1310
Volumes		
Vessel	(m^2)	3260
Plasma	(m^2)	2410
Wall loads		
Nuclear heating (graphite)	(MWm^{-3})	12
Radiation	(MWm^{-2})	0.15
Divertor loads		
Radiation	(MWm^{-2})	0.3
Conduction (averaged)	(MWm^{-2})	0.89
Nuclear heating (graphite)	(MWm^{-3})	12

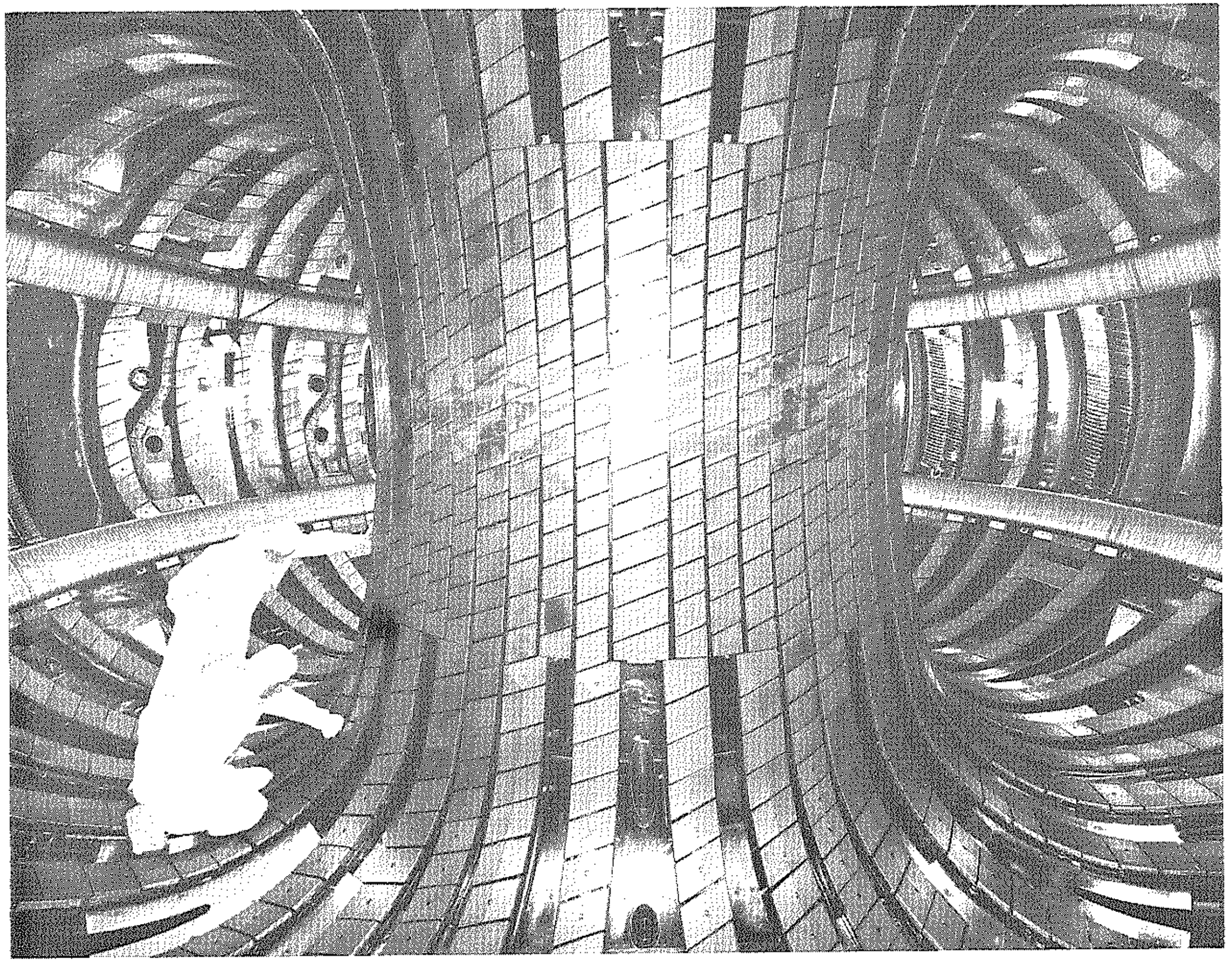

Fig. 1 The inside of the JET vacuum vessel in its present state showing belt limiter, octant-joint and bellows protection, inner wall protection and RF-antennae

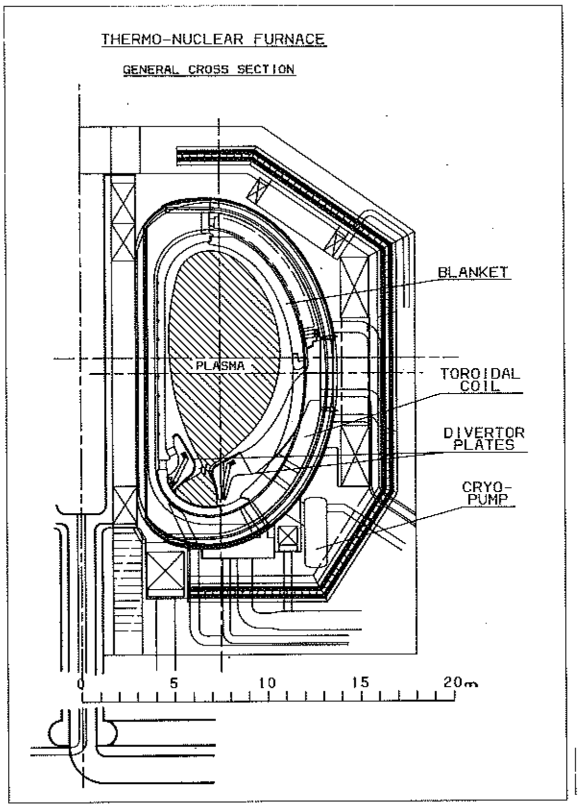

Fig. 2 Cross section of the thermonuclear furnace

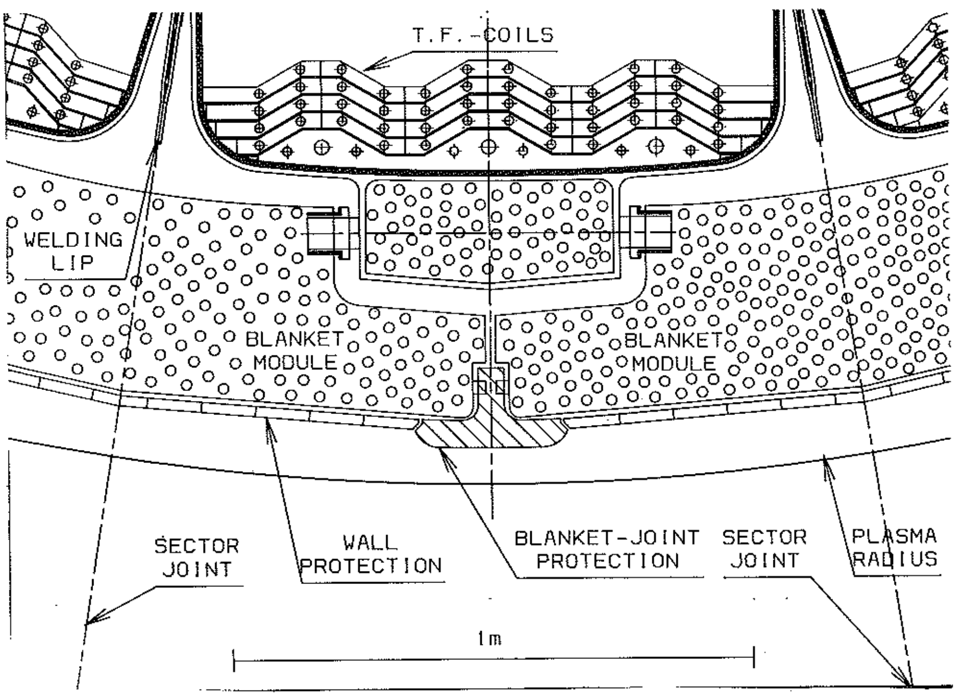

Fig. 3 Inner wall protection, plan view

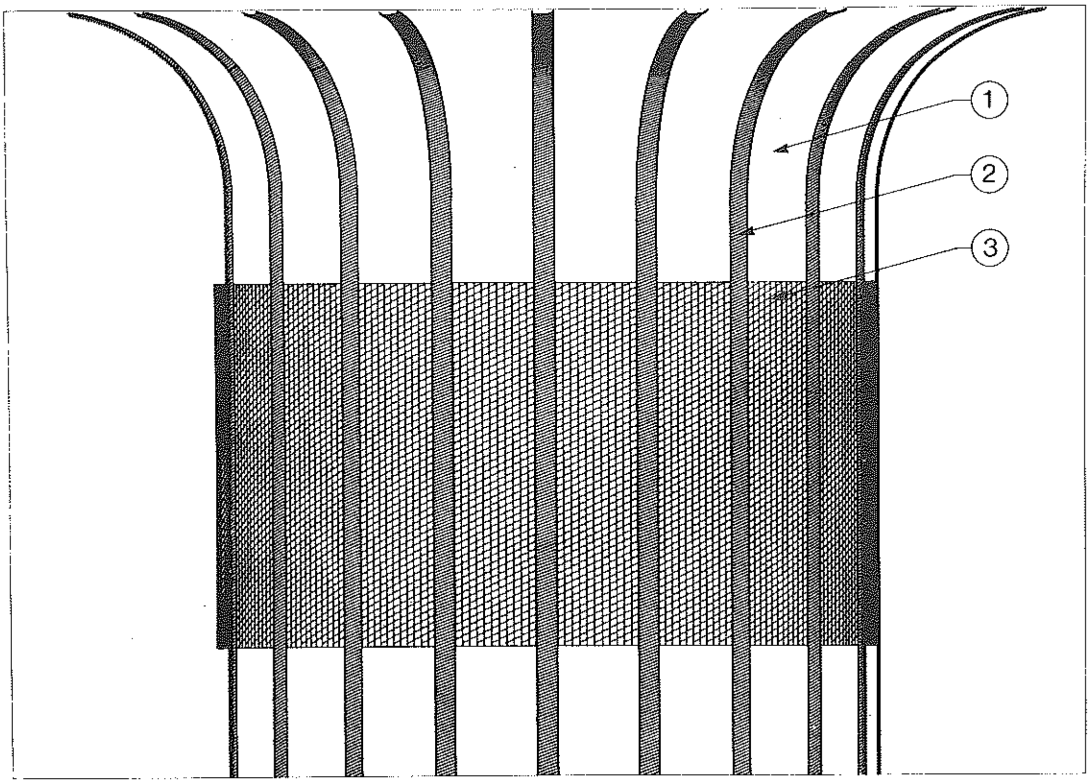

Fig. 4 Inner wall protection, three dimensional view.
Carbonized area[1], poloidal graphite rings[2], wall tiles[3]

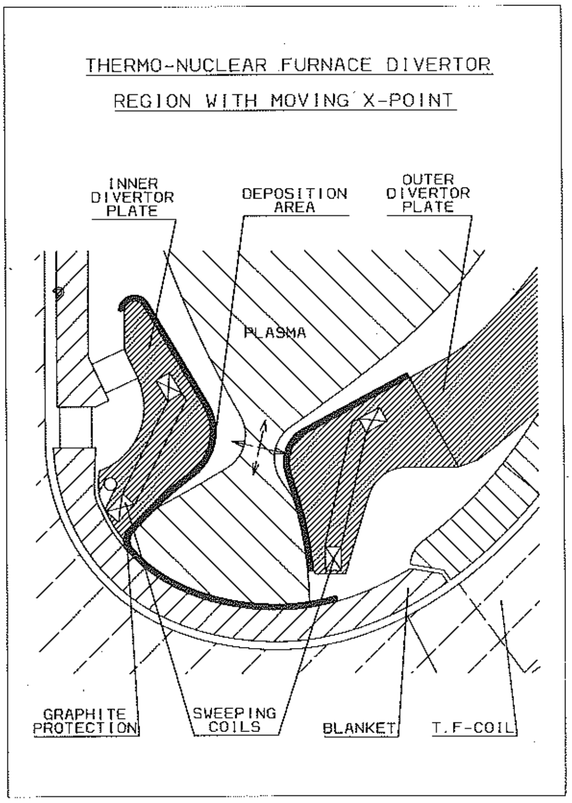

Fig. 5 Cross section through divertor area