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Technical Aspects of Impurity Control at JET: Status and Future Plans

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** See Appendix 1*

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TECHNICAL ASPECTS OF IMPURITY CONTROL AT JET: STATUS AND FUTURE PLANS

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ABSTRACT

The salient feature of 1989-1990 operation at JET has been the use of beryllium as a first wall material. Technical and safety aspects of JET operation with beryllium are described.

The use of beryllium has substantially improved the plasma purity and as a consequence a record fusion product ($n_D T_i \tau_E$) of $9 \times 10^{20} \text{ m}^{-3} \cdot \text{keV} \cdot \text{s}$ has been achieved. Impurity influxes however prevent the achievement of higher plasma parameters and reaching steady state conditions.

A new divertor configuration has been proposed for JET with a view to study impurity control, fuelling and exhaust, in conditions relevant for the next generation of machines. The latest design features a multi-coil configuration which gives substantive operational flexibility. A technical description of the major components of the divertor including the internal coils, the target plates the pumping and fuelling systems is given.

I INTRODUCTION

The Joint European Torus (JET) is the central and largest project of the fusion programme of the European Community. This programme is coordinated by the European Atomic Energy Community (EURATOM).

The JET objective as defined in 1973 is to obtain and study a plasma in conditions and with dimensions approaching those required in a thermo-nuclear reactor [1]. This objective implies that deuterium-tritium (D-T) mixtures will be used in JET to study the production and confinement of alpha particles generated by fusion reactions and the consequent heating of the plasma by those alpha-particles. Technical aspects of and progress towards D-T operation at JET have already been reported [2, 3, 4].

A major area of research which has been identified since the beginning of the project is the study of plasma-wall interactions, the control of the plasma purity and the related requirements for plasma fuelling and exhaust.

Since the start of JET operation in 1983, a large and continuous effort has been made to minimise the detrimental effect of plasma impurities. Initially, the first wall facing the hot plasma was metallic (inconel). From 1984 to 1988, carbon was used as a first wall material. Graphite tiles were used to clad the walls and glow discharge techniques allowed to transform the remaining metallic surfaces into metal carbides or to coat them with hydrogenated carbon films. Graphite tiles were also used for limiters and X-point target plates[5].

Operation with a carbon first wall has been successful. The plasma current was raised to the record value of 7 MA. At lower plasma current (3 MA), quasi steady state conditions were achieved with T_i and T_e above 5 keV for 20 seconds. The plasma heating systems also performed up to their design specifications and were able to provide 35 MW of power to the plasma in combined operation. Physics results established JET at the forefront of fusion research and a fusion parameter ($n_D T_i \tau_E$) of $2.5 \times 10^{20} \text{ m}^{-3} \cdot \text{keV} \cdot \text{s}$ was achieved in H-mode operation[6].

With an all carbon first wall, the attainment of higher plasma parameters is however limited by impurities, mostly carbon and oxygen, coming from the walls. This effect can be characterised by the dilution factor n_D/n_e (n_D : deuterium density; n_e : electron density) which cannot be maintained much above 0.6 even with moderate heating power in the range of 5 to 10 MW. Carbon impurities from the graphite tiles simultaneously dilute the plasma, increase the radiated power and decrease neutral beam penetration.

Although plasma dilution is observed during long pulses at low plasma input power, the carbon influx is very strong at higher power when the surface temperature of the carbon tiles reaches temperatures of 1200-1300°C. In the latter case the carbon influx is thought to be due essentially to radiation enhanced sublimation and self sputtering. Self sputtering can increase the carbon influx in an avalanche like process when the self sputtering coefficient increases with temperature and reaches a critical value^[7]. At JET this effect called the "carbon catastrophe" results in a sharp decay of all plasma parameters and a fall of fusion performance measured by the neutron yield from deuterium-deuterium (D-D) reactions.

For these reasons, beryllium was proposed as a first wall material as it was expected that it could provide superior performance because of its lower atomic number and its gettering properties for oxygen. A full comparison between carbon and beryllium as first wall materials and the rationale in favour of beryllium has been given in^[8]. Also preparatory experiments conducted in the ISX.B^[9] and UNITOR^[10] tokamaks indicated that beryllium was a viable limiter material. Beryllium was first introduced into JET in 1989 and has been used with great success during the 1989 and 1990 experimental campaigns.

II STATUS OF FIRST WALL DURING THE 1989-1990 EXPERIMENTAL CAMPAIGN

Inboard Wall

The vessel configuration is shown on figure 1. The inboard wall is covered with graphite protection tiles. In the vicinity of the equatorial plane where the heat load is highest during normal operation and plasma disruptions, the tiles are fibre reinforced graphite and have a very much higher resistance to thermal shock than normal fine grain graphite tiles. The tiles have been aligned and can act as an inner bumper limiter with a power handling capability of 400 MJ. It should be remarked however that useful plasma discharges with an acceptable plasma dilution can only be sustained at much lower energy levels. Observations of the tiles showed that even though most tiles remained at temperatures below 600°C, some were heated sometimes up to 3000°C. Carbon sublimation often leads to plasma disruptions. Plasma energy inputs of about 20 MJ were able to produce this effect.

Figure 1 shows a discontinuity in the wall protection above and below the equatorial plane. This is due to the fact that internal saddle coils for the stabilisation of plasma disruptions have not been installed yet^[5]. Installation of these coils is planned for 1992.

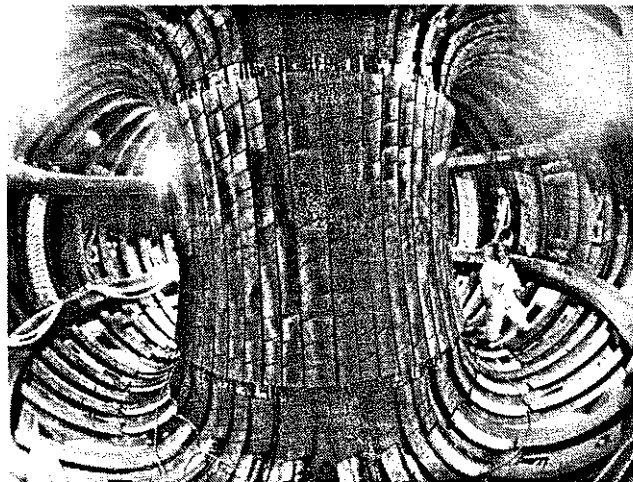


Figure 1: Present status of the JET vessel

X-point tiles

When the machine is operated in the X-point configuration, wall protections are also required where the magnetic separatrix intersects the vessel wall. It should be noted that JET can operate in a single-null configuration with the X-point at top or bottom, and also in a double-null configuration. The wall protections near the top X-point are fibre reinforced graphite tiles which have been carefully shaped and aligned to minimise local hot spots on the tiles. In particular the leading edge of each tile is shadowed by the adjacent tile. In addition, sweeping of the X-point position has been carried out in order to increase the effective strike area and reduce the surface temperature of the tiles. Sweeping in the radial direction (frequency ≈ 1 Hz, amplitude ≈ 5 cm) was clearly effective at delaying the onset of carbon influx into the plasma. In the double-null configuration, vertical sweeping produced the same effect by alternatively loading the top and bottom tiles. A further improvement was achieved by injecting gas in the X-point region. This delayed the carbon influx and allowed the sustenance of 5.3 seconds long H-mode discharges. With these precautions it has been possible to achieve an energy deposition of 30-40 MJ per pulse on these tiles before the onset of the "carbon catastrophe".

The wall protections near the bottom X-point consisted also of graphite tiles during the 1989 campaign. In 1990, these tiles have been replaced by beryllium tiles. The beryllium tiles which have been used for this purpose were designed to be protection tiles for the Ion Cyclotron Resonance Heating (ICRH) antennae and are therefore not an optimum design for an X-point target plate. Because of the curvature of the tile profiles and shadowing effects, the useful total strike length in the toroidal direction is only about 1.6m, whereas the toroidal circumference is 17m.

The decision to use these tiles as X-point protections was taken in order to gain experience with beryllium target plates prior to the installation of much larger and specially designed beryllium target plates by the end of 1990 (see section V). The beryllium X-point protection tiles can be seen in figure 1. There are a total of 32 radial rows of tiles distributed uniformly along the torus. There is so far only limited operational experience with these tiles. Some surface melting has been observed with an energy deposition of about 20 MJ per pulse. This surface melting produces surface irregularities but did not seem to affect operation. Melting was observed only at the outboard (low field) side of the X-point indicating a strong asymmetry in power flow between the inboard and outboard sides. At these energy levels a beryllium influx was observed within a few hundred milliseconds after the start of X-point operation.

Limiters

When the machine is operated in the limiter configuration, the plasma is leaning against the belt limiter^[11]. The belt limiter consists of two toroidal rings above and below the equatorial plane of the machine, at the outboard wall. It is designed so that the tiles are radiatively cooled by a radiator structure which can be water cooled. The tiles can be easily exchanged. In 1989 and 1990, beryllium tiles have been used rather than graphite tiles. The beryllium employed is S-65B (Brush Wellman) cold pressed and sintered. The front face of the beryllium tiles is castellated in order to avoid the deep propagation of surface cracks. Experiments carried out at Sandia Laboratory USA, have shown that thermally induced surface cracks do not propagate when castellations are provided.

Although the belt limiter was designed with a power handling capability of 40 MW during 10 seconds, its use has been limited by plasma behaviour^[12]. The design value of the peak heat flux was about 5MW/m² but the actual peak heat flux in operation has reached much higher local values due to uneven loading between the top and bottom belts, scrape off thicknesses smaller than expected and edge loading of the tiles.

Beryllium influxes from localised hot spots at the limiter surface terminate low density high performance pulses. The reason for these hot spots is not clear even after a careful examination of the tiles during the 1989-1990 shutdown. The misalignments between adjacent tiles can be of about 0.3mm and this could perhaps explain hot spots at the exposed leading edge of some tiles. Local melting at the leading edge can produce irregularities or droplets on the tile surface, thus increasing the flux deposition and the damage can propagate from the edge across the tile

surface. This scenario is supported by observation of melting at tile edges and at the tessellation cuts. About 10-15% of the tiles showed evidence of edge heating. Surface melting which did not seem to originate from the edge was also seen and was often associated with surface cracks. Surface cracks can only be initiated by large heat loads and it has been suggested they could be the result of disruptions. About 5% of the plasma facing area showed this kind of damage.

Melting of beryllium tiles was only superficial and did not affect the tile integrity or operation although surface unevenness of about ± 1 mm was observed in some areas. Tile melting was not distributed uniformly along the toroidal direction, instead there was a clear correlation with the ripple of the toroidal field: most melt marks were found between toroidal field coils. This observation demonstrates that an accurate alignment of tiles is essential. The radial variation of plasma position due to field ripple is only 1.5 mm at the face of the limiter.

The energy deposition on the belt limiter has generally been kept well below the design value of 400 MJ. With a careful selection of the plasma shape so as to maximise the plasma foot print on the limiter, it has been possible to apply 90 MJ of heating energy with an acceptable beryllium contamination. Moreover, at high plasma densities and by tailoring the gas feed during the discharge, it has been possible to feed into the plasma 30 MW of power for 6 seconds while keeping Z_{eff} at about 1.5 (dilution 0.83). This should be compared with the graphite limiter where a maximum of 50 MJ could be applied without carbon catastrophe but with a plasma dilution of about 0.5.

Evaporators

The evaporators are essential elements of beryllium operation^[13]. Each evaporator consists of a bell shaped beryllium head (3kg) which is internally heated by a graphite resistor. During plasma operation the head is retracted inside a port but for evaporation it is pushed inside the vessel and heated up to a temperature close to 1200°C.

Initial operation of the evaporator was satisfactory but there was a sharp decrease in the evaporation rate only a few hours after the start of operation. This was attributed to a beryllium and beryllium oxide dust layer which was found on the head. The dust may be due to oxidation by residual gases in the vacuum vessel. Also the impurities present in the sintered material tend to accumulate on the surface when beryllium evaporates. The dust layer acts as a cold shroud which prevents evaporation. More recently, beryllium heads made of cast material (rather

than sintered), which is purer, have been used. There is still a small reduction of the evaporation rate after a few hours but much less than with sintered material. The beryllium heads have now all been replaced with cast material.

There are four beryllium evaporators equally spaced along the torus. The beryllium coat on the vessel walls and graphite tiles is of course not uniform but an average thickness of 20nm is achieved in one hour of evaporation. By the end of 1989 the average coverage of the vessel was about 0.8µm.

The beryllium coat on the graphite X-point tiles erodes within 1 to 5 discharges and then an equilibrium tends to establish between erosion and redeposition. The impurity reduction in the plasma remains however unchanged over tens of discharges. Evaporation sessions, typically 2 to 5 hours long have been carried out routinely, once or twice a week in 1989 and 1990.

Beryllium evaporation has also been found to be an extremely efficient and quick method to recondition the vessel for plasma operation after a vessel opening. The residual gas analyser shows a dramatic reduction of gaseous contaminants in the vessel after a beryllium evaporation.

III TECHNICAL AND SAFETY ASPECTS OF BERYLLIUM HANDLING

Since the introduction of beryllium in the JET vessel in July 1989, there have been several interventions requiring personnel access inside the beryllium contaminated vessel, and one major shutdown during which one eighth (octant) of the machine was dismantled for the replacement of a faulty toroidal field coil^[14]. This shutdown involved work in two shifts during 6 months and a total of 8,000 man-hours have been spent inside the vessel. There has been so far no safety incident involving beryllium on the JET site.

There is considerable experience available at JET on how to tackle work inside the contaminated vessel, and how to handle contaminated components, tools and protective clothing outside the vessel. It should be noted that the techniques developed for beryllium handling may also become useful in the future to protect workers against, and avoid the spread of contamination by activated dust from the vessel.

The central piece of equipment for work inside the vacuum vessel is the Torus Access Cabin (TAC) which includes all the facilities for the safe access of personnel inside the vacuum vessel^[15]. The TAC can support 5 men working with full protective suits with an external air

supply. Air locks are also provided for the transfer of small tools and components. During the 1989-1990 shutdown, the TAC operation has been generally satisfactory but many improvements were found necessary and implemented to the breathing air supply, ventilation plant and water wash system. Also extra cabins were docked onto the TAC to accommodate the flow of personnel, provide additional space for change facilities and additional storage space for components protection tiles taken out of the vacuum vessel.

Plasma operation produces significant quantities of fine beryllium dust which becomes easily airborne and would in the absence of respiratory protection be inhaled by workers inside the vacuum vessel. At the beginning of the October 1989 shutdown, the vessel was water washed to reduce the beryllium airborne and surface contamination. Following the wash, the airborne contamination was generally well below the statutory level ($2\mu\text{g}/\text{m}^3$) above which respiratory protection is mandatory. It was found, however, that sudden large rises in airborne contamination would occur when components inside the vessel were disturbed or dismantled. This was attributed to pockets of beryllium dust which may have escaped the water wash, or possibly dust which may have been collected by water during the wash and may have accumulated in certain areas. For this reason respiratory protection was worn at all times inside the vessel. Depending on potential contamination, full suits with external air supply or face masks with filters have been used. For grinding work, protective helmets with a powered filtered air supply have been worn. It was found that wearing suits did not affect the efficiency and speed of work of the personnel.

For work requiring an opening of the vacuum vessel but no access inside, bagging techniques have been developed to avoid the spread of any beryllium dust. A dedicated workshop with specialist equipment for welding PVC has been set up to manufacture tailor made isolators able to bag the components, mostly diagnostics, which have to be disconnected from vessel ports.

Large pieces of equipment cannot be introduced into the vacuum vessel through the TAC. Access is provided at another horizontal port, where the remote handling Articulated Boom is routinely used to introduce and accurately position heavy components inside the vessel^[4]. The Articulated Boom is itself enclosed in an air tight PVC tent attached onto the vessel port. Components are transferred in and out of the tent through an air lock and a slight under pressure is maintained inside to ensure that contamination is well contained.

Beryllium related work on contaminated tools or components is carried out in a beryllium facility which has been erected inside the JET Assembly Hall. The facility has a working area of 72m² and includes an access system, a component transfer air lock, a filtered ventilation plant and a waste water collection system.

Suits used for in-vessel work have to be decontaminated, cleaned and checked for damage before re-use. This work is done in a dedicated suit cleaning facility. During the shutdown 90 suits/week were used from a complement of 300 suits. After each working session the suits were double-bagged and transferred to the cleaning facility. Suits were cleaned with water and detergent inside and outside and then checked for residual beryllium contamination. Following an acceptable analysis result, the suits were then transferred to another area for repair and final inspection before re-use.

A beryllium analysis laboratory has been set up for the analysis of beryllium samples. Airborne contamination or surface contamination must be collected on filter paper which is then analysed by the Atomic Absorption Spectrometric Technique. During the shutdown, more than a hundred samples were analysed every day. The samples originated from in-vessel work where each man carries a personal air sampler, from fixed air monitors in various areas, from smear samples on protective suits after cleaning, and other smear samples required to clear decontaminated tools or components or entire work areas.

All solid and liquid beryllium contaminated wastes are carefully monitored and disposed of in accordance to strict procedures complying with regulations.

IV SUMMARY OF RESULTS ACHIEVED WITH A BERYLLIUM FIRST WALL

The chief effect of beryllium as a first wall material has been to reduce the plasma dilution. This effect is illustrated in Table 1 which gives the average impurity concentrations and dilution factor for various operating phases^[16-17].

The gettering properties of beryllium removes oxygen from the plasma. The elimination of carbon sputtering by oxygen also results in a strong reduction of the carbon concentration.

Another beneficial effect of beryllium is that it pumps deuterium very strongly during the discharge. A recycling coefficient as low as 0.9 has been observed with the beryllium belt limiter, whereas values of 0.99 were obtained with an

unconditioned graphite limiter. At the same time, beryllium retains after the discharge only a small proportion, about 10 to 20% (60% for graphite) of the deuterium injected in the vessel during the pulse.

Table 1. Average impurity concentrations (%) and dilution factor for the various operating phases

	C Phase	Be/C Phase	Be Phase	
			Limiter	X-point
Oxygen	1	0.05	0.05	0.05
Carbon	5	3	0.5	1.5
Beryllium	-	1	3	1
$n_d(o)/n_e(o)$	0.6	0.8	0.85	0.9

C phase = carbon first wall
 Be/C phase = beryllium evaporation only
 Be phase limiter = beryllium evaporation and solid beryllium tiles on the limiter
 Be phase X-point = beryllium evaporation and graphite X-point tiles

It should be stressed that in contrast to beryllium, graphite behaves as a reservoir of hydrogenic atoms which are released by plasma contact. For this reason, graphite walls become an uncontrollable source of particles during the discharge. Graphite walls can only be depleted from the stored atoms by frequent and time consuming conditioning sessions with helium (glow discharge or tokamak discharges).

The properties of beryllium have a great impact on operation since density control is much easier, operation at low density becomes possible (hot-ion mode), and the wall is no longer deconditioned by disruptions.

From the improved plasma purity, an improvement of nearly all plasma parameters ensues. Details are given in^[16-17] and the results can be summarised by the record values achieved in pulse No. 20981.

Table 2. Pulse No. 20981

(n_D , n_e and T_i are values at the plasma centre)

X-point double null configuration. H-mode

$I_p = 4\text{MA}$ PNB = 16 MW

$n_D = 3.7 \times 10^{19} \text{m}^{-3}$

$T_i = 22 \text{keV}$ $n_D \cdot T_i \cdot \tau_E = 9 \times 10^{20} \text{m}^{-3} \cdot \text{keV} \cdot \text{s}$

$\tau_E = 1.1 \text{sec}$

$n_D/n_e = 0.92$

D-D fusion power: $P_{D-D} = 40 \text{kW}$

Computed equivalent D-T fusion power

$P_{DT} = 13 \text{MW}^*$ (for 0.1 second)

*Total power (thermal + beam/plasma) for injection of 140 kV D beams in a tritium plasma.

V THE PUMPED DIVERTOR: AN IMPURITY CONTROL CONCEPT RELEVANT FOR THE NEXT STEP

Although impressive, the results described in section IV were achieved in transient conditions. The high quality period of a plasma pulse lasts only for a second or less, and the D-D neutron count, which is a true measurement of fusion performance decays rapidly as impurities penetrate the plasma.

There is scope to improve the situation and lengthen the duration of the useful part of a pulse. X-point target plates carrying beryllium tiles will be installed in JET at the lower X-point position by the end of 1990. These dump plates have a much larger surface than the existing X-point tiles and will follow very closely the theoretical shape of the torus in both the toroidal and poloidal directions. Sweeping of the X-point position will also be performed as already explained in section II. This will result in a very substantial increase of the strike area and it is expected that useful X-point discharges will be sustained at higher power and for longer duration.

This approach may improve JET's performance but is however not relevant for the machines of the next generation which will operate for long pulses, a few 100 to a few 1000 seconds, and therefore require steady state conditions for the plasma.

It is also necessary to obtain more data on the operation of the divertor systems required by the next step devices. The problems of controlling the back-flow of impurities from the divertor target to the plasma, and of the target plate erosion have never been studied with plasma parameters and pulse durations which are of some relevance for next step machines.

Due to its size, plasma performance and long pulse capability, JET is in a unique position to carry out such studies.

The objectives of the new JET impurity control programme are "to demonstrate effective methods of impurity control in operating conditions close to those of the next step tokamak; that is a stationary plasma of thermonuclear grade in an axi-symmetric pumped divertor configuration".

VI TECHNICAL DESCRIPTION OF THE JET PUMPED DIVERTOR

Basic principles and main components

The key concepts of the pumped divertor have already been explained^[7,18].

The impurities generated at the divertor target plates should be confined there by a flow of plasma particles directed towards the target plates. This confinement relies on friction forces between impurities and plasma particles, and to be effective requires a strong plasma flow which should be enhanced by recycling of neutrals from the target plates. The connection length along magnetic field lines and between the target plate and the X-point region should also be long enough, that is of the order of 3 to 10 metres^[7].

A low temperature, high density target plasma should form in front of the target plate. This target plasma should:

- Radiate some of the incident power and somewhat decrease the peak heat load on the target plates.
- Screen the target plates from the plasma particles and reduce the impurity production.
- Screen the plasma from the target plate impurities.

The main components of the pumped divertor are:

- The divertor coils which produce the required magnetic configuration.
- The target plates.
- The cryopump which will help in controlling the main plasma density.

The magnetic configuration

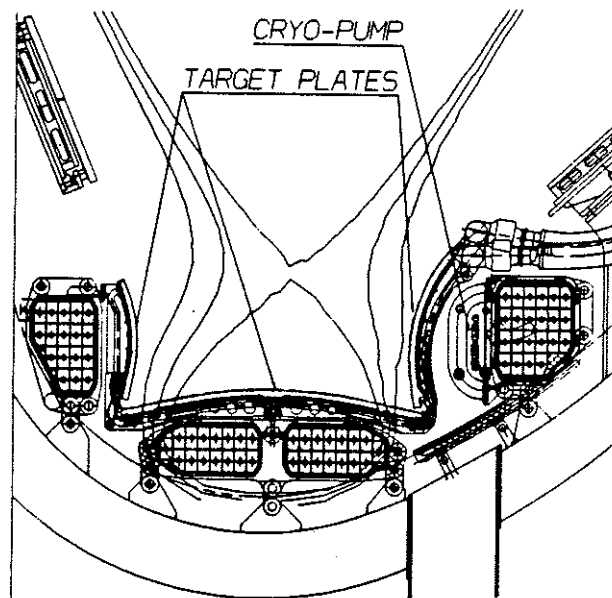


Figure 2: The JET pumped divertor components

The configuration features four divertor coils which allow many different configurations to be obtained (Figure 2)^[19]. It should be noted that

all four coils carry currents in the same direction. The two bottom central coils produce the X-point and have been made as flat as possible to increase the volume available to the plasma. The two side coils allow a reduction of the poloidal field in the region between the X-point and the target plate, thus changing the pitch of the magnetic field lines and consequently increasing the connection length.

Since all four coils will have independent electrical supplies, a great flexibility can be achieved in the type of magnetic configuration. This is illustrated in Figures 3 and 4 and Table 3 which show two typical cases, the so called "fat" and "slim" plasma configurations.

Table 3: Parameters of X-point plasmas

Configuration	Fat	Slim
Plasma current (MA)	6	5
Plasma volume (m ³)	88	75
Connection length (m)	3.1	8.2
Safety factor q ₉₅	2.2	2.4
Growth rate (s ⁻¹)*	270	800
Sum of divertor coil currents (MA _t)	0.74	1.5

*Growth rate of plasma vertical instabilities in the absence of vertical position feed-back control.

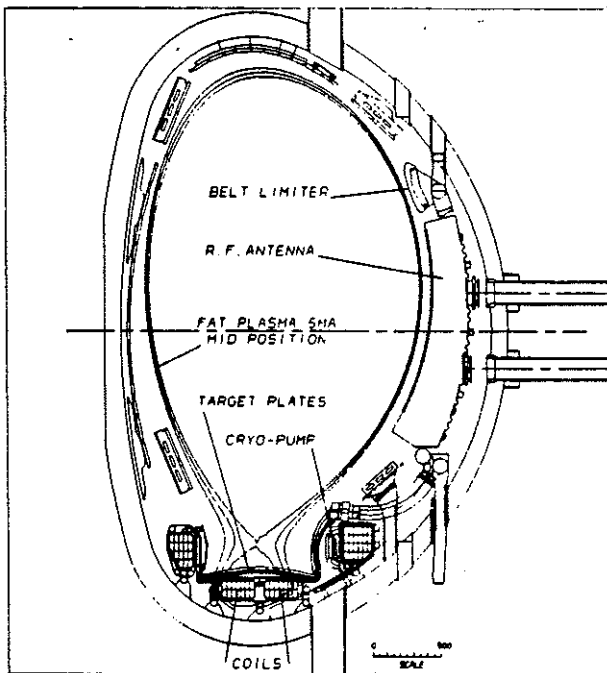


Figure 3: Fat plasma configuration at 5MA

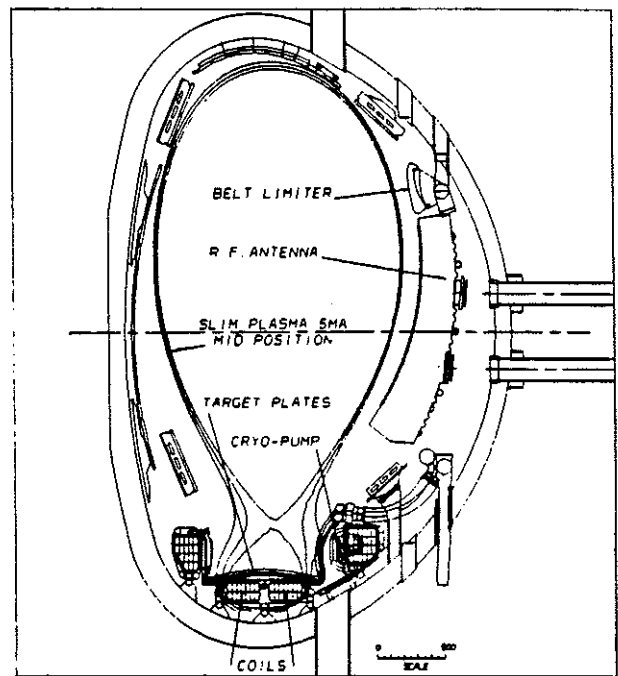


Figure 4: Slim plasma configuration at 5MA

The side coils also allow the connection length to be adjusted independently of the plasma current and separately on the inboard and outboard sides of the X-point.

Sweeping of the strike region on the target plates can be easily achieved by shifting radially the barycentre of the divertor coil currents. A total sweep amplitude of 20 cm is possible without significant changes of the connection lengths inboard or outboard.

Table 3 also shows the growth rate of vertical instabilities which is larger than the present values of 150 to 280 s⁻¹. The stabilisation of divertor plasmas requires a fast vertical position control system which is under development.

The divertor coils

The coils are conventional and use water cooled copper conductors with epoxy glass and kapton insulation. They are enclosed in thin vacuum tight inconel cases and will have to be cooled continuously when the vessel is at its operating temperature of 350°C. The coils will be assembled inside the vessel from preformed one-third turn segments. After assembly of the coil in the case and completion of the final welds on the case, vacuum impregnation with epoxy resin will take place.

The coils include 15 to 21 turns and carry typically 0.6 MA_t. The forces acting on the coils

can be large during vertical instabilities of the plasma when flux variations increase the coil currents. The vertical force can reach 400 tonnes on the outermost coil and the total net vertical force transmitted to the vessel is 900 tonnes. These forces are restrained by hinged supports which allow differential expansion between the vessel and the coils.

The target plates

The target plates are made up of three parts. The horizontal plates intersect the heat flux conducted along field lines while the vertical side plates receive the radiated power from the divertor target plasma. The horizontal and vertical plates are split into 384 segments grouped in 48 modules of 8 segments.

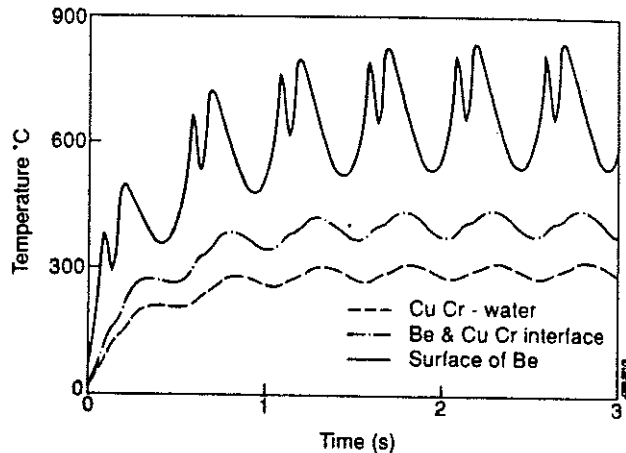
The target plates are designed for a total steady state conducted power of 40 MW which gives a maximum power flux density of 10-12 MW.m⁻² when sweeping of the X-point is taken into account^[20]. The side plates may receive up to 5 MW.m⁻² in case the full power is radiated by the divertor plasma. These power density levels require the use of highly efficient heat sinks. Hypervapotrons have been used extensively for the JET Neutral Beam systems and can cope safely with steady state heat fluxes up to 15 MW.m⁻². Figure 2 shows the layout of the hypervapotrons and the cooling water pipework.

The surfaces facing the plasma will be clad with a 2 mm thick beryllium layer. The choice of beryllium is supported by the results achieved with a beryllium first wall in JET but is not ideal since beryllium impurities in the divertor plasma will only radiate a negligible fraction of the incident power. The choice of any other, higher Z, material would however entail the risk of impurities migrating back to the vacuum vessel and jeopardizing the benefits of a beryllium first wall.

The strength of the bond between the beryllium cladding and the copper-chromium of the hypervapotron is critical. Silver brazing at about 650°C is the preferred method of fabrication but the strength of the joint is strongly reduced when the temperature exceeds 350-400°C. In the longer term, other cladding techniques such as explosion bonding, or plasma spraying which has potential for in-situ repair, may be preferable.

Analytical studies have shown that an average flux density of 12 MW.m⁻² and a sweeping frequency of 2 Hz give acceptable temperature excursions (Figure 5) for both the beryllium surface and the beryllium copper interface. Hot spots, particularly at the edge of each segment must be avoided and, to this end, each segment is tilted so that the edge is shadowed by the adjacent

segment. All segments will have to be aligned with an accuracy of 0.2-0.3 mm. The beryllium layer should be castellated to minimise the plastic strain. This will be achieved by brazing small tiles with dimensions of 6-10 mm.



Temperature evolutions at the beryllium surface and copper-beryllium interface.

Large forces can act on the hypervapotrons due to currents flowing during disruptions. Some eddy current paths have been eliminated by a careful design of the mechanical attachments. However during vertical instabilities it is believed that currents flow in the poloidal direction and these currents can be shared between the vessel and the hypervapotrons. Forces of 2 tonnes per metre could be produced on each of the 384 segments. The hypervapotrons are firmly clamped onto steel beams which are themselves attached to the lower divertor coils.

The cryopump

For pumping, a cryopump has been selected because it has no tritium inventory after regeneration, it is reliable and there is experience and cryogenic systems available at JET.

The pump includes a chevron type liquid nitrogen cooled radiation screen and helium cooled pipes which cryo-condense the incoming gases. This pump is completely surrounded by water cooled structures which intersect the radiated power from the hot vessel and from the divertor plasma^[21].

During plasma pulses the most severe heat load on the helium cooled pipes is expected to come from the neutrons and gamma rays produced during D-T operation. In the case of operation at $Q \approx 1$ with a neutron production rate of 10^{19} s^{-1} , a power of 4.5 kW would be absorbed by the helium and the stainless steel conduits. For

this reason, the helium tubes have thin walls and are slightly corrugated for increased strength. The heat capacity of the helium content (40 litres) is about 50 kJ for a 1°K temperature rise and may therefore limit the pulse duration but only if operation close to $Q = 1$ is achieved.

The cryopump itself has a nominal pumping speed for deuterium at 300°K of $500,000 \text{ l.s}^{-1}$ but what really matters is the probability for neutrals emitted at the target plate to reach the pump. This has been estimated using a code provided by KFA-Jülich^[22]. Neutrals can reach the cryopump through gaps between the hypervaportrons. The pumped fraction depends crucially on the target plasma temperature and varies from 0.5% to 3% for electronic temperatures between 70 and 10 eV.

The pump is split into quadrants in the toroidal direction and there will be two cryo-supplies common for two quadrants. The liquid nitrogen shield consists of blackened copper alloy baffles brazed onto stainless steel tubes for the forced flow of liquid nitrogen. The helium loop is supplied with a forced flow of supercritical helium and includes six thin-walled stainless steel tubes connected in series. The tubes are mounted in brackets which are plasma coated to reduce eddy currents. The brackets are supported from the radiation shield by thin stainless steel wires.

Fuelling

Pellet fuelling is expected to play a key role in controlling the plasma density profile, plasma flow and impurity and alpha-particle concentrations in the plasma. In order to provide operational flexibility and keep all options open a wide range of pellet velocities is foreseen.

Fast pellets at velocities of about 4 km.s^{-1} should be used to fuel the plasma centre and flush impurities and alpha-particles towards the plasma edge. The development of fast guns at JET has now reached an encouraging state and such guns should be available for divertor operation^[23].

Medium velocity (1.5 km.s^{-1}) pellets should fuel intermediate layers inside the plasma and help control the density profile and establish the plasma flow towards the target plate. Discussions are underway between JET and the US-DOE for the supply of adequate guns.

Low velocity (up to 500 m.s^{-1}) pellets should finally fuel the outermost plasma layers and enhance the flow of plasma particles towards the target plates.

In addition to pellet injection, gas injection nozzles are foreseen to increase if required the plasma flow towards the target plates. Gas nozzles are also foreseen inside the triangular

region formed by the magnetic separatrix and the target plate so as to enhance the local flow of particles towards the target.

Ion cyclotron resonance heating

The Ion Cyclotron Resonance Heating (ICRH) system will include 8 antenna modules in 4 groups of two. As shown in figure 3 the antennae feature a deep housing to bring the RF conductor close to the plasma boundary for good power coupling. For the slim plasma configuration shown in figure 4, the antennae will be pushed forward and tilted. The housing supports and the RF conductor are designed to make this displacement a relatively simple operation requiring only a short access inside the vacuum vessel.

Diagnostics

In view of the experimental nature of the pumped divertor experiment, an extensive range of diagnostics are being designed to obtain data on the target plasma temperature and density (Langmuir probes, microwave interferometer, microwave reflectometer, electron cyclotron absorption, Thomson scattering), the impurities in this plasma (XUV/VUV spectroscopy, visible spectroscopy), the radiated power (bolometers), the neutral gas pressure (pressure probes), and the magnetic configuration at the target plates (flux loops and magnetic probes). Thermocouples at the target plates and cooling water pipes will yield data on temperature distributions and total incident power.

VIII CONCLUSIONS

Impurity control at JET has so far concentrated on the development and use of passive elements such as low Z materials for wall protection, limiter tiles and X-point target plates. Graphite and more recently beryllium have been used with great success but impurity influxes prevent the attainment of higher performances and steady state conditions.

A new pumped divertor configuration is proposed for JET which would address the problem of impurity control in operational conditions close to those of the next generation of machines. The construction and operation of the JET pumped divertor requires an extension to the JET Experimental Phase. It is planned that installation would take place in 1992 and operation with the pumped divertor would start in 1993. This extension if approved will allow essential data to be obtained on the operational domain of next step machines, the physics of the divertor and technological aspects such as the choice of materials facing the plasma.

REFERENCES

- 1 The JET Project. Design Proposal. Commission of the European Communities. EUR-JET-R5 (1975).
- 2 M. HUGUET, E. BERTOLINI, Main features implemented in the JET facility for D-T operation. Fusion Technology journal, Vol 10. No.3, p.1398-1403 (1986).
- 3 R. HAANGE et al, General Overview of the Active Gas Handling System at JET. Toronto Conf (1987).
- 4 T. RAIMONDI, The JET experience with remote handling equipment and future prospects, Fusion Eng. and Design **11** (1989) p.197-208 (North Holland, Amsterdam).
- 5 M. PICK et al, Integrated design of the new JET In-vessel components. Proceedings of the 15th Symposium on Fusion Technology vol 1, p.771 Utrecht (The Netherlands). (September 1988).
- 6 R.J. BICKERTON AND THE JET TEAM. Latest JET results and future prospects, 12th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research, Nice (France) (October 1988) 9IAEA-CN 50/A-III.2.
- 7 P.H. REBUT, P.P. LALLIA, B.E. KEEN, Impurities in JET and their control. Proceedings of the 13th Symposium on Fusion Engineering. Vol 1, p.227, Knoxville (October 1989) IEEE catalog 89 CH 2820-9.
- 8 P.H. REBUT et al, Use of beryllium as a first wall material in JET JET Report R(85)03.
- 9 R.C. ISLER et al, Nuclear Fusion **25** (1985) p.1635.
- 10 J. HACKMAN AND J. UHLENBUSCH, J.Nucl. Mater. **1280129** (1984) p.418.
- 11 G. CELENTANO et al, The JET belt limiter, Proceedings of the 14th Symposium on Fusion Technology, Vol 1. p.581, Avignon (France) (September 1986).
- 12 E. DEKSNIS et al, Damage to JET beryllium tiles, Proceedings of the 9th Symposium on Plasma Surface Interactions, Bournemouth (May 1990). To be published.
- 13 K. DIETZ et al, Beryllium in JET, A report on the operational experience, Proceedings of the 13th Symposium on Fusion Engineering, Vol.1, p.512, Knoxville (October 1989) IEEE catalog 89 ch 2820-9.
- 14 M.A. PICK et al, Toroidal field coil exchange and related octant exchange at JET, Proceedings of the 16th Symposium on Fusion Technology, London (September 1990). To be published.
- 15 S.J. BOOTH et al, Beryllium related maintenance on JET proceedings of the 16th Symposium on Fusion Engineering, London, (September 1990). To be published.
- 16 M. KEILHACKER AND THE JET TEAM, Overview of results from the JET Tokamak using a beryllium first wall, Phys. Fluids **B2(6)**, June 1990, p.1291-1299.
- 17 K.J. DIETZ AND THE JET TEAM. Effect of beryllium on plasma performance in JET. Proceeding of the 17th meeting of the European Physics Society. Amsterdam. 1990. To be published.
- 18 M. HUGUET, E. BERTOLINI AND THE JET TEAM, Technical status of JET and future prospects, Proceedings of the 13th Symposium on Fusion Engineering, vol.1, p.491, Knoxville (October 1989). IEEE catalog 89 CH 2820-9
- 19 J.R. LAST et al, The JET divertor magnetic configuration and coil design. Proceedings of the 16th Symposium on Fusion Technology. London (September 1990). To be published.
- 20 E. DEKSNIS et al, Design of high heat flux components for the JET pumped divertor. Proceedings of the 16th Symposium on Fusion Technology, London (September 1990). To be published.
- 21 W. OBERT et al, The JET pumped divertor cryopump, Proceedings of the 16th Symposium on Fusion Technology, London, (September 1990). To be published.
- 22 D. REITER, E. DEKSNIS, EIRENE users manual, JET Report (in press).
- 23 P. KUPSCHUS et al, Upgrading the JET pellet injector with a two-stage light gas gun prototype and future planning. Proceedings of the 13th Symposium on Fusion Engineering, vol.2, p.1293, Knoxville (October 1989) IEEE catalog 89 CH 2820-9.

APPENDIX 1.

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