ENGINEERING EXPERIENCE IN JET OPERATIONS
JET's latest technical and scientific results and engineering development

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ABSTRACT

The inherent flexibility of JET's original concept has permitted several engineering upgrading and modifications, to address a large variety of plasma and fusion physics issues. The most recent major modification has been the installation of an axisymmetric single-null pumped divertor (MarkI), successfully operated in the experimental period 1994-95. Following the divertor optimization programme a new, more closed, divertor configuration has now been installed (MarkII), which has shown a better power handling capability and substantially improved neutral particle retention. A key feature of the new design is the possibility to replace the divertor target plate structure using full remote handling techniques following extended D-T operations. Toroidal asymmetries of vessel forces due to Vertical Displacement Events (VDE) and halo currents were experienced since 1994, leading in some cases to sideways movements of the vessel of several mm. This has required to modify and upgrade the vacuum vessel support system. Gap control of plasma position and shape, and machine protection systems have been developed further, leading to increased experimental availability. Future development foresees the installation of a MarkII Gas Box divertor structure, while studies are underway to increase the toroidal field capability from 3.45T to 4T, and the additional heating power by increasing the NB injector output from 80kV, 60A to 120kV, 60A and by using wide band matching for ICRF.

1. INTRODUCTION

1.1. JET objectives and design concept

The global objective of JET is to obtain and study a plasma in dimensions and physics parameters approaching those needed in a thermonuclear reactor, thus providing necessary physics and engineering data for the design and for the operation of a next step device (such as ITER), eventually using D-T gas mixtures.

Since its first operation in June 1983, the configuration of the JET plasma has been modified to allow new physics issues to be studied and its performance has been steadily increased to well above the original technical specifications, culminating in the first ever D-T experiment (PTE, Preliminary Tritium Experiment) in November 1991, with the delivery of 1.7MW of fusion power and 2MJ of fusion energy.

During the years, the machine went through a number of modifications and upgrading, suggested by physics results and considered necessary for further progress to be achieved. This engineering development could be implemented without replacing any of the major tokamak components (vacuum vessel, toroidal field coils, poloidal field coils, mechanical structure), thanks to the original design philosophy of JET: large plasma volume (~150m^3) and current (4.8MA), relatively modest toroidal magnetic field (3.45T), D-shaped plasma, vacuum vessel and toroidal coils, low aspect ratio, achieved by transferring the inward force of the toroidal magnet through the central solenoid to the inner supporting ring. Consequences of these choices are long operating pulses (up to 20s at full performance), limited stresses in electro-mechanical components and remote handling compatibility. Moreover the main diagnostics are installed beyond the shielding walls and most power and energy is taken directly from the HV grid [1,2].

1.2. Machine development and results

Concerning the upgrading in performance (fusion triple product $n_D\tau_E T_D$ and equivalent energy gain Q_{DT} a dilution factor deuteron density/ electron density n_D/n_e), there were two main limiting factors: the decay of the energy confinement time τ_E with plasma heating (Fig.1) [3,4] and the influx of impurities from the vacuum vessel walls . The first problem was counteracted by increasing the plasma current above the design value (from 4.8MA to 7.0MA) and by modifying the poloidal coil configuration to achieve X-point plasmas up to and above 5MA. This major upgrading of the electro-mechanical system has required JET to modify and enhance the capability of the poloidal coil power supplies, including plasma control and to re-assess the ultimate electro-mechanical capability of the tokamak system, by finite-element modelling of the vacuum vessel, toroidal and poloidal coils and mechanical structure and by load testing of the prototype toroidal coil well above the design values. The second limiting factor was first dealt with by using low 'Z' materials as plasma facing components, i.e. carbon and/or beryllium, moving from inconel vessel walls and 8 (small) discrete graphite limiters to

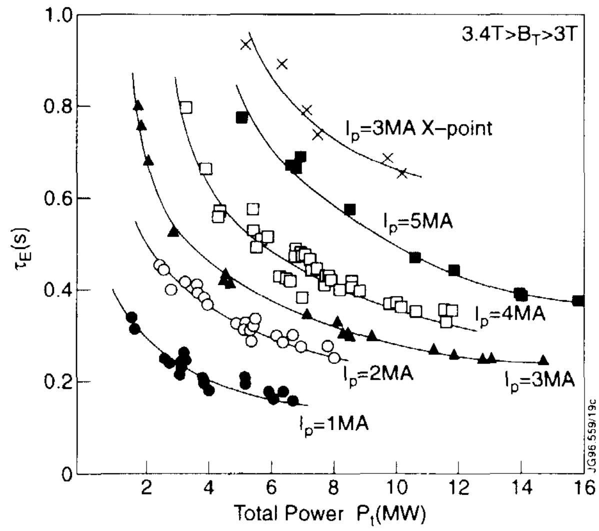

Fig.1: Behaviour of the energy confinement time τ_E with total heating power P_t in limiter (L-mode) and in X-point (H-mode) configurations

carbon toroidal belt limiters and a 50% covering of the inconel wall with graphite tiles, and finally by using beryllium tiles for one of the belt limiter and for the X-point target plates. This wall structure was supplemented by wall carbonisation first and by beryllium evaporation later (performed between daily operating sessions).

By progressively implementing all the above modifications, the triple fusion product $n_D\tau_E T_D$ went from $0.12\times10^{21}m^{-3}skeV$ (1983) to $0.9\times10^{21}m^{-3}skeV$ (1991) and the energy gain Q_{DT} equivalent went up from 0.01 to 1.07. Other significant performance achievements were a dilution factor $n_D/n_e\geq0.9$ and $Z_{eff}\leq2$, and, in separate pulses, an ion density $n_D = 4\times10^{20}$, an ion temperature $T_D = 30keV$ and an energy confinement time of $\tau_E = 1.8s$. This level of global performance was considered sufficiently attractive to perform the first ever controlled thermonuclear fusion D-T experiment, using a mixture of 11% T and 89% D, producing a peak fusion power of 1.7MW (more than 50% by thermalized reactions) and ~2MJ of fusion energy.

These results provided a clear indication that a tokamak fusion reactor should be conceived as a D-shaped machine, with high plasma current and large plasma volume, elongated plasma cross section with X-point configuration and low 'Z' materials for plasma facing components (see ITER) [5].

2. THE DIVERTOR PROGRAMME

However, a passive control of impurities is not sufficient to maintain high performance for more than one second, due to the combination of MHD instabilities and excessive accumulation of impurities in the X-point region. Therefore, a tokamak reactor requires an active control of the impurity influx into the plasma, because they increase the plasma radiation losses and dilute the fuel. The only known way to implement an active impurity control in a tokamak, is to equip the machine with a pumped divertor, which controls impurity levels, particle and energy exhaust and enhances plasma energy confinement ('H-mode').

The extension of the JET Joint Undertaking from 1992 to 1996, has been specifically granted to study the behaviour of a tokamak plasma with an axisymmetric pumped divertor (Mark I). The recent extension to 1999 has been mainly, but not solely, motivated by the need to tailor the divertor configuration to the ITER requirements. Therefore the divertor programme (Mark I, Mark II, Mark II Gas Box) became the focus of JET physics and engineering activities (Fig.2). Addressing the divertor issue implied a further major upgrading of the JET machine, involving installation of new poloidal coils, power supplies, ICRF antennae, diagnostics and a completely new configuration of the vacuum vessel first wall [6,7].

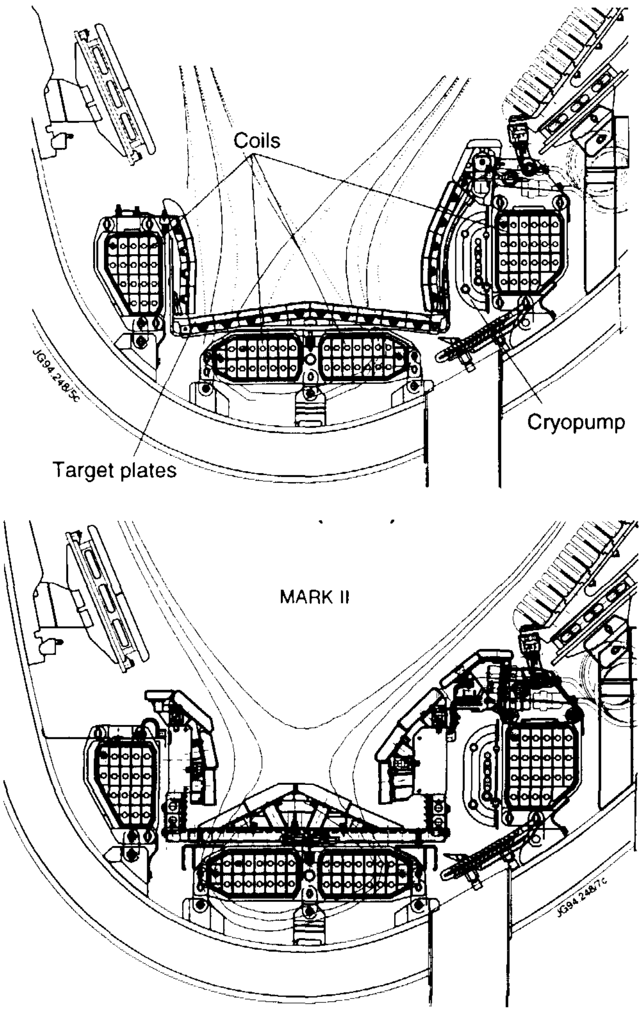

Fig.2: Sequence of JET divertor configurations with progressively more closed divertors

3. MARK I DIVERTOR

3.1. Design, manufacture and installation

Main components of the divertor are (see Fig.2):

(a) Four copper poloidal coils, Freon cooled, installed at the bottom inside the vacuum vessel allow magnetic configurations with X-point at a suitable distance from the target plates to be created, and they are supplied by 4 AC/DC converters (PDFA), with X-point sweeping capability;

(b) The target plates, arranged in a W-shaped contour, support the CFC tiles and collect the power released from the plasma;

(c) The toroidal cryopump, anchored to the outer divertor coil, permits the plasma density in the divertor region to be controlled, when cold gas is injected to minimise ionisation of impurities.

The new first wall assembly includes four pairs of ion cyclotron radio frequency (ICRF) antennae designed to match the divertor plasma shape, a re-shaped and re-positioned lower hybrid current drive (LHCD) launcher, 12 discrete poloidal limiters on the outer wall and 16 inner wall guard limiters, all covered with graphite tiles. In addition two sets of 4 saddle coils installed at the top and at the bottom of the vessel, should allow the control of m=2, n=1 MHD modes to be studied.

The refurbishing of the first wall has also required JET to modify existing diagnostics and to install new ones to measure divertor plasma parameters.

The divertor plasmas are more vertically unstable (open loop instability growth-rate γ~600-1000s^{-1}), due to the increase of the average plasma to wall distance, which reduces the passive stabilization effect of the vessel walls, and to the higher quadrupolar field associated with the divertor plasma magnetic configuration. A new set of fast radial field amplifiers (FRFA), 5kV, 5kA with a 2ms of response time over the full voltage range (-10kV, +10kV) was then provided.

The strategy of plasma control had to be completely re-assessed, leading to a new plasma position and current control system (PPCC), which allows control of the plasma boundary in real time, using 'intelligent' software for a selective control of plasma gaps (9) and/or poloidal coil currents (9).

Finally the new machine configuration with greatly enhanced electromagnetic equatorial asymmetry, required a new coil protection system (CPS), which, using the same technology as PPCC, prevents the coils from been operated outside allowed limits (current, voltage, thermal and mechanical stresses) and detects electrical faults.

The frequent modifications and upgrading of the JET machine through the years have resulted in an adjustment of the basic machine parameters [6]; in particular the installation of the divertor structure inside the vessel led to a substantial reduction of plasma volume (by ~20%).

3.2. Basic physics results [8]

The key physics result obtained was the demonstration that the pumped divertor works. In fact with 32MW of combined NB and RF heating, up to >180MJ were injected without any sign of 'carbon bloom', while without the divertor only 15MJ of injected energy would lead to a 'carbon bloom'. The care taken in the design and in the installation of the CFC divertor tiles, the successful use of the cryopump, and X-point sweeping have been instrumental in achieving longer, cleaner and more stationary H-modes (~20s with Z_{eff}~1).

It is worth noting that, in spite of the reduction of the plasma volume of ~20%, global performance almost identical to the one obtained without the divertor (and with a larger plasma volume) were obtained: Q_{DT} equivalent~1, $n_D\tau_E T_D$~$1.0\times10^{21}m^{-3}$skeV and the record value of the reaction parameter $R_{DD}=9.4\times10^{16}$reactions/s. Reliable plasma operation up to 6MA was achieved with a record plasma stored energy of 13MJ.

An important function of a divertor for a fusion reactor is to limit the energy deposited in the target plate, which can be achieved by enhancing the energy spread by radiation. By injecting a mixture of deuterium and nitrogen (or neon) in JET it was shown that radiation accounts for up to 80% of the energy released by the plasma (Fig.3).

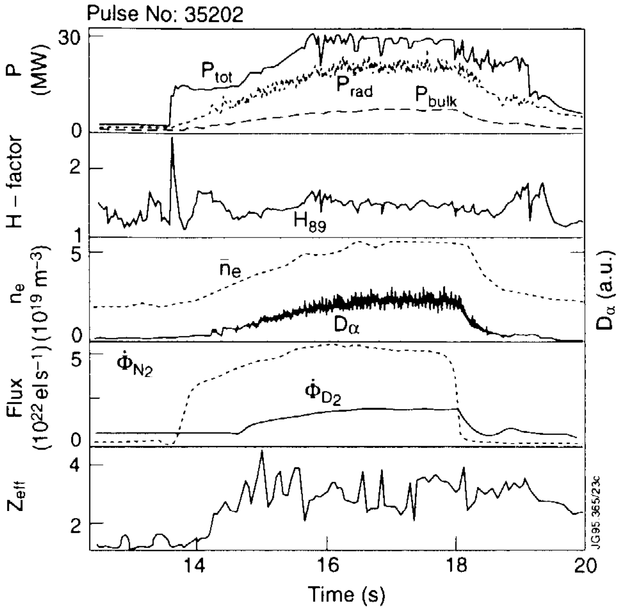

Fig.3: Radiative divertor H-mode plasmas with high power combined heating, showing up to 80% of radiated power

Finally two specific experiments requested by ITER were performed: the first to assess the effect of toroidal field ripple on plasma behaviour and performance, proving that the ripple produced in ITER with 20 toroidal coils should be acceptable, and the second to compare beryllium and CFC as divertor target plate material, showing very similar plasma performance, although beryllium melting could not be avoided.

4. MARK II DIVERTOR

4.1. Divertor design, construction and installation

The Mark II divertor has been conceived to provide a much closer configuration, leading to enhanced particle and impurity retention in the divertor chamber. It has been designed in order to minimize the amount of modification required to install different divertor configurations at a later stage (such as Mark IIGB). In addition, since extended periods of D-T operation are foreseen (DTE1 with Mark II and DTE2 with Mark IIGB), required modifications will have to be performed with the vessel highly activated (~6000μS/h) and therefore remote handling techniques will have to be used.

Mark II (see Fig.2) consists of a new inconel water cooled continuous toroidal tray, which acts as a support structure (above the divertor coils, but, unlike Mark I, with independent supports). On top of this structure tile carriers and CFC tiles are installed and can be subsequently replaced with different divertor configurations by means of remote handling. As part of the overall shutdown preparation an accurate full size mock up of the vessel was set up, using the spare machine octant as its central element, (In- Vessel Training Facility, IVTF) and used to test installation both hands-on and with remote handling techniques.

The Mark II installation work was completed in March 1996 and after an extensive subsystem and machine integrated re-commissioning without and with plasma, experiments started in June 1996 (Fig.4).

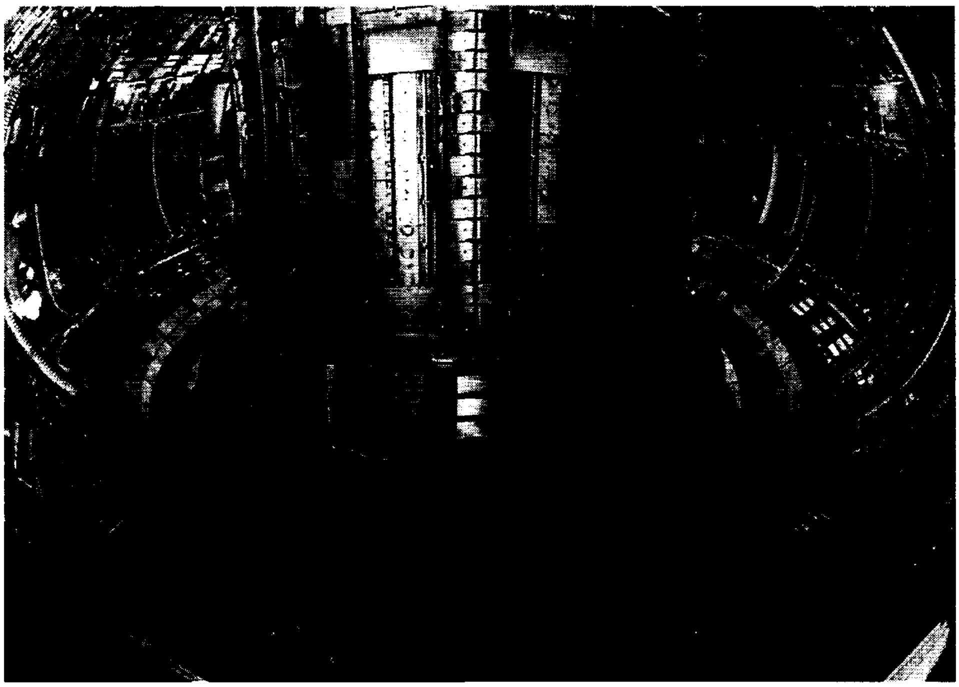

Fig.4: JET first wall configuration with Mark II divertor

4.2. Initial experimental results [9]

Only a few key issues will be briefly treated in the following, to support the improvement obtained in divertor behaviour with Mark II and to highlight fusion performance achieved.

(a) A comparison between Mark I and Mark II is shown in Fig.5, where the calculated levels of power, necessary, if applied for 10s, to raise the temperature of the tiles for the two divertors to 1400oC, are plotted, making X-point sweeping unnecessary (as required by ITER). The improvement is due to two major changes: the reduction of the angle between the exposed surface of the tiles and the poloidal projection of the magnetic field line, obtained by tilting them closer to the orientation of the flux surface, and the increased size of the tiles, which leads to an improved alignment with the neighbouring tiles and to the reduction of the number of gaps.

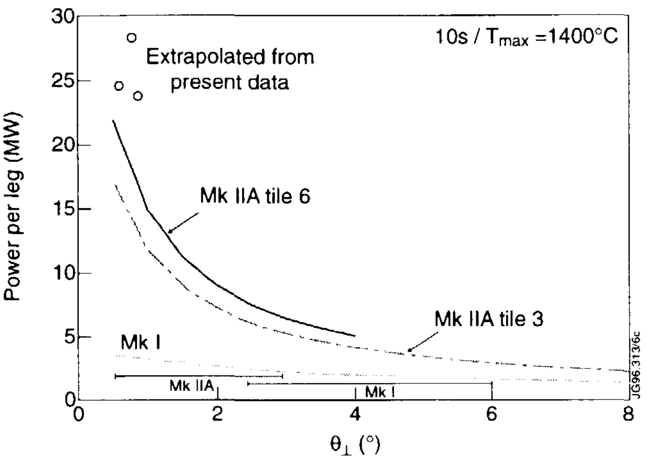

Fig.5: Divertor power handling capability: comparison between MarkI and Mark II, showing better performance of Mark II

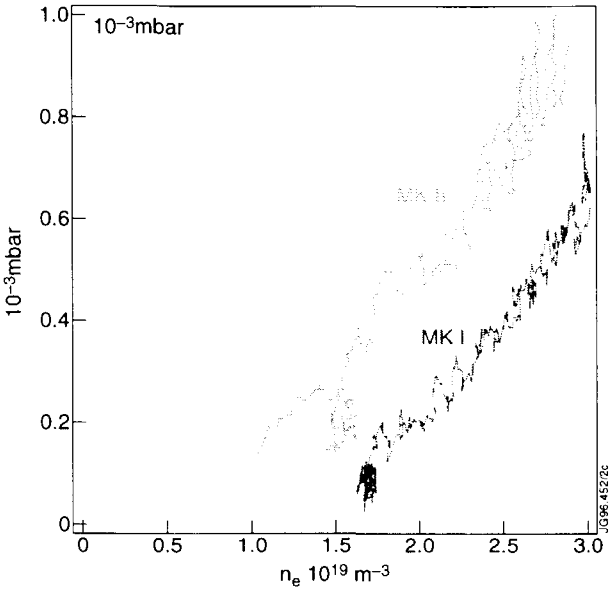

Fig.6: Neutral pressure at the cryopump: comparison between Mark I abd Mark II

(b) As expected the more closed configuration of Mark II resulted in a more effective retention of neutrals in the divertor region and near the cryogenic pump. In particular, at a given average bulk plasma density and for similar magnetic configuration, a higher pressure by a factor of ~2 of hydrogenic neutrals has been obtained in the proximity of the cryopump, resulting in a substantially increased pumping rate (Fig.6). Although preliminary, these results indicate that an improvement of the exhaust efficiency for impurities may be achieved.

(c) In view of the D-T campaign (DTE1), aiming at achieving high fusion power outputs (~10MW for >1 second), the major goal of JET 1996-97 operations, intensive experimental activity is underway to prepare suitable scenarios for the maximisation of fusion performance (Fig.7). Best performance has been achieved in the hot-ion regime with NB (18MW) and ICRF (6MW) combined plasma heating, resulting in a record value of plasma stored energy (14MJ) and in a neutron yield at the level of the previous campaign (~4×10^{16}neutrons/s).

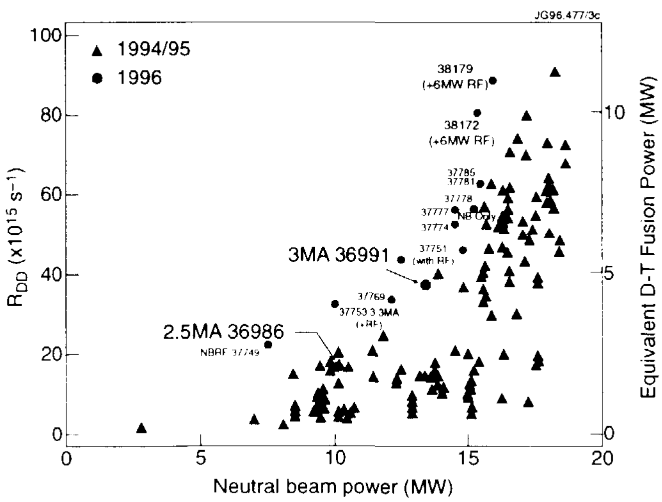

Fig.7: Fusion yield obtained in hot-ion mode in the 1994/95 (Mark I) and in 1996 (Mark II) experimental campaigns

(d) Plasma shear optimization studies are also actively pursued, due to the dramatic potential for increasing global fusion performance. Improved energy confinement with peaked density and temperature were already obtained in JET with pellet fuelling and, more recently, they have been established (without pellet injection) in TFTR and in DIII-D. An internal transport barrier is formed, which favours peak temperature profiles (Fig.8). A peak ion temperature T_i=30KeV and a record values of the electron temperature T_e=16KeV were reached. Very promising results in global performance have been obtained already, using combined NB and ICRF heating. An improvement in central confinement is indicated by a second rise in neutron production by a factor of 3 after the transition, reaching a neutron yield identical to the one in the hot ion mode (~4x10^{16} neutrons/s).

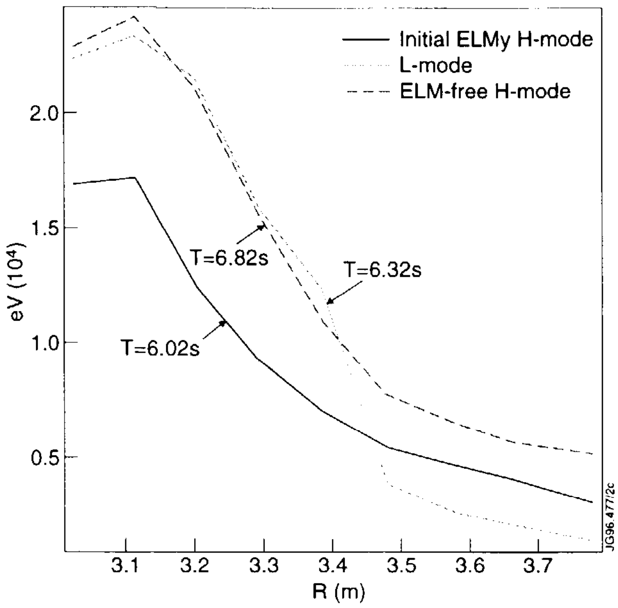

Fig.8: Ion temperature profiles in discharges with improved core confinement by shear optimisation

5. ENGINEERING ISSUES AND DEVELOPMENT

The implementation of the divertor programme required substantial machine and subsystem modifications, and upgrading and additional development was required to make the machine suitable to operate with Mark II.

5.1. Remote handling

The necessity to move from Mark II divertor to Mark IIGB divertor configuration, immediately after completion of DTE1, has required JET to plan this replacement by using remote handling techniques. The remote handling tools developed at JET throughout the years, to carry out routine maintenance, repairs and replacements, are centred around the *Mascot force reflecting servomanipulator*. The Mascot can be positioned where desired by large transporters, in-vessel and ex-vessel. The operator performs the job himself from a remote control room, through the master-slave device ('man in the loop'), because the sophisticated servo and control systems of Mascot give the operator the tactile sensation of actually doing the job with his hands. (Fig.9). In order to further ensure the operability of the system, about 20% of the Mark II modules have been installed by remote handling [10].

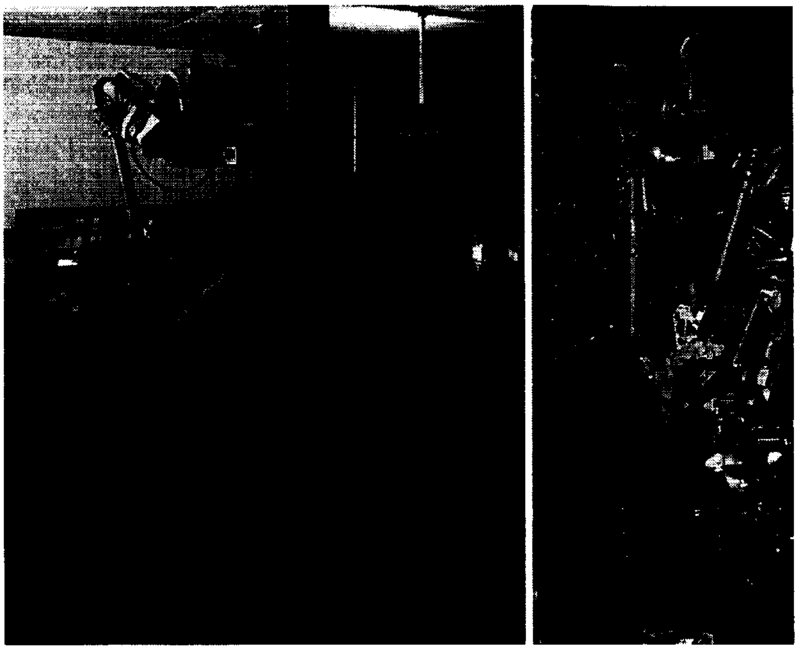

Fig.9: Installation of Mark II divertor tiles by remote handling using the master-slave Mascot servomanipulator

5.2. AGHS (Active Gas Handling System)

The AGHS is a full gas reprocessing plant, built to collect the gases from the Torus, to remove impurities from hydrogen, to isotopically separate the hydrogen gas into streams of protium, deuterium and tritium, to store the deuterium and tritium in U-beds for re-use and to inject D and T back into the Torus. Isotope separation makes use of cryo-distillation and gas chromatography. It was designed for a maximum daily throughput of up to 5 moles of tritium, 15 moles of deuterium and 150 moles of protium.

Commissioning with trace tritium (~0.08g) was performed already with a tritium-hydrogen gas mixture. Full tritium commissioning with about 3g of tritium is underway at present, to test the complete process using all AGHS subsystems [11].

5.3. Plasma control

The accuracy and flexibility of PPCC in controlling plasma discharges have been further enhanced to better meet the requirements of operation with Mark II, by re-designing both the *Shape Controller* (SC) and the *Vertical Stabilization System* (VS), including both hardware and software. SC was modified to increase from 9 to 15 the number of plasma-wall distances (gaps) to be controlled, since this type of control is to be preferred because plasma shape and position do not change during the experiment (Fig.10). Induced currents in the new toroidal support structure of the divertor would not allow the magnetic field null formation for breakdown, therefore VS has now the capability of generating a current proportional to the current induced in this structure, re-establishing the magnetic field null at the centre of the vacuum vessel [12].

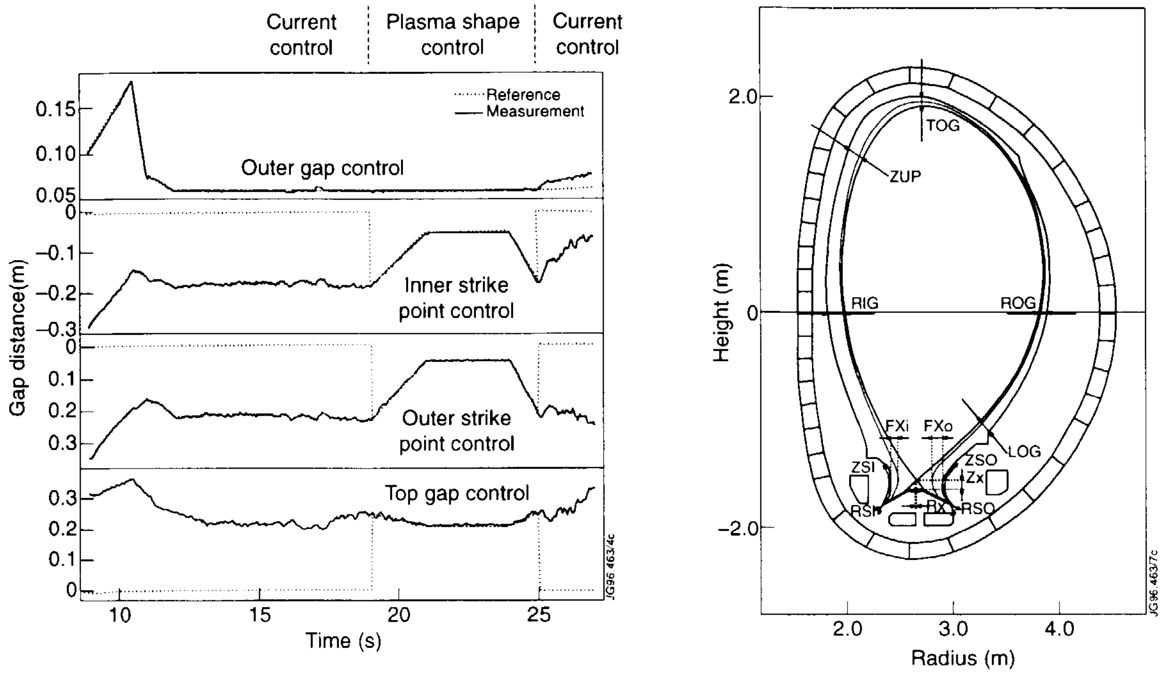

Fig.10: Plasma position and shape control, showing accuracy of PPCC gap control

The AC-DC convertors of the PDFA supplying the four divertor coils generate harmonics at a frequency of multiples of 600Hz. These coils are strongly coupled to the (bottom) magnetic field measurement detectors used for plasma control. *Harmonic power tuned filters* were installed, capable of 32dB attenuation for 600 and 1200Hz harmonics and 20dB for other harmonics. However the current induced in the divertor coils and circulating in the filters at plasma breakdown prevented plasma initiation. The filters are now being modified, adding a 10Ω resistor in series with each filter, to be switched off 100-200ms after breakdown [13].

5.4. Coil protection

The Coil Protection System (CPS), has been extended to include *ampere-turn protection* for both the toroidal coils (supplementing the existing hardware protection, based on voltage comparison between coils, DMSS) and the divertor coils.

Work is underway to implement *additional protecting functions*, based on the measurement of radial and vertical expansion of the toroidal coils, on the measurement of the forces on the supporting ring and collar teeth of the mechanical structure and on the shear stress on the coil insulation, calculated in real-time from the measurement of the coil current and the (normal) flux loops [14].

5.5. Additional heating and pellet injection

The operational flexibility of the Neutral Beam Injectors has been enhanced by further alleviation of limitations arising from 'shine-through' exposure of in-vessel structures by installing CFC tiles at appropriate locations. The beam-line calorimeters have been replaced with new calorimeters of enhanced power density loading capability ($20MWm^{-2}$), thus allowing longer conditioning/commissioning pulses. The beam lines have been made compatible with tritium in preparation for DTE1: 8 beam-lines will be operated at 80kV, 55A and supplied with deuterium (13.6MW) and 8 beam-lines, supplied with tritium, will be operated at 160kV, 30A (12MW).

The ICRF antennae were removed during the shutdown to incorporate a number of modifications. Additional capacitance was added to the cross-over straps linking the inner conductors to the incoming vacuum transmission lines, the antennae were displaced 6mm inwards and the lower straight section of the limiter sections were modified to better follow the plasma curvature, and several modifications were introduced in the electronic circuitry. As a result 15MW of coupled power to H-mode plasmas have so far been achieved in dipole phasing.

The Pellet Centrifuge, designed to deliver strings of 2-3mm deuterium pellets of up to 1 minute duration at a repetition rate of $40s^{-1}$ and velocities from 50 to 600m/s, is now operational and 5Hz strings of pellets have been launched into the plasma for pellet windows up to 5s.

5.6. Control and data acquisition system (CODAS)

Several improvements have been implemented to make operation activities more efficient. Amongst others, a new programme, the Object Monitoring System (OMS), substantially reduces the time for a particular plant mimic to appear on the screen, the Count-Down programme has been redesigned making better use of the UNIX computer network, and in-vessel video images are now taped routinely on the six video recorders of diagnostic KL1. Significant improvements in the ease of operation has been achieved by the use of the Real Time Power Control System (RTPCS), which controls the power of ICRF, LHCD and NBI in following the pattern defined for the pulse [15].

5.7. Plasma diagnostics

The JET comprehensive set of diagnostics undergoes continuous updating and upgrading [16]. The Lithium Beam Diagnostic (Fig.11) is a newly installed diagnostic, the main purpose of which is to provide continuous edge electron density profile measurements. The source injects 60keV lithium atoms vertically into the torus, with an equivalent current of 0.6mA. Emission from the beam-plasma interaction is collected using a periscope: its mirror is rotatable so that data may be taken at various minor radii. Within the periscope a lens directs light onto 50 optical fibres, thus sampling emission from a 17 cm long section of the beam with spatial resolution of 3mm. The fibres extend outside the Torus Hall to spectrometers fitted with CCD arrays of detector. The emission profile is analysed to deduce the required electron density profile.

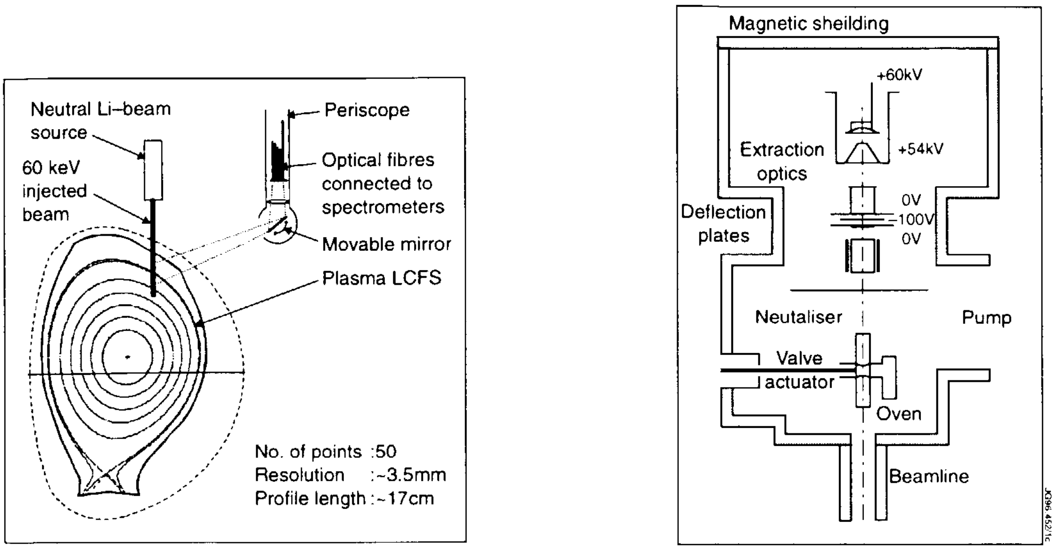

Fig.11: Schematic overview of the new lithium beam diagnostic for continuous edge electron density profile measurements

6. EFFECTS OF PLASMA BEHAVIOUR ON STRUCTURAL COMPONENTS

6.1 VDE History in JET

Since the early phase of operation JET experienced vertical instabilities and disruptions (now called VDEs, Vertical Displacement Events) causing vessel forces and displacements well beyond the levels considered in the original design. In fact the vacuum vessel was only supported by vertical elastic suspensions at each octant joint and at each Main Horizontal Port (MHP).

The first serious VDE appeared in May 1984 (pulse 1947): during the current ramp to 3MA, plasma elongation increased to 1.7, the vertical stabilization system exhibited growing oscillations leading to a power amplifier trip and to a downward VDE with subsequent current quench at 2.7MA. The vessel experienced a vertical force of ~2.5MN with a vertical transient displacement of ~10mm. Improvement were immediately undertaken on the vertical stabilization system by doubling the output voltage of the amplifiers and reducing by a factor of~2 (to 2ms) their response time. As an intermediate measure, two additional vertical supports were installed on each MHP (Fig.12) and, when ready in 1987, bottom and top inertial brakes were installed on each Main Vertical Port (MVP), each capable of withstanding 200 tons per octant. They allow vessel expansion during vessel heating and cooling (JET operates with hot vessel, ~300^{o}C), while blocking fast vessel movements. In addition, two inner restraining rings were welded inside the vacuum vessel to increase stiffness. The next VDE of major concern occurred in September 1989, during a severe disruption at 4MA (pulse 20802). The plasma moved 1 meter downwards producing a peak force of 2.6MN. Very large radial displacements (~10mm)

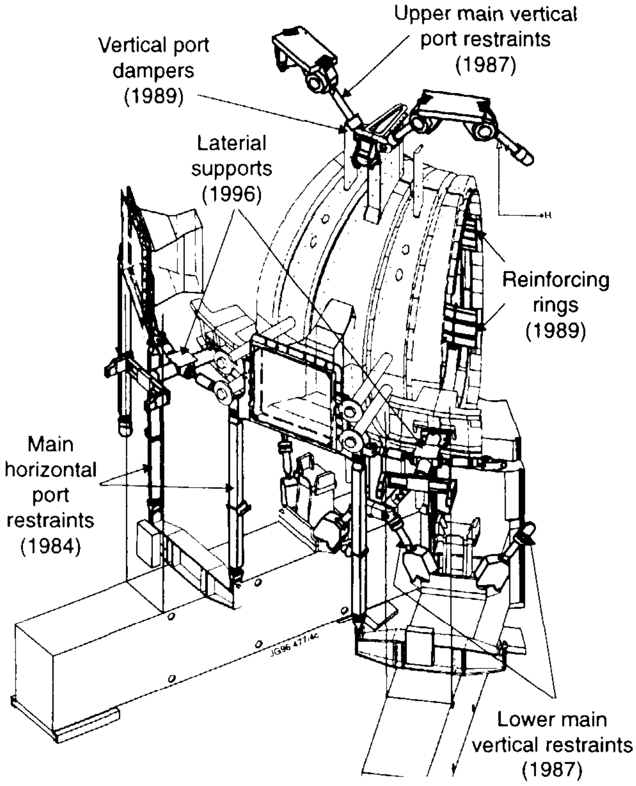

Fig.12: Evolution of vacuum vessel supports and restraints including new lateral supports to limit sideways displacements

of the lower and upper MVP were measured in the opposite direction, indicating an oscillating rolling motion. Therefore, radial hydraulic dampers were installed on each MVP. Another major VDE experienced in 1990 (pulse 22148), a disruption at 6MA with evolution of plasma current similar to pulse 20802, proved that the new supports were working properly, limiting the radial MVP displacement to <4mm in spite of the much larger vertical force of 4.5MN. A continuous upgrading of *machine instrumentation* went together with the development of the machine toward increased performance, including instrumentation to measure forces and displacements of the vacuum vessel and induced voltages and/or current in the in-vessel components.

6.2. VDEs in divertor configuration

It was always assumed, at JET and elsewhere, that the vertical forces were essentially toroidally symmetric. Therefore, when analysing a VDE event in May 1994 (pulse 29429), it was surprising to see that these forces were, instead, toroidally asymmetric [17].

In June 1995 (pulse 34708) vertical stabilization was lost, at a plasma current of 3.5MA, leading to a plasma upward vertical displacement of 1.2m. The vertical force was only ~1.8MN, but not uniform toroidally, and the most striking observation was that the radial displacement measured indicated that the whole vessel was displaced sideways, in direction octant 1 to 5, by 5.6mm. Moreover, the plasma current centroid in octant 3 becomes higher than

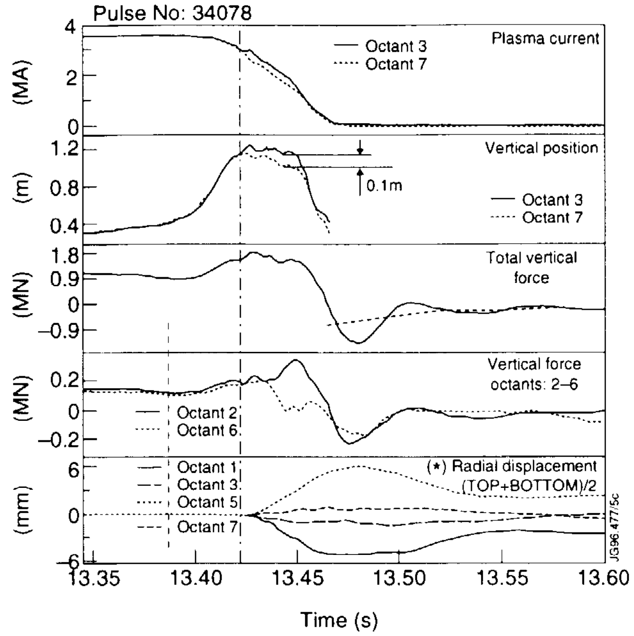

Fig.13: Description of the vertical displacement event (VDE) in pulse No. 34078, showing vacuum vessel sideways displacement and tilting of the plasma ring

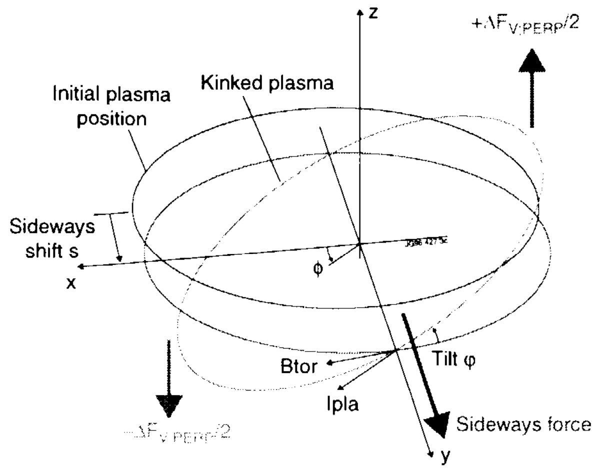

Fig.14: Plasma kink-mode model proposed to explain sideways forces and displacements

in octant 7 by 0.1m, at the start of the VDE (Fig.13). This can be interpreted as a mode m=1, n=1 ('kink' mode), equivalent to a tilting of the plasma giving a resulting force acting on the plasma, due to the interaction with the toroidal magnetic field (Fig.14). The q at the boundary was 0.8, therefore the condition for a kink instability was satisfied. The plasma mode was found to be almost fully locked, that way the sideways magnetic force was in one direction only (octant 1 to 5). Since each plasma element is in quasi-stationary equilibrium, this force would be balanced by other toroidally asymmetric forces, such as induced and halo current forces [18]. It was found that, indeed, toroidal asymmetries in vessel forces correspond to asymmetries in the toroidal distribution of the halo currents, flowing to the mushroom tiles installed toroidally on top of the vessel for protection purposes (Fig.15). While the total halo current accounts in general for≤ 20% of the plasma current, the local halo current may exceed the average by a factor of ~2.

Previous disruptions with and without divertor were then re-assessed and, in a number of them, similar asymmetries were found, more significant with divertor plasmas.

During the subsequent shutdown, it was discovered that these sideways motions caused damage to the seals of the two rotary valves connecting the MHP to the neutral beam injection boxes (octant 4 and 8). A new system of hydraulic lateral restraints linking the MHP to mechanically fixed bridges between the mechanical structure the transformer limbs was installed (see Fig.12). They form a complete belt surrounding the vessel, and a reduction of the radial displacement during VDEs is expected [19]. It is obvious, however, that the appropriate way to

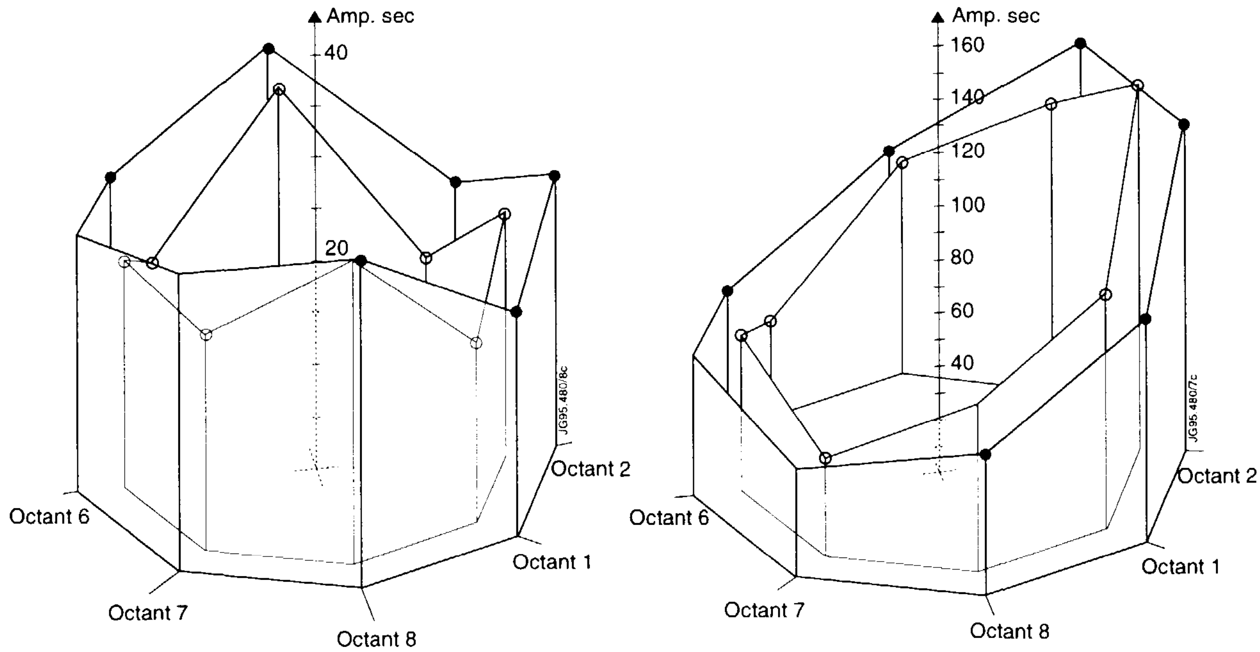

Fig.15: Time integrated mushroom tile halo currents in VDEs with and without torus sideways displacement

counteract these forces and displacements, would be to support the vessel directly and not through the ports. Unfortunately the engineering relevance of VDEs was not known to the designers of JET and the other large tokamaks in the mid seventies. Therefore, when the first serious event occurred in 1984, the vessel was not accessible and the only choice was to support the vessel at the MVPs and the MHPs and to measure forces and displacements at the ports.

In conclusion, the knowledge acquired so far is insufficient to devise means to prevent such events, while conducting an aggressive experimental programme at JET. Our understanding and intuition suggest that, as an electrical machine, the tokamak should not be compared to a transformer, but rather to a motor (or a generator), where the plasma (rotor) is moving. As in an electrical machine, both mechanical and magnetic accuracy are needed to limit the so called 'unbalanced magnetic pull', which make the rotor oscillate around the mechanical axis. Therefore, the adverse effects of VDEs should be reduced, if the vessel walls would be smooth and toroidally symmetric and if 'error' magnetic fields would be reduced as much as possible.

7. FUTURE PLANS AND DEVELOPMENT

7.1. Operation with D-T gas mixtures with Mark II and Mark IIGB

As it appears in the Table I, a 4 month period of D-T operation is planned in early 1997 (DTE1, Deuterium-Tritium Experiment 1), aiming at the production of ≥10MW of fusion power for several seconds, using a 50%-50% D-T gas mixture and with a neutron budget of $2x10^{20}$

neutrons (to limit maximum level of activation for Mark IIGB installation). Aims of the experiments are long fusion burn at Q_{DT}~1, so as to allow studies on α-particle heating to be initiated, studies on isotopic effects on confinement scaling and on H-mode detached radiative divertor plasmas. This extended period of D-T operation will also allow testing of the performance of the JET facilities in active conditions, such as the AGHS and the relevant diagnostics.

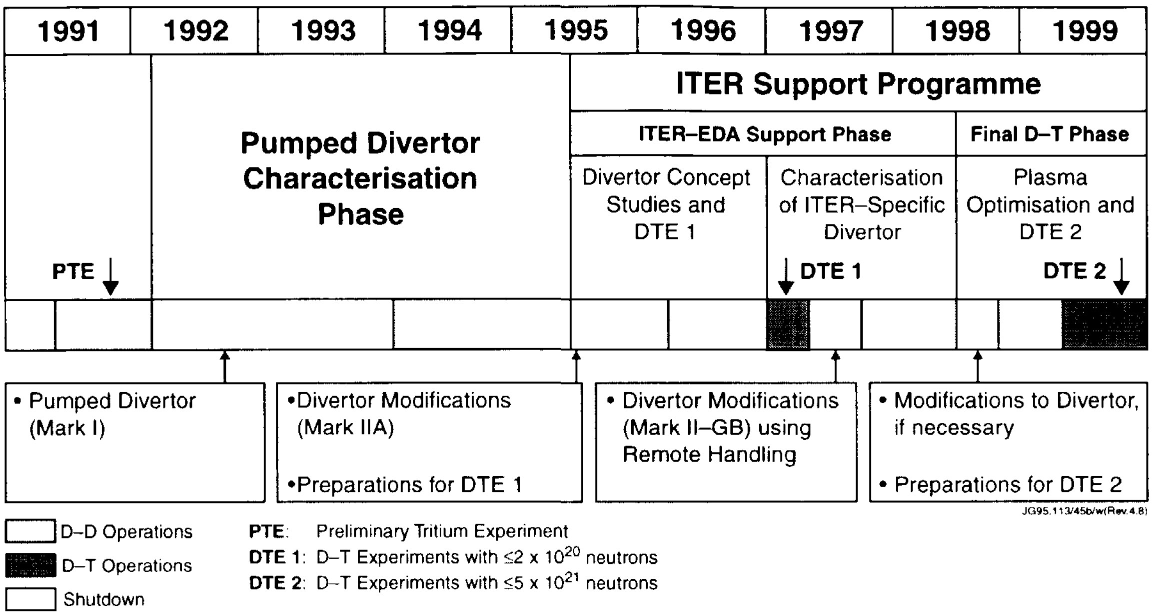

Table I: JET programme to the end of 1999

The gas box divertor, Mark IIGB (see Fig.2), will then be installed using remote handling technology. It should operate in such a way that most of the power from the plasma to the divertor is radiated. Both target plate and tile carriers will be manufactured in CFC, due to the difficulty to shield all areas of supports [20].

A more extended period of operation in D-T (DTE2), is planned for the second part of 1999 (neutron budget of 2×10^{21} neutrons). DTE2 should take advantage of all performance improvements implemented in the intervening period, such as magnetic shear optimization, upgrading of the toroidal magnetic field and of both NB and ICRF systems, and of the experience gained during DTE1. The study of α-particle heating will then be a major objective of the experiment.

7.2. Machine upgrading

The most straightforward way to increase global fusion performance is to upgrade machine parameters and additional heating power. In the late eighties JET has increased plasma current capability from the design value of 4.8MA to 7MA in limiter mode and more recently, to 6.0MA in H-mode divertor plasmas. Extensive studies are now being performed to assess the possibility to increase the *toroidal magnetic field* from 3.45T to 4.0T [21]. These studies are conducted on three levels:

(a) FEI stress analysis has shown that, for a variety of high performance scenarios, the shear stresses on the interturn insulation and tension on the copper brazed joint are well within the design capabilityof the coils;

(b) Evaluation of the manufacture documentation, including tests on insulation and on brazed joint samples and on the prototype coil, indicate an acceptable margin of safety at 4.0T;

(c) It is well known that between 1989 and 1991 three TF coils developed interturn electric short-circuits due to water leaks. The water coolant was replaced by Freon and no more faults became apparent. It has been decided to use these coils to perform further test. One of the coils has been cut for inner visual inspection and to extract samples, now under testing for shear stress in the interturn insulation (Fig.16). Deflection tests will also be performed on a spare coil and on one of the (electrically) faulty coils.

Fig.16: Cuts of TF coil 3.1 for visual inspection and for extracting samples for insulation shear stress testing

When all these tests will be completed the JET governing bodies will considere allowing operations above 3.45T. To avoid any delay, the order for the upgrading of the TF power supplies has been placed already.

Experiments conducted in the *Neutral Beam* test bed using a prototype modified accelerator structure have demonstrated the capability to increase the ion beam current from the present 30A to 60A at 140kV [22]. While minor modification would be required by the NB injectors, additional power supplies (PS) are required, since the present PS are for 60A at 80kV (or for 30A at 160kV). Taking into account cost considerations it has been decided to consider an upgrading of the power supplies to 60A at 120kV for one box (with an increase of ~7MW of beam delivered power). The design is based on AC/DC converters and on IGBTs as switching and regulating elements instead of tetrodes, used in the existing power supplies [23].

Studies have also been initiated on a power upgrade of the *ICRF* heating system, by using wide band matching [24].

8. SUMMARY AND CONCLUSIONS

- The key results of the first ten years of JET operation (without divertor) have shown, that large plasma volumes, high plasma currents and X-point magnetic configurations are essential features of a fusion reactor.
- This has been possible, because the inherent flexibility and design margins built into the original JET design have permitted extension of machine engineering performance well beyond the design parameters.
- The PTE1 fusion experiment, performed in 1991, led to the first production of controlled thermonuclear power up to 1.7MW for about 1 second, showing however that passive control of impurities with carbon and/or beryllium as first wall material is not sufficient to control the impurity influx into the plasma.
- With the extension of the JET Joint Undertaking to the end of 1999, the JET programme can now address in a comprehensive way the active control of impurity issue, testing different divertor configurations (Mark I, Mark II and Mark IIGB).
- The experiments conducted so far with Mark II have shown the expected better performance as compared with Mark I, namely a much improved power handling capability and a better retention of neutrals in the divertor region.
- The key physics objective during the next few months is to optimize plasma configuration leading to high plasma performance, so that DTE1 can meet its goal of high Q_{DT} and high neutron yield.
- The technology of remote handling for in-vessel installation of the Mark IIGB structure under active conditions has been fully proved in the IVTF and in the installation of the Mark II divertor target plates.
- The AGHS is now operational and final commissioning with 3g of tritium is underway, while the infrastructure for an extended D-T operation is in place.

- New physics scenarios, such as magnetic shear optimization, and engineering upgrading such as the increase of the toroidal magnetic field to 4.0T and the increase by several MW of additional heating power of both neutral beam and radio-frequency, will allow JET to enter in depth into the a-particle physics of a burning plasma with DTE2, planned for the second part of 1999.
- The damaging or potentially damaging effects of plasma behaviour on the structural components of the machine following VDEs, are to be considered a major problem to be addressed in present divertor tokamaks and in ITER.

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