JET WITH A PUMPED DIVERTOR TECHNICAL ISSUES AND MAIN RESULTS

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ABSTRACT

The most recent modification to JET has been the installation of a single-null pumped divertor, for active control of plasma impurities. This is to address central physics issues relevant to the design of a 'next step' tokamak. Experiments conducted during the 1994-95 campaign, with plasma currents up to 6MA, have shown that the *Mark I* divertor, which makes use of strike point sweeping across the target plates, is a suitable tool to control the influx of impurities in the plasma core. The operation of a tokamak with a pumped divertor has been characterised in detail. However the divertor configuration must be optimised to better meet ITER requirements. Therefore an improved (more closed) divertor structure, which may not require sweeping, is under assembly at present (*Mark II*). It is designed, in addition, to allow divertor tile structures to be fully replaceable by remote handling techniques, following D-T fusion experiments. New types of events involving electromechanical interactions of plasma with the vessel and in-vessel structural components have been encountered, due to plasma vertical instabilities and disruptions (such as toroidal asymmetries of vacuum vessel forces and side-ways vessel displacements). The physics and engineering experimental work performed in JET is primarily dedicated to the finalisation of the ITER design.

I. INTRODUCTION

The Joint European Torus (JET) has operated since June 1983. The inherent flexibility of the machine original concept [1] has permitted a great variety of plasma and fusion physics issues to be addressed. This has been possible due to engineering modifications and upgradings to improve plasma configurations and machine performance. These changes have followed the evolution in physics understanding and priorities towards the definition of parameters required in a fusion reactor, without major alteration to the basic machine structure [2].

Operation with a wide range of plasma currents and additional heating powers has confirmed the nearly linear scaling of energy confinement time with plasma current and its degradation with input power. Moreover, the H-mode regime was established at plasma currents up to 3MA in a X-point magnetic configuration. This showed a doubling of the energy confinement time compared with the L-mode, and confirmed earlier findings obtained with smaller tokamaks, ASDEX in Europe and PBX in the USA. These JET results, obtained with reactor relevant plasma parameters, led to a major upgrading of JET's electromagnetic system, to increase the plasma current capability from 4.8 to 7MA in limiter configuration and to make X-point magnetic configurations possible in excess of 5MA.

It was clear, however that impurities were playing a key role in hampering further progress in plasma performance. Therefore, a second major upgrading was implemented. This consisted in progressively covering the Inconel vacuum vessel walls with low-Z materials, with graphite tiles and frequent wall carbonisation at first, and with beryllium tiles and Be evaporation later. Dilution factors (deuteron density/electron density, n_D/n_e) exceeding 90% and Z_{eff} less than 2 were achieved. As a consequence of all these improvements a fusion triple product $(n_D \tau_E T_i)$ above $0.9 \times 10^{21} m^{-3}$ keV s and a D-T equivalent Q_{DT} exceeding unity were achieved.

These parameters were considered suitable to perform the first ever controlled thermonuclear fusion experiment towards the end of 1991. This led to the production of 1.7MW of fusion power using a mixture of D(89%)-T(11%).

All the basic JET features, including D-shaped toroidal coils, vacuum vessel and plasma, single null magnetic configuration, high plasma current and beryllium as a primary choice for first wall material are incorporated in the present ITER design [3].

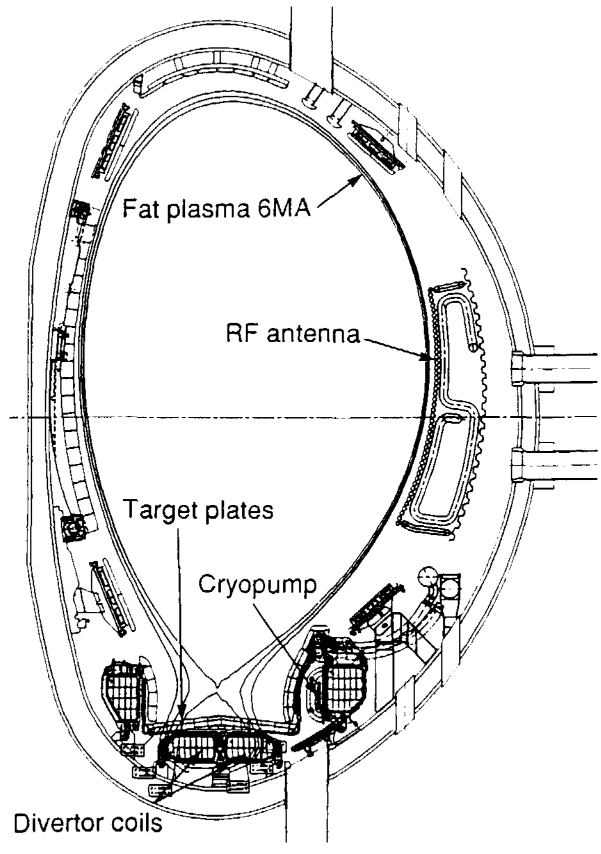


Fig 1: Design features of the Mark I divertor first wall structure, showing poloidal coils, target plates, cryopump and RF antennae

II. The Divertor Programme

The success in achieving short-lived high plasma performance in the 1991-92 experimental campaign, clearly indicated that a passive control of the impurities was not sufficient. Performance decayed abruptly after 1-2 seconds (*carbon bloom*), due to a combination of MHD instabilities and excessive production of impurities in the X-point region, which would be ionised and trapped in the plasma (**Fig.1**).

Active control of particle and power exhaust is required in JET and even more so in a fusion reactor, where long burn is a key requirement. This critical issue has been addressed in JET by installing a pumped divertor (*Mark I*) at the bottom of the vacuum vessel during the 1992-93 shutdown [**4,5,6,7**].

A. The pumped divertor

Main components of the Mark I divertor are:

• *Four Freon cooled copper poloidal coils,* which create an X-point magnetic configuration at sufficient height to allow installation of divertor components and control of the magnetic flux expansion in the divertor chamber. The coils allow for strike-point sweeping to reduce the specific power load to the divertor tiles;

• The target plates, which collect the power released by the plasma, consist of an inertially water cooled Inconel structure, which supports accurately shaped CFC tiles to eliminate exposed edges. These tiles have been replaced by beryllium tiles for comparison of performance;

• The toroidal cryo-pump anchored to the outer divertor coil, consists of a water cooled baffle, a liquid nitrogen copper back panel, an array of liquid helium cooled pipes and a chevron structure. Its function is the control the plasma density in the divertor chamber, in particular when cold gas is injected to minimise ionisation of impurities.

B. Further significant machine modifications

B.1.*First Wall* -The installation of the divertor implied a reduced plasma volume and a new plasma shape: therefore a complete re-design of the vacuum vessel first wall was required [**8**]. The eight *ion cyclotron radio frequency (ICRF)* antennae were replaced, to satisfy the plasma proximity requirement for effective RF power coupling; the *lower hybrid current drive (LHCD) launcher* was re-shaped for the same reason. The old toroidal limiters have been removed and replaced by *12 discrete poloidal limiters* on the outer wall for RF antennae and wall protection, and *16 inner wall guard limiters* were also installed. All limiters carry properly shaped graphite tiles. In addition eight *saddle coils,* four at the top and four at the bottom of the vessel were installed, to be used for control of MHD instabilities (m=2, n=1 modes) and for the study of toroidal Alfven eigenmodes, TAE (of concern for alpha particle confinement). The installation of divertor and of new in-vessel components required modification of existing *diagnostics* and the installation of new ones for measurements in the divertor region [**9**]. The new JET's first wall configuration is shown in **Fig.2**.

B.2.*Control and Protection* - Divertor (elongated) plasmas are more vertically unstable and require plasma-wall gap control, thus a new *plasma position and current control* (PPCC) system had to be installed. The system is designed with 'intelligent' software to control plasma-wall gaps and poloidal coil currents [**10**]. Similar technology has been used for the new *coil protection system* (CPS), necessary to cope with the greatly enhanced electromagnetic equatorial asymmetry of the machine. It comprises a wide range of protections in respect to overcurrents, overvoltages, limits to thermal and mechanical stresses, and model-based fault detection [**11**].

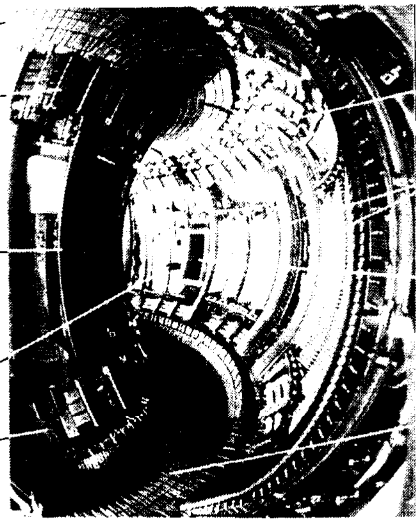


Fig. 2: The new JET first wall configuration, following the installation of Mark I divertor

B.3.*Power Supplies* - Four new 12-pulse thyristor *poloidal divertor field amplifiers (PDFA),* 500-650V, 40kA d.c. supply the divertor coils with strike-point sweeping capability [**12**] and a new *fast radial field amplifier system (FRFA),* based on GTO's technologies, 5kV, 5kA (or 10kV, 2.5kA), response time 0.2ms, meets the requirements for plasma vertical position control [**13**]. A system of four 1.5kV, 3kA *disruption feedback amplifiers (DFAS)* supply the saddle coils [**14**]. These power supplies, like most of those already existing at JET, are supplied directly by the 400kV grid. Thus, an additional 50MVAR of *reactive power compensation* was required to limit excessive AC voltage drops at 400kV and 33kV busbars.

The cross-section of JET with the pumped divertor poloidal coils is shown in **Fig 3**, whilst **Table I** compares the evolution of the main JET machine parameters. An important penalty (~20%) in plasma volume, adversely affecting performance, had to be paid to install the divertor within the existing machine structure.

III. Operation with the Mark I Divertor

Extended commissioning, first without and then with plasma, was necessary to become familiar with the 'new' JET, before the start of the experimental programme in May 1994.

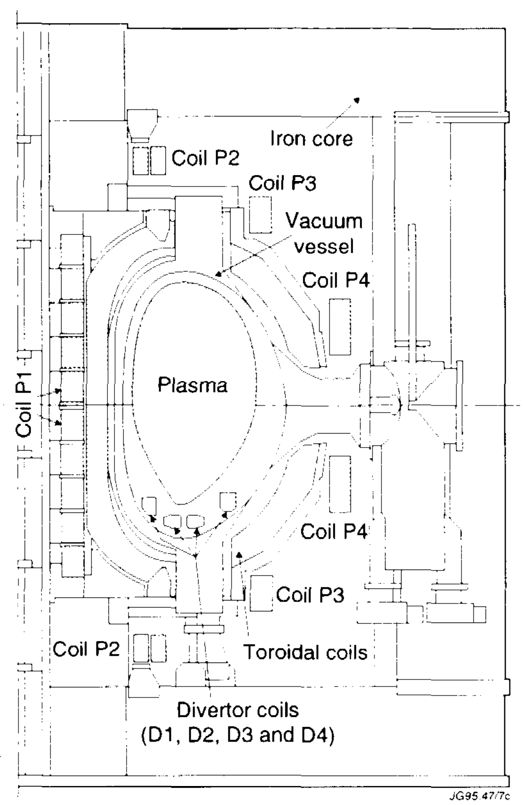


Fig 3: JET cross section, showing the new divertor magnetic configuration

Table I
Evolution of principal JET parameters with JET upgrading and with divertor

Parameter	Original 1983	Upgraded 1987	Divertor 1994
Plasma minor radius, a (m)	1.25	1.25	0.95
Plasma half height, b (m)	2.10	2.15	1.75
Plasma major radius, geometrical centre RO (m)	2.96	2.96	2.85
Plasma volume (m3)	150(a)	130(a)	-
	-	105(b)	85(b)
Plasma aspect ratio, RO/a	2.37	2.37	3.0
Plasma elongation, b/a	1.68	1.9	1.85
Toroidal magnetic field (at RO), BTO(T)	3.45	3.45	3.6
Flat top pulse length t (s)	10(a)	10(a)	10(a)
Plasma current, IP(MA)	4.8	7.0(b) 5.0(c)	6.0
Transform flux, f (Wb)	34	42	42
Neutral beam power, (MW)	15	20	22
Ion cyclotron power, (MW)	15	20	2
Lower hybrid power, (MW)	-	5	10

(a) Longer at reduced plasma current
(b) Limiter
(c) X-point

The most important result obtained was the demonstration that the pumped divertor operated successfully. In fact, in the X-point configuration (without divertor) an input of only 15MJ of injected energy would lead to a 'carbon bloom', with sudden termination of the high performance phase. With the divertor instead, up to >180MJ (from 32 MW of combined NB and RF heating) were injected without sign of discharge deterioration. The care taken in the design and in the installation CFC divertor tiles (which eliminated sharp edges), the successful use of the cryopump and X-point sweeping (a suitable technique to reduce target temperature from about 1000°C to 600°C or less) have been instrumental in achieving longer, cleaner and more stationary H-modes. As a result *plasma performance* could be maintained for *long pulses* (~20s, ~40τ_E) with Z_{eff}~*1* (**Fig. 4**).

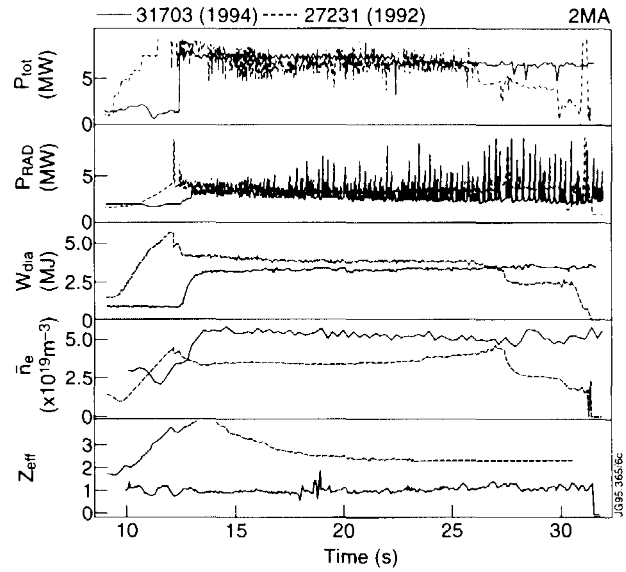


Fig 4: Comparison of similar discharges with pumped divertor (1994) and without divertor (1992), showing significant reduction in Z_{eff}

Electron temperatures of T_e~*15keV* were obtained with both combined heating and RF only. Full *current drive up to 3MA* was demonstrated with 7MW of LHCD power.
Although the plasma volume had to be reduced by about 20%, the best global performance almost identical to that without divertor was achieved. JET record value of the reaction parameter, R_{DD}=*9.4x10*^{16} *reactions/s*, was obtained. Moreover reliable divertor operation up to plasma currents of 5 and 6MA was established with record plasma stored energy of ~13MJ with 18MW of NB injected power and with combined RF-NB heating (**Fig.5**).
Initially, *ELM-free H-modes* were difficult to establish and to be maintained for several seconds. It should be emphasised, of course, that a reactor (ITER) needs to operate with ELM's to prevent accumulation of impurities (alpha-particles) in the plasma core. The

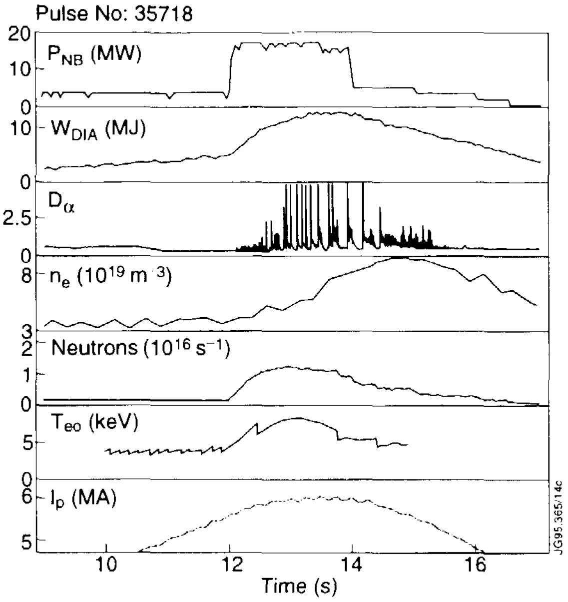


Fig 5: ELMy H-mode discharge with plasma current of 6MA, 18MW of NB power, showing 13MJ of stored energy

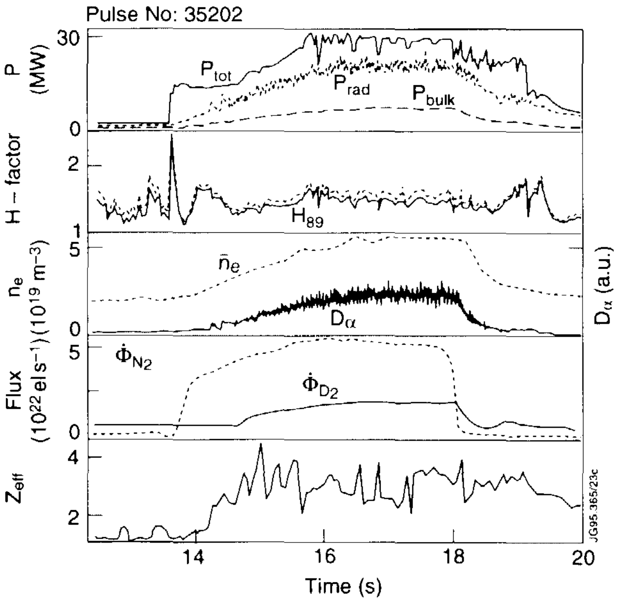


Fig 6: Typical pattern of radiative divertor experiments, with 32MW of combined heating and N_2 seeding, leading to 75% of power exhausted by radiation

appearance of *giant-ELM's* has often led to premature termination of the discharge **[16]**. These are issues deserving further studies, in the future.

A key issue for a reactor divertor is to limit the direct deposition of power on the target plates. This can be achieved by enhancing the *energy spread by radiation* on a larger surface. Preliminary JET experiments conducted by injecting a mixture of deuterium with neon or nitrogen, have shown that radiation may account for up to ~80% of the energy released by the plasma, albeit with some reduction in confinement (**Fig 6**). This subject will be specifically addressed at a later stage of the JET programme when Mark IIGB (gas box) divertor will be installed.

Two specific experiments, requested by the ITER designers were also performed in JET. The first experiment was to investigate the effect of toroidal field ripple. By re-arranging the d.c. power supplies to the toroidal coils in two independent sets, each supplying 16 of the total of 32 toroidal coils, it has been possible to vary the *toroidal magnetic field ripple* in the ITER-relevant range of 0.1% to 2% at the plasma edge. As predicted, losses of thermal and high energy particles (1MeV tritons and 125keV NB ions) were very small, while losses of intermediate energy particles (larger than 10keV) were somewhat higher than predicted. Therefore, ITER ripple (with 20 TF coils) should not have any effect on alpha-particle heating (and thus on the ITER global performance). The only critical ripple issues may be the load on the first wall due to alpha-particle losses and a reduction in plasma rotation **[17]**. The second experiment dealt with the *comparison between CFC and Beryllium.* Since the primary choice for ITER first wall is still beryllium, the CFC divertor tiles have been replaced with castellated beryllium tiles (to reduce thermal stresses). The most significant CFC plasma scenarios have been repeated and, in global sense, similar physics results have been obtained(**Fig.7**), in establishing and maintaining H-modes and in energy confinement times, while Z_{eff} values were slightly lower with Be tiles (oxygen is further reduced). The energy deposition on the beryllium tiles appears to be mainly toroidally symmetric . Since the melting temperature of beryllium is ~1300°C, it is considered impossible to operate a tokamak (and even more so a reactor like ITER) without some localised Be melting, caused by impulse heat loads and plasma disruptions. Therefore, experiments have been conducted causing intentional surface melting of the beryllium tiles, culminating in a *controlled beryllium melt experiment.* The preliminary conclusion is that beryllium

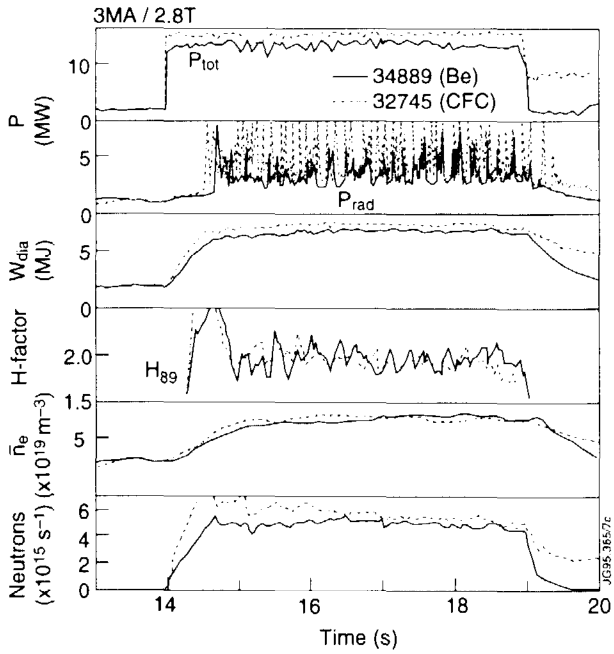


Fig 7: Comparison of two equivalent steady-state H-mode discharges, showing similar behaviour with CFC and Be target tiles

could be used as divertor target plate material for ITER. However, heat fluxes of $25MW/m^2$ expected in ITER following disruption events, would not provide a sufficient degree of passive self-protection due to enhanced radiation (vapour shielding). Therefore an active protection device should be considered [**18**]. Many other specific physics issues have been studied in great detail and the results are being progressively published, while the large amount of data collected are still being analysed.

B. Engineering Experience and Prospects

Such a new and extensive experimental campaign has been made possible by the extended commissioning of the 'new' machine and by the reliability of the major machine subsystems. A total of *7389 pulses* were performed, 23% were commissioning (with and without plasma) and 77% were operational pulses. A key role has been played by PPCC system, in successfully handling a great variety of discharge scenarios and by CPS in the effective protection of the machine with its new complexities and new type of faults not experienced before.

B.1. Plasma Position and Current Control (PPCC) System
The flexibility of the new PPCC has greatly simplified JET's operations, since it allowed a fairly accurate control of the plasma boundary. PPCC has proven to have the capability of *controlling the plasma distance* from the outer, inner and top vessel wall, the X-point position and the plasma current simultaneously, and to change control behaviour during the pulse to satisfy the different requirements of the various phases of the pulse. Switching from coil current to gap control does not affect plasma equilibrium (**Fig.8**). As regard *vertical stabilisation*, although the performance is presently limited by the noise introduced in the feedback signals by thyristor switching of the divertor power supplies, the system has demonstrated the ability to cope with plasma with growth rates of $800s^{-1}$. This has allowed the Mark I campaign to be conducted virtually without vertical instabilities not generated by plasma internal events. Improvements are now underway: a tuned power filter will be installed at the PDFA d.c. output leading to ≥32dB attenuation of the 600Hz and 1200Hz ripple noise; the computational power to allow more complex algorithms to be used will be increased, to include Soft X-ray signals for vertical stabilisation purposes; plasma shape control will be upgraded to achieve a more accurate control of the X-point position and the simultaneous control of the two strike-point positions [**19**].

B.2. Coil Protection System (CPS) CPS has extended the JET operating range by allowing the use of dynamic thresholds based upon on-line calculations (e.g. the central P1 coil current threshold which is a function of the toroidal field current). It has introduced a *real-time evaluation of the coil models* that made possible fault detection from un-measured quantities such as short-circuit currents, transducer faults, wrong power supply polarities, etc. A typical example of CPS intervention is given in **Fig.9**. It shows the protective action for excessive P1 average current (the six P1 central coils can carry up to 60kA because of the pre-compression of the toroidal field coils, while the outer coils can carry 40kA only). The algorithm has been optimised to minimise the overshoot of the current in the P1 end coils. TF coil protection includes transverse force protection (due to the tokamak torque), the coil interturn fault detection and the earth leakage protection. CPS has operated with a high degree of reliability [**20**]. CPS will now be progressively upgraded to a *Torus Protection System (TPS)* to include monitoring and analysis of stresses in the vacuum vessel.

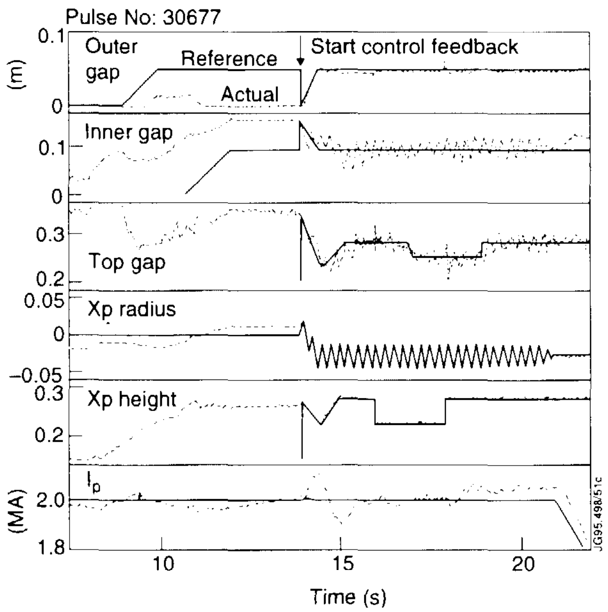


Fig 8: PPCC control of the divertor magnetic configuration with 4Hz X-point sweeping, showing the accuracy of gap controll

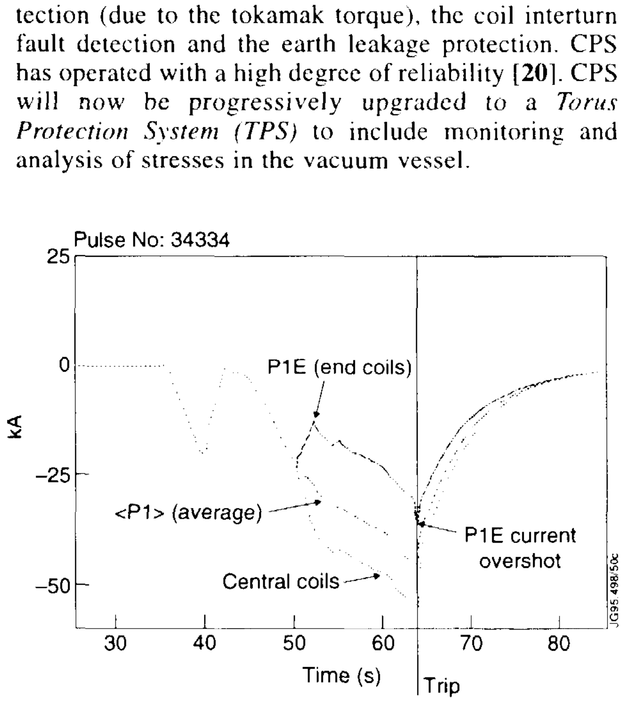


Fig 9 A typical example of CPF protective action against excessive current in the central solenoid end coils (PIE)

C. Plasma-Machine Interactions

The JET vacuum vessel consists of eight double wall octants, each made up of four bellows and five rigid sectors. Each octant carries a main horizontal port and two (top and bottom) main vertical ports for access.

The vessel has two inboard in-vessel and two outer out-vessel reinforcing rings. In vessel components are welded on the inner wall. The vessel support system allows for thermal expansion (the vessel operates at 250-320°C) and contains vessel movements in plasma operation. Plasma displacements and disruptions, in fact, cause impulse vertical and horizontal forces to be applied to the vacuum vessel, in-vessel components and divertor coils, leading to vessel movements and stresses. The plasma displacements *induce currents* in the structural components and *halo currents* flowing from the plasma to the structural components. The forces are generated by the interaction of these currents with the magnetic field.

Two types of vessel movements were considered in the design of the vessel and of supports: rolling and rocking motion, due to the fact that the centroid of the vertical forces applied on each octant is not in line with the reaction forces at the main vertical port restraints, causing a twisting moment around the toroidal axis, and a net inward motion caused by the toroidal current induced by plasma disappearance and by the poloidal current induced by the change in the diamagnetic flux.

We became aware of new phenomena during the 1994-95 divertor experimental campaign, namely, vertical *forces* on the vessel are *not toroidally symmetric*, causing overloading and additional shear stresses and *vessel side-ways motion*, caused by non symmetric horizontal forces applied to the vacuum vessel. While the basic structure of the first wall did cope extremely well with these forces and stresses, some auxiliary in-vessel components were damaged (saddle coils, beryllium evaporator head, glow discharge cleaning electrode, reciprocating probe, earthing straps and tile support rail for ICRH antenna), because when they were first designed, the consequences of these phenomena were not yet fully appreciated by tokamak designers [**21**].

While appropriate design modification have made the 'weak' components suitable for JET operation, these phenomena are not fully understood and JET is dedicating a special engineering-physics effort in this area. This implies theoretical work, analysis of data, implementation of new measurements. It is clear however that stresses in the vessel and in other structural components depend now not only on the magnetic field and the plasma current (as assumed by tokamak designers in the past) but also on plasma configuration and disruption scenario. There is no doubt that the finalisation of ITER design would benefit from progress in this work.

IV. Future Development

The JET Council has recently supported the extension of the JET Programme up to the end of 1999 and the Project is proceeding accordingly, while the extension of the JET Joint Undertaking is now being considered by the Council of Ministers of the European Union. Highlights of the extended JET Programme are divertor studies using divertor configurations as close as possible to reactor requirements and fusion experiments with D(50%)-T(50%) mixtures, DTE1 in 1996-97 and DTE2 in 1999 (**Table II**).

A. Divertor Development

It is important to validate the divertor modelling by comparing the most encouraging results obtained with Mark I with the performance of Mark II divertor, which provides a much closer configuration, enhancing neutral particle and impurity retention in the divertor chamber (**Fig.10**). *Mark II* divertor is being installed during the present shutdown (June 1995-March 1996).

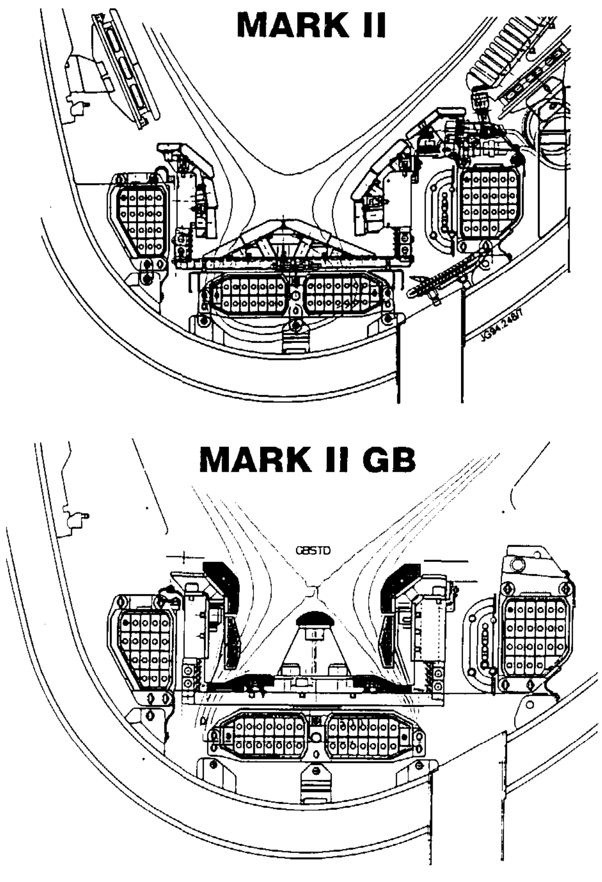


Fig 10: Cross-sections of the Mark II divertor now being installed, and of the future Mark II GB divertor

The support structure consists of a *continuous toroidal tray* (assembled in radial sectors) on top of which the target plates are installed and can be subsequently replaced by different divertor configurations, such as *Mark IIGB*, a configuration closer to ITER needs while maintaining the divertor coils and the toroidal tray in position.

B. Deuterium-Tritium Experiments

Following the preliminary Tritium Experiment in 1991, it was decided to further develop the machine to a more reactor-like configuration by the installation of a pumped divertor, prior to further D-T experiments.

Two periods of D-T operation are currently foreseen: a limited period (DTE1) to start towards the end of 1996 (four months) and a more extended period (DTE2) in 1999, after completion of divertor studies with the Mark IIGB (eight months).

As a result of JET's long pulse capability and its control of impurities, a key objective of *DTE1* would be the production of *≥10MW* of fusion power for several seconds, so that a-particle heating could play a role in the plasma power balance. In addition, a crucial objective would be the study of isotopic effects on confinement scaling and H-mode threshold power. Operation in ELMy-H-mode detached radiative divertor plasmas would also provide valuable information for ITER divertor design. It is expected to carry out the planned programme with a total neutron production of $2x10^{20}$ *neutrons*. DTE2 will have similar physics objectives and it is expected that through the experience gained DTE1 and after a period of plasma optimisation using techniques such as profile control, the fusion performance and the effects of a-particle heating of the plasma would be more significant. A total budget of $5x10^{21}$ *neutrons* is foreseen for *DTE2*.

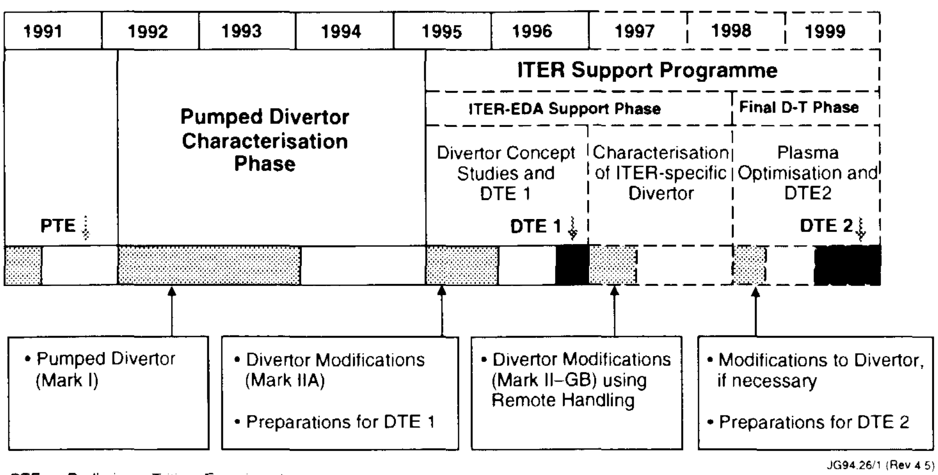
In performing such an extended D-T operation, full use will be made of the *Active Gas Handling System (AGHS)*, a gas re-processing plant to collect the gases from the torus, neutral beam system, pellet injection and diagnostics, to purify and isotopically separate these gas mixtures and to re-inject pure tritium and deuterium into the torus [**22**]. Therefore, the JET D-T experiments would also provide useful engineering experience on the technology of tritium handling for a tokamak reactor.

C. Remote Handling

Although the neutron budget for DTE1 represents a small fraction of the total neutron fluence foreseen in the life of JET, 12 to 18 months of cool-down would be required to allow a manned in-vessel intervention. Therefore, besides its physics characteristics, the Mark II divertor has an important and novel feature: it has been designed to allow replacement of the divertor target structure by full remote handling techniques. The tile carriers will be handled and positioned by the Mascot IV servomanipulator mounted on the articulated boom transporter, through octant No. 5 port. Navigation and pre-positioning will be carried out automatically using *teach-and-repeat techniques* . A shorter version of the articulated boom (Tile Handling Transfer Facility, THTF) will be used for transfer of the tile carriers between the Torus Hall and the vessel, through octant No. 1 main horizontal port. All R.H. tools exist and have been used in the past, but the THTF, which is now under manufacture. To validate the actual capability of the remote handling approach envisaged, part of the Mark II divertor target plates will be installed by remote handling. A mock-up test bed has been prepared, which makes use of the spare machine octant. Shutdown time for replacing Mark II with Mark IIGB divertor is of the utmost importance to limit machine down-time. Therefore, preparation for remote installation of the Mark II tile carriers have included *1000 hours trial* with the Mascot IV, Articulated Boom, Viewing Cameras and associated control systems performing a typical tile exchange for 24hrs/day, for five days per week. The availability of the boom was 97% and that of Mascot was 99%. Moreover two months of mock-up trials were performed to establish the precise techniques to be used and to create the 'teach-and repeat' files [**23**]. These results give a high degree of confidence on thetime-scale eventually foreseen for the Mark II to Mark IIGB target plate exchange with an active machine following DTE1 (**Fig.11**).

It will be the first time that such a complex remote handling operation has been performed on an active machine, providing most valuable engineering experience for the finalization of ITER design.

Table II
JET Programme to the end of 1999



PTE: Preliminary Tritium Experiment

DTE 1: D-T Experiments with 2×10^{20} neutrons

DTE 2: D-T Experiments with 5×10^{21} neutrons

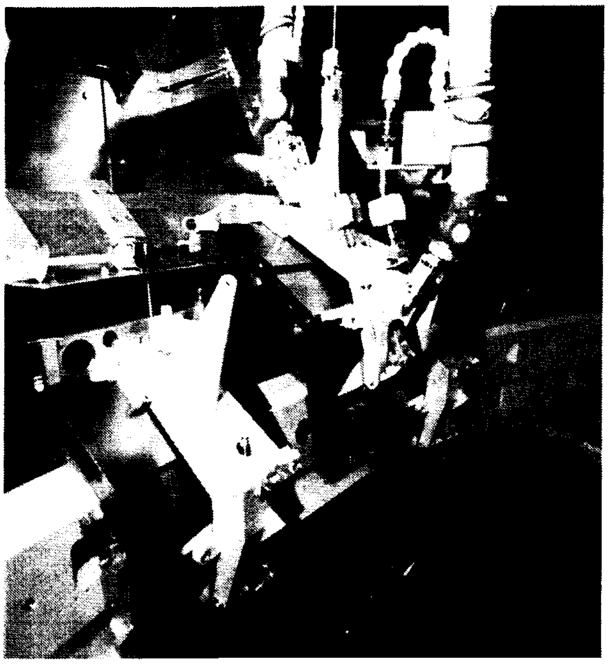


Fig 11: Installation of Mark II divertor side target plates in the JET in-vessel mock-up, using remote handling techniques

D. Machine Enhancements

D-T global fusion performance (fusion triple product $n_D t_E T_i$ an energy gain Q) would benefit tremendously by certain engineering enhancements of the JET machine. Consideration is now being given to the possibility of increasing the toroidal magnetic field from 3.45T to *4.0T for ~10s* flat top, and by an increase of *4 to 6MW* in the injected neutral beam power.
A preliminary assessment of the electromechanical capability of the JET coils and of the mechanical structure indicate that, for the scenarios considered at 6MA plasma current, forces and stresses are still acceptable (shear stress in the central solenoid P1 electrical insulation ≤20MPa and well within the capability of the TF coils and of the mechanical structure). The design for upgrading the current capability of the TF power supplies from 67kA to ~80kA is underway.
The design to upgrade each one of the 16 NB injectors from 80kV, 60A to 120-140kV, 60A is being performed. What is needed is to procure 16, 40-60kV, 60A power supplies to top-up the present ones. These new power supplies are based on thyristor AC/DC rectifier followed by an IGBT high frequency inverter, for limiting the DC output voltage ripple and for NB injector protection. No major modification to the present injectors would be required.

V. Conclusions

The following points summarise the main conclusions that can be drawn:
• Twelve years of JET operation have been essential to define the *plasma volumes* and *plasma currents* required for the design of a reactor-like tokamak;
• The inherent flexibility of the JET design has allowed substantial modifications and *engineering upgradings* of the machine, including an axisymmetric pumped divertor, with the capability to modify its configuration;
• With the *Mark I divertor*, the key problems of power and particle exhaust have been successfully addressed;
• *High fusion performance* has been demonstrated in JET with a divertor, in spite of a 20% reduction of the plasma volume to accommodate the divertor structure;
• The foundations have been laid for further divertor studies with a *Mark II* version and for D-T operation;
• The technologies for *remote handling* of the divertor target replacement under active conditions and for handling tritium have been demonstrated;
• Progress has been made in the analysis and in the understanding of the effect of plasma *vertical instabilities and disruptions* on machine structural components. This allows appropriate measures to be taken for in-vessel component design modification and for more effective machine protection;
• Design work is underway for *enhancing the machine capability* to allow further enhancement of plasma global performance;
• Most of the physics and engineering experience gained during the 1994-95 experimental campaign is relevant to the *finalisation of ITER design.*

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