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JET with a Pumped Divertor: Design, Construction, Commissioning and First Operation

JOINT EUROPEAN TORUS JET

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ABSTRACT

Eleven years of JET operations have brought studies of tokamak plasmas up to reactor relevant performance. This has resolved some key issues for the design of a 'next step' machine. However it has been clearly shown that active control of the influx of impurities is essential for a long burn fusion reactor. Therefore, a single-null pumped divertor has been installed inside the JET vacuum vessel to study impurity control, and it is now operational. It consists of four poloidal coils (manufactured inside the vessel), CFC target plates, and a toroidal cryo-pump. The new magnetic configuration produces plasma shapes which required a complete redesign of the first wall, including RF antennae and limiters. New power supplies were introduced for the divertor coils and for plasma fast vertical position control. Additional cooling systems, a new digital plasma control system, a comprehensive machine protection system and new diagnostics to measure plasma parameters in the divertor region were also installed. Well controlled 4MA discharges have already been achieved.

1. INTRODUCTION

The Joint European Torus (JET), in operation since 1983, is the largest experiment of the coordinated fusion research programme of the European Union. The mission of JET is to provide the key physics and plasma engineering data for the design of the 'next step' tokamak, by operating and studying a plasma in conditions and dimensions approaching those needed in a tokamak thermonuclear reactor [1]. JET design started in 1973. The machine was then constructed and started operation in June 1993.

The essential objective of JET is to obtain and study a plasma in conditions and dimensions approaching those needed in a thermonuclear reactor. The realization of this objective involves four main areas of work:

(i) the scaling of plasma behaviour as parameters approach the reactor range;

(ii) the plasma-wall interaction in these conditions;

(iii) the study of plasma heating;

(iv) the study of alpha particle production, confinement and consequent plasma heating.

The results obtained during eleven years of JET operation and engineering upgrades and the prospects of the role of JET in the development of fusion research in Europe have been evaluated by an European Joint Working Group (JWG). This body was set up by the Fusion Technical & Scientific Committee-Programme (FTSC-P)and by the Programme Committee (PC), to advise the Consultative Committee on the Fusion Programme (CCFP) on the establishment of priorities for the 1994-98 European Fusion Programme. The JWG stated that

JET is an exceptional instrument for the scaling towards the Next Step and therefore, *extension of JET's operation beyond 1996...should be examined by the appropriate bodies...*". The JET governing bodies are now considering a proposal of the JET Director for an extension to end 1999.

The main reason for such an extension is the recognition of the unique role of JET in providing vital information and data for the development of the ITER design concept. In addition it should assist in the finalization of key issues, such as the pumped divertor for the active control of impurities and the role of alpha-particle physics in an extended Deuterium-Tririum (D-T) experimental campaign at high (i.e. reactor grade) plasma performance. JET will not only provide physics data but also possible engineering solutions to ITER relevant design issues.

This paper describes the design, construction and commissioning of the 'new' JET. Initial results and their relevance to ITER issues are described in a companion paper [2], also presented at this Symposium.

2. JET DESIGN CONCEPT

Within certain technical and financial constraints, priority was given in the JET design to a tokamak with large plasma volume and current, capable of extended D-T operation [3]. This led to a machine with toroidal coils, vacuum vessel and plasma of 'D'-shape cross-section. These basic features, conceived in 1973, now common to other tokamaks, including ITER, were somewhat controversial at the time. In fact the concept of JET departed considerably not only from the outline design proposed by the European Study Group that preceded the work of the JET Design Team, but also from those of the other large tokamaks being designed in the same period, such as TFTR (Tokamak Fusion Test Reactor) in the USA, JT-60 (JAERI Tokamak-60) in Japan and T-15 (Tokamak-15) in the USSR (Fig.1).

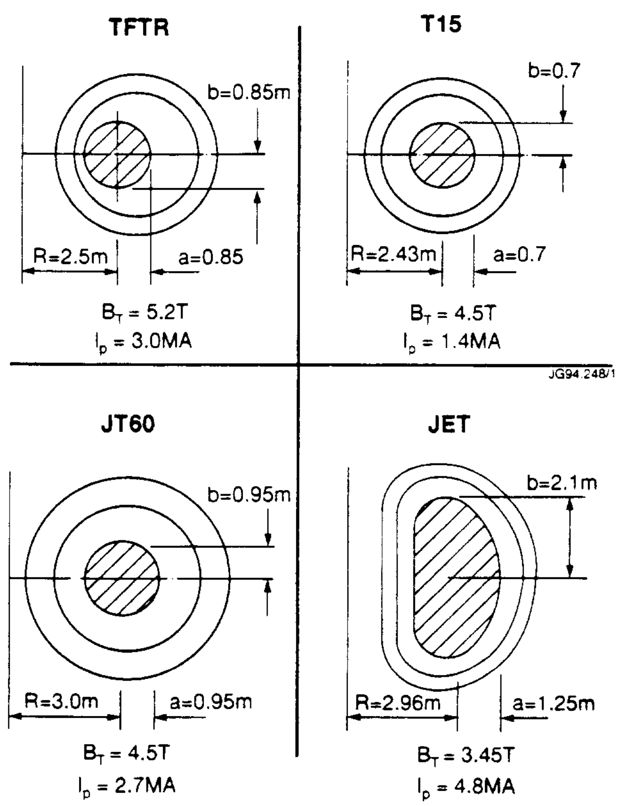


Fig.1: Original design of large tokamaks: comparison of cross sections

Including the objective of an extended D-T phase for JET, made design solutions heavily determined by remote handling capabilities. Remote handling transporters and tools were conceived and developed coherently with the machine design. Moreover, the machine building was designed to nuclear standards, and key diagnostics systems were located outside the radiation containment building (Fig.2). Finally, the Active Gas Handling System (AGHS) was conceived as a full D-T fuel reprocessing plant.

It is due to this vision in conceiving the JET tokamak as an advanced and flexible device that JET is and remains ready to meet new challenges encountered in fusion research. For several years, JET has operated well above design parameters and with newly developed magnetic configurations. Therefore, JET is a unique tool within the world programme and its results are important for the effective design of ITER.

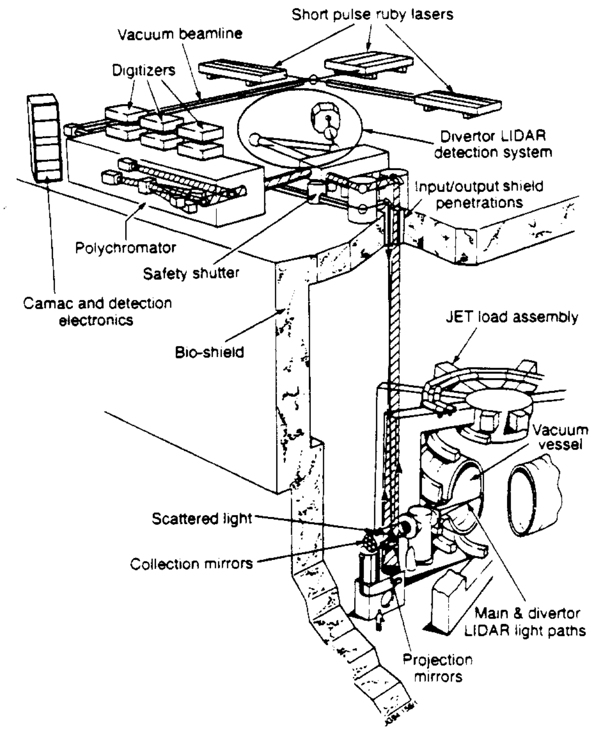


Fig.2: Diagnostics beyond the radiation barrier: LIDAR Thomson scattering

3. JET RESULTS AND DEVELOPMENT

A great variety of plasma and fusion physics issues have been already addressed and these have been reported in many international conferences and published in various scientific journals. This has been made possible by the inherent flexibility of the machine concept, which has also permitted far reaching engineering developments without significant modification to the original tokamak structure. Two major interventions, implemented in 1987-88 and in 1989-90 are mentioned here.

3.1. Upgrading of the Electromagnetic System

Operation in a wide range of plasma parameters and plasma currents (1 to 5MA) during the 1986 experimental campaign, has confirmed the nearly linear scaling of energy confinement time with plasma current, and plasma energy degradation with input power to the plasma. Moreover, it was possible to produce X-point configurations with plasma currents of up to 3MA in JET, thus establishing *High (H)-mode* plasmas. The results confirmed early findings with ASDEX in Europe and PBX in the USA, that, at a given plasma current, the energy confinement time was more than twice that achieved with plasma in the limiter configuration, i.e in the *Low (L)-mode* (Fig.3).

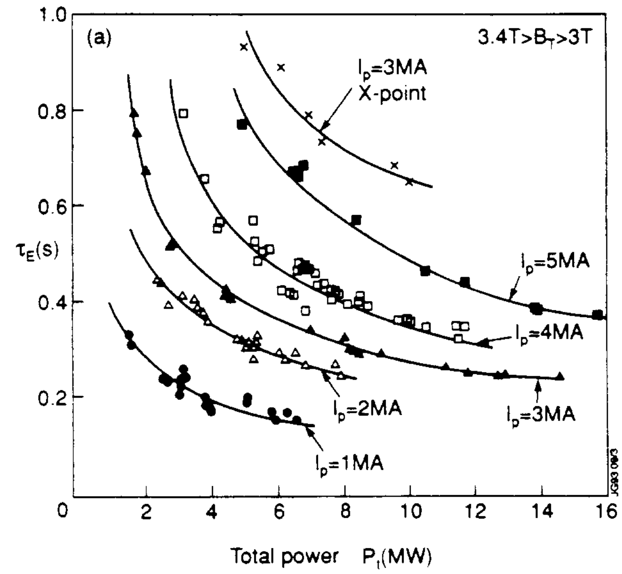


Fig.3: Energy confinement time versus heating power in limiter (Low-mode) and in X-point (High-mode) configurations

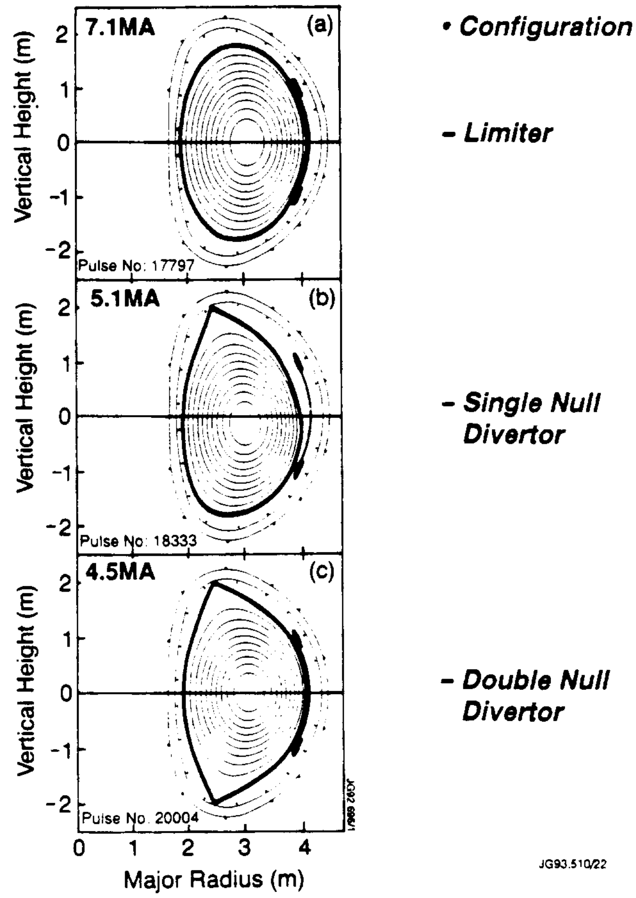


Fig.4: JET main operating configurations following upgrading of the electromagnetic system (1987-88)

During the following two years the JET capability was extended to 7MA in limiter configuration and to 5MA in X-point configuration, although only up to 3.5MA was the X-point inside the vacuum vessel (Fig.4). This was achieved by modifying and upgrading the poloidal power supply system and by re-assessing the ultimate electro-mechanical capability of the tokamak system, involving finite element modelling of vacuum vessel, magnet coils and mechanical structure [4, 5].

3.2. Passive Control of Impurities

This important upgrading would have not been sufficient on its own to reach high plasma performance. In fact, since early JET operation, it was clear that impurities were playing a key role in hampering progress in plasma performance. Therefore, the inconel vacuum vessel was progressively covered with graphite tiles supported by frequent wall carbonization. However, the plasma effective charge Z_{eff} was still too high ($Z_{eff} \geq 3$). It was then decided to use beryllium as a first wall material, due to its low atomic number (Z=4) and to its excellent capability of gettering oxygen (Fig.5). Graphite tiles were replaced with beryllium tiles fitted in critical areas of the vessel wall (in the X-point region). In addition,beryllium evaporation was performed between pulses, to cover other first wall materials.This was a major commitment since the use of beryllium can be a hazardous exercise. Therefore, JET had to acquire the appropriate technology, which involved the use of protective clothing for in-vessel work, continuous monitoring of beryllium levels and a well equipped beryllium laboratory.

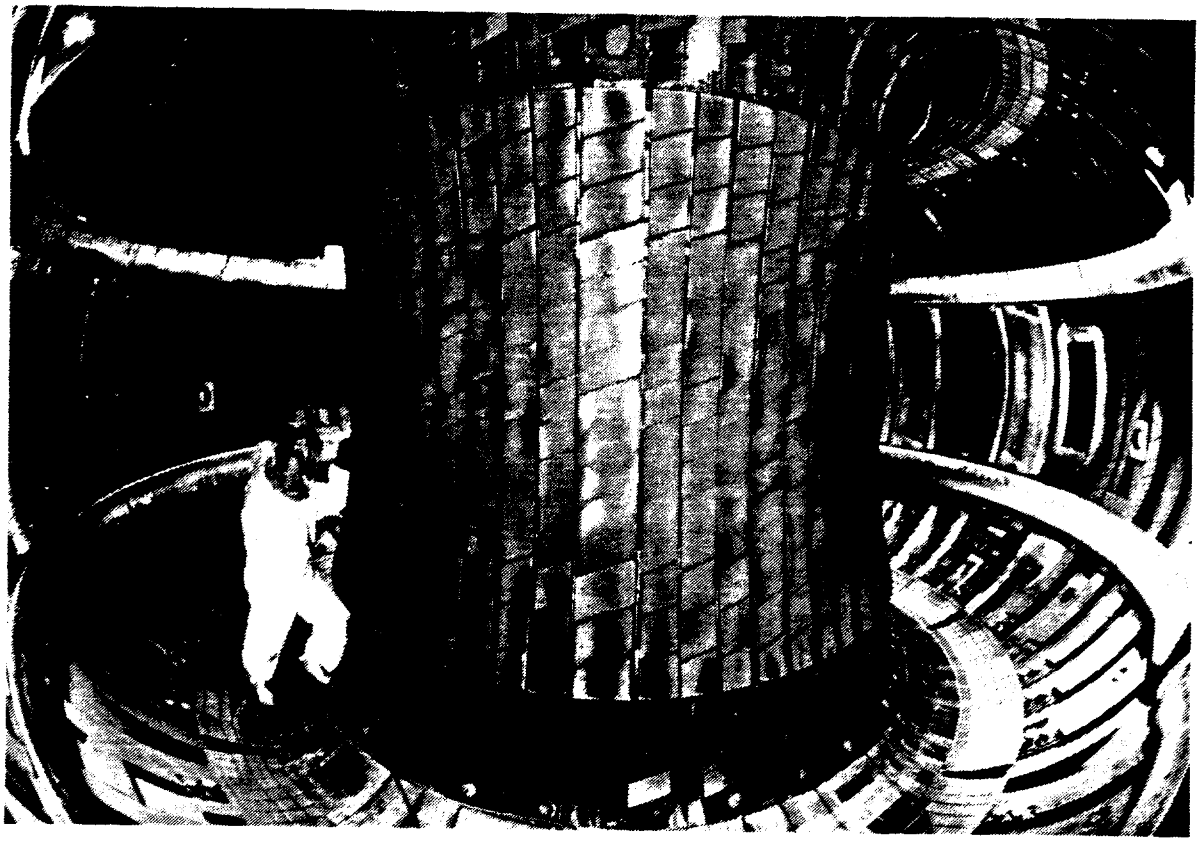


Fig.5: Vacuum vessel walls following first wall material upgrading, with graphite and beryllium tiles and beryllium evaporators

Dilution factors (deuteron density/electron density, n_D/n_e) exceeding 0.9 and Z_{eff} lower than 2 were achieved. As a consequence, the triple fusion product reached a value of $n_D\tau_E T_i$ $>0.9\times10^{20}\ m^{-3}keVs$, which is only a factor of 6 below ignition requirements, and an equivalent $Q_{DT}>1$ (breakeven) was also obtained [6].

This level of performance was considered suitable to conceive and perform the first ever controlled thermonuclear fusion D-T experiment towards the end of 1991 [7], with the production of 1.7MW of fusion power using a mixture of D(89%) and T(11%).

4. THE PUMPED DIVERTOR

The successful 1991-92 experimental campaign clearly indicated that passive control of impurities would allow high plasma performance to be maintained for only about one second, due to a combination of MHD instabilities and excessive production of impurities in the X-point region. Control of particles and power exhaust appeared mandatory for a fusion reactor.

This critical issue is being addressed in JET with the Mark I axisymmetric pumped divertor, which was installed during the 1992-93 shutdown and is now operational [8, 9, 10].

4.1. Divertor Design

The main components of the divertor are the four poloidal coils, the target plates and the cryo-pump (Fig.6):

- *The four coils* allow creation of a variety of magnetic configurations with an X-point at a suitable distance from the target plates, to sweep the strike-point so that the energy released from the plasma can be more evenly distributed and to control the magnetic flux expansion in the divertor chamber. The coils are fabricated in copper and cooled with freon 113. They are encased in a thin inconel casing, protected by thermal shields to reduce heat flux to the coils (JET operates with vessel temperatures of 250-300°C);

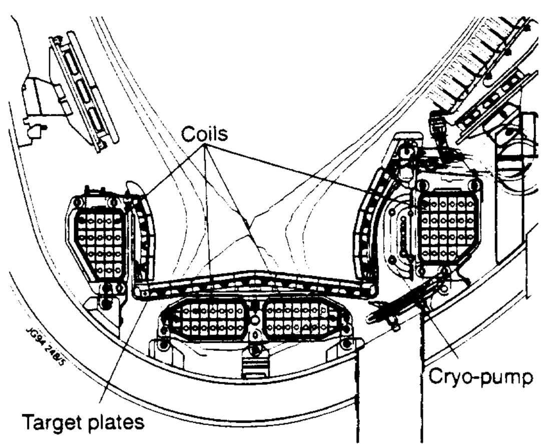


Fig.6: Main components of the pumped divertor: poloidal coils, target plates and toroidal cryo-pump

- *The target plates* are inertially cooled, and are arranged in a shallow W-shaped contour. They collect the power released from the plasma and consist of a water cooled inconel structure, which supports the carbon fibre composite (CFC) tiles, which are accurately shaped to avoid hot-spots. These tiles may be replaced later with beryllium tiles for comparison of performance;
- *The toroidal cryo-pump* allows control of the plasma density in the divertor region, when cold gas is injected to minimize the ionization of impurities. It is anchored on the outer divertor coil and consists of a water cooled baffle, a liquid nitrogen cooled copper-backed panel, an array of liquid helium cooled pipes and a chevron structure.

With the new divertor, the plasma volume is reduced by about 20% and the plasma shape also differs from the previous one. Therefore, the whole first wall had to be re-designed.

Since *the ICRF antennae* require adequate proximity to the plasma to achieve effective power coupling, new ones were built with non-symmetric shape and were installed at a distance of more than half metre from the vessel wall. The new antennae are now installed in four pairs. The grid of the *LHCD launcher* had to be re-shaped and the launcher re-positioned. *Limiters* are still required for ICRF antennae and wall protection and for plasma start-up. The old belt limiters would no longer match the new plasma cross-section. Twelve discrete poloidal limiters on the outer walls and 16 inner wall guard limiters were manufactured and installed, all covered by graphite tiles. Eight *saddle coils*, four at the top and four at the bottom of the vessel were also installed. These are used to control MHD instabilities (m=2, n=1 modes).

The complete refurbishing of the first wall required existing *diagnostics* to be modified and/or re-located and new diagnostics were installed to measure divertor plasma parameters.

4.2. Construction of the Divertor

The installation of the divertor and the re-construction of the first wall required an extended and well planned shutdown (March 1992-January 1994) with work organized in shifts, 16 to 24 hours per day for 6 to 7 days per week. This work was organized in three stages:

Stage 1 prepared the vacuum vessel for the new first wall installation. It comprised three main activities:

- Removal of all internal and necessary external components from the torus;
- Replacement of faulty toroidal coils No.4.2 (Octant No.4, position 2) and No. 4.3;

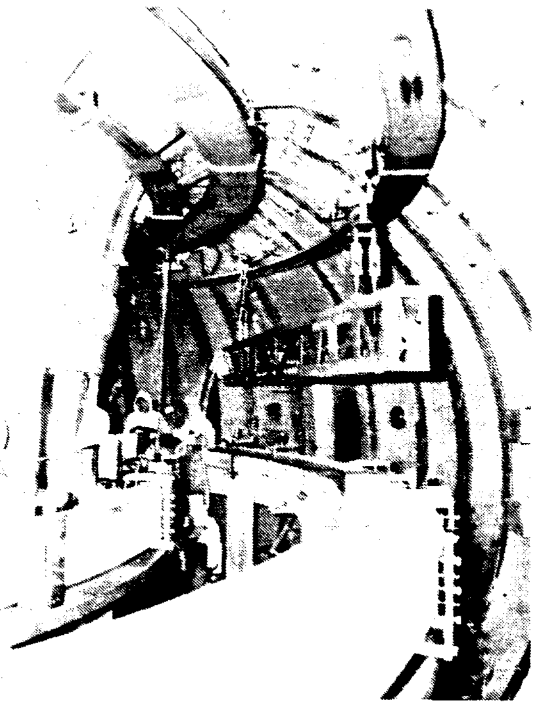


Fig.7: Construction of the four divertor coils inside the vacuum vessel

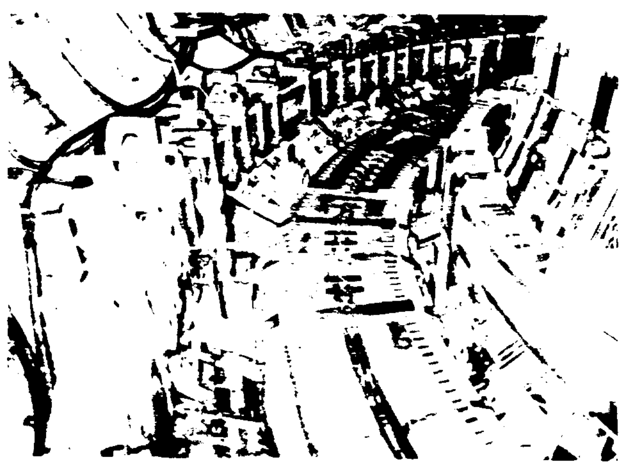


Fig.8: Divertor coils in their final position fully assembled inside the inconel casing

- Preparation for the fabrication of the divertor coil, including installation of jigs and full clean-up of the vessel, so that the coil construction work could be carried out without using full air-line suits. In fact, at the end of Stage 1 the radioactivity dose rate dropped from 85 to 5μSv/h and tritium and beryllium in-air concentrations were down to non-detectable levels.

Stage 2 was for the fabrication of the four poloidal coils inside the vacuum vessel, in a very restricted space. This was a completely new experience, requiring new methods of fabrication and new procedures. In fact, the work in factory was limited to the production of preformed and insulated hollow copper bars of 1800mm^2 cross-section and 1/2 or 1/3 circumference long; to the manufacture of the tools; and to the qualification of the brazing process. The bars were inserted into the vessel through a main horizontal port and positioned (Fig. 7). Each brazed joint had to be performed on curved bars, using transportable brazing tools, and mechanically and X-ray tested to high standards. The coil impregnation took place inside the final casing, a 3mm thick inconel structure (Fig 8).

Stage 3 started with a thorough clean up of the vessel by blasting with CO_2 pellets, to remove any surface contamination, possibly caused by the coil construction process. This stage progressed with the location of the coils in their final position with the required accuracy (~2mm radially). The divertor structure, water cooled inconel plates, graphite tiles and cryo-pump were then assembled. Finally the remaining components, poloidal and belt limiters, RF antennae, LH launcher and saddle coils were installed (Fig. 9).

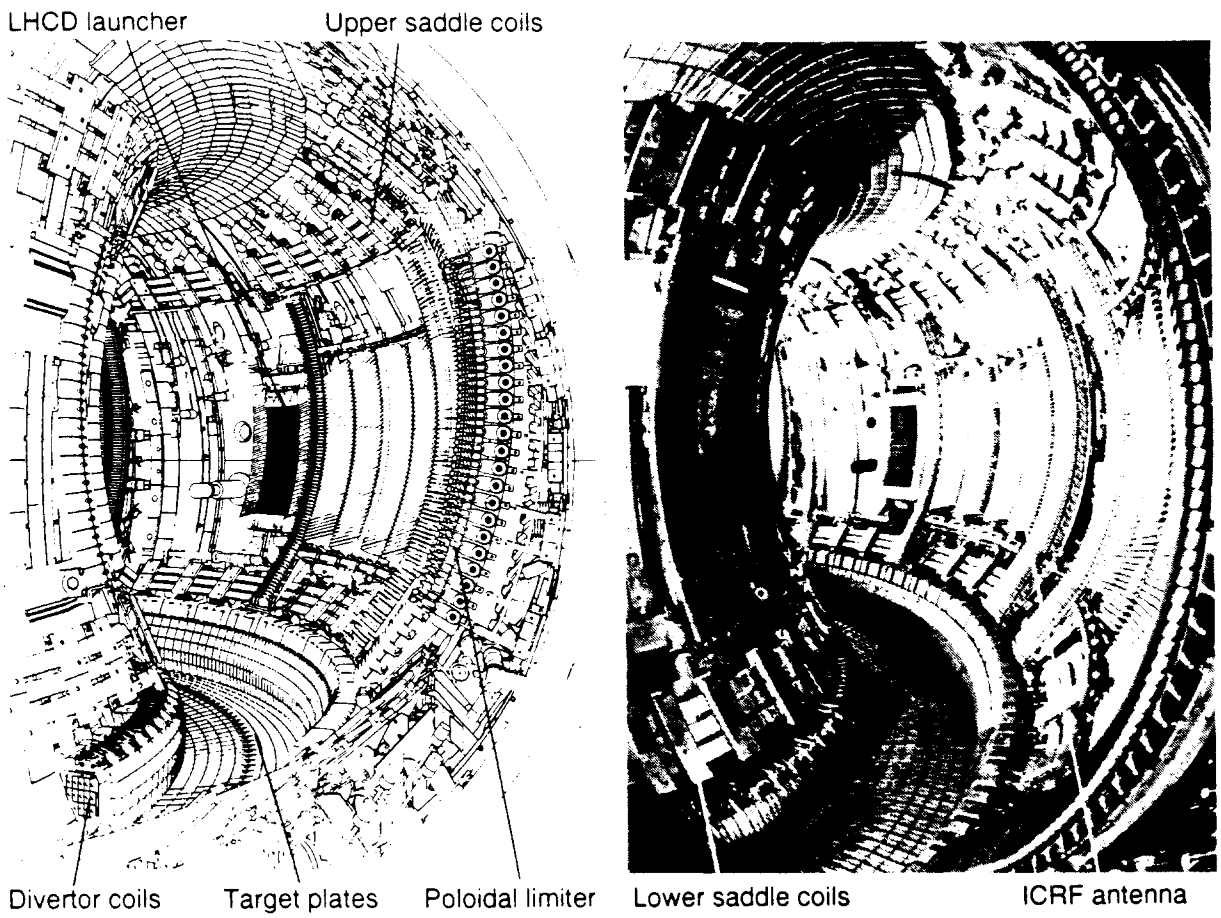


Fig.9: The new first wall of JET, showing main components

4.3 Additional Work

To make use of the pumped divertor and of the 'new' JET, several JET subsystems had to be modified and/or procured.

4.3.1 Power Supplies

- Four new Poloidal Divertor Field Amplifiers (PDFA), 500-650V, 40kA DC, supply the divertor coils individually and these include strike-point sweeping for enhanced divertor thermal load capability;
- The divertor plasmas are more vertically unstable than previously, thus requiring new Fast Radial Field Amplifiers (FRFA) with a rated DC output of 5kV, 5kA (or 10kV, 2.5kA) and a response time of 2ms over the full voltage range [11];
- The saddle coils are supplied by four units of a Disruption Feedback Amplifier System (DFAS), 1.5kV, 3kA output with a response of up to 10kHz [12];
- Using the full capability of the modified machine would lead to exceed the voltage drop limit at 400kV (2.5%), therefore the existing Reactive Power Compensation (RPC) system had to be enhanced from 150 to 200MVAR [13].

4.3.2 Cooling System

- Freon was selected as the coolant for the divertor coils (pressure of 14bar and mass flow of $20m^3/h$), with an independent circuit for each coil, including fluid recirculation to limit the coolant differential temperature to $\leq 20^oC$, thus limiting shear stresses in the fibre-epoxy electrical insulation [14].
- A major upgrading of the cryo-plant was required in order to supply the divertor pump, the LHCD system and the tritium handling plant, thereby leading to a total capacity of 1.3kW at 4.4^oC, 450 l/hr of liquefaction capacity.

4.3.3 Diagnostics

- New diagnostics had to be provided to study the plasma in the divertor configuration, such as a spectrometer to measure poloidal emission profiles from different impurity ionization stages, an interferometer to measure plasma electron density and temperatures (Fig.10), etc.
- A number of existing diagnostics had to be modified and/or re-located to retain access to the plasma following the installation of the divertor.
- Among the comprehensive JET diagnostic system are 14 diagnostics specially dedicated to the divertor, of which 11 have been already installed.

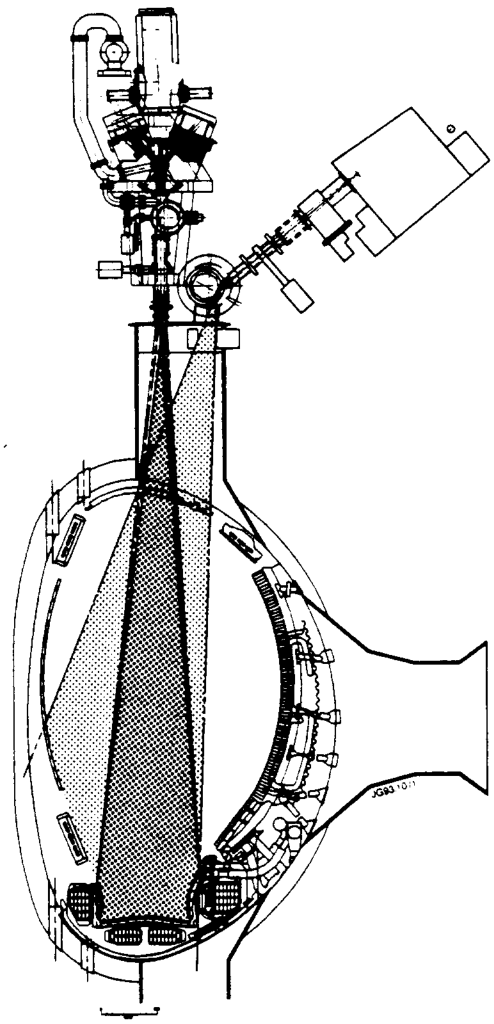


Fig.10: New spectrometer to measure parameters of impurities in the divertor region

4.3.4. Plasma Control

Due to the complexity of the plasma configurations with divertor [15], a new Plasma Position and Current Control (PPCC) system had to be conceived, designed and manufactured, for plasma current and shape control and for vertical stabilization of the plasma ring.

A significant part of this system is based on evaluating the plasma boundary in real time, by deriving information on the plasma position, on distances to the plasma-vessel wall and on the X-point location from local magnetic measurements. The software is 'intelligent' in that it allows Real-time selection of those plasma parameters which are controlled and those which are only monitored according to the plasma behaviour. The new vertical stabilization system is designed to stabilize plasma growth rates of up to $1000s^{-1}$ and the control algorithm is based on a proportional controller for feedback of the current moment derivative and a slower proportional-integral control of the power supply current. The proportional gain is decreased in real-time during the pulse.The implementation uses VME (Versa Module Eurocard) instrumentation and the 'engine' is based on DSP (Digital Signal Processing). The PPCC is now operating successfully in controlling plasma scenarios as required and is a major tool for the versatility of JET operations.

4.3.5 Coil Protection

An 'intelligent' Coil Protection System (CPS) was also provided, due to the greatly enhanced electromagnetic equatorial asymmetry of the machine. The main objective of CPS is to prevent coil operation outside design limits, to detect electrical faults and to protect the coils against electrical mechanical or thermal overstressing [16]. The protection system is fully digital, using three high performance Digital Signal Processors (DSP), and new dedicated transducers have been installed to make the protection independent of the control systems.

The protection algorithms can be classified into five categories:

- Conventional overcurrent and overvoltage protection for all circuits;
- Thermal stresses (e.g. I^2t) in all coils as a function of their initial temperatures;
- Mechanical stresses in the poloidal field (PF) system coils, including divertor coils, based on ampere-turn and flux sensors;
- Real-time integration of toroidal field and poloidal field circuit equations and comparison with the measured currents;
- Thermal model of the divertor coils, (since they are exposed to the vessel temperature of 250-300^{o}C) in order to keep the temperature of the epoxy under control, where inputs are the vessel and the coil casing temperatures, the coolant flow, and coolant inlet/outlet temperatures.

There is now scope to extend CPS to the protection of other major machine subsystems such as the vacuum vessel and the mechanical structure. CPS will then become MPS, Machine Protection System.

4.3.6 Work Unrelated to the Divertor

- The *Control and Data Acquisition System* (CODAS), based on a network of minicomputers, underwent major modification and upgrading. The 10 year old Norksdata computers were replaced with Sun computers using UNIX as an operating system and ETHERNET for communication. Signal conditioning systems and some data conversion systems use EUROCARD, while the rest of the instrumentation is based on CAMAC and high performance VME (Versa Module Eurocard) standards.
- The *Active Gas Handling System* (AGHS) has been completed and fully commissioned without tritium. It is capable of handling a daily throughput of up to 5 moles of T_2, 15 moles of D_2 and 150 moles of H_2 .

- Further development in *Remote Handling* took place during the shutdown. The Articulated Boom (in-vessel transporter) has been fitted with a new controller to enhance flexibility; the Telescopic Articulated Remote Mast (TARM) for ex-vessel activities has been commissioned; and the In Vessel Inspection System (IVIS) has been fitted with new viewing tubes. These tools were used for in-vessel work, and for the installation of the RF antennae in particular.
- *Pellet Injection* will be performed by a new centrifuge, capable of delivering long strings of 2-3mm pellets at velocities of up to 600ms^{-1} and a high speed single pellet injector (~4kms^{-1}) is in the latest phase of development.

5. MACHINE STATUS

5.1 New JET parameters

In Table I, the main parameters of JET with a pumped divertor are compared with those of previous experimental campaign. With the electromagnetic upgrading carried out in 1987-88, it had been possible to reach large plasma currents and plasma volumes with magnetic limiter. However the new JET should achieve an even larger plasma current in divertor configuration, at somewhat reduced plasma volume (~20%), due to the space required inside the vessel for the divertor structure (Fig.11 and Table I).

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	2.10	**1.75**
Plasma major radius, geometrical centre R_o (m)	2.96	2.96	**2.85**
Plasma Volume (m^3)	150$^{(a)}$ –	130$^{(a)}$ 105$^{(b)}$	– 85$^{(b)}$
Plasma aspect ratio, R_o/a	2.37	2.37	**3.0**
Plasma elongation, b/a	1.68	**1.9**	**1.85**
Toroidal magnetic field (at R_o), B_{TO}(T)	3.45	3.45	**3.6**
Flat top pulse length t(s)	10$^{(a)}$	10$^{(a)}$	10$^{(a)}$
Plasma current, I_p(MA)	4.8	**7.0$^{(a)}$** **5.0$^{(b)}$**	**6.0**
Transformer flux, ϕ (Wb)	34	**42**	**42**
Neutral Beam power, (MW)	15	**20**	**24**
Ion Cyclotron power, (MW)	15	**20**	**24**
Lower Hybrid power, (MW)	–	**5**	**10**

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

Table I: Evolution of JET design parameters

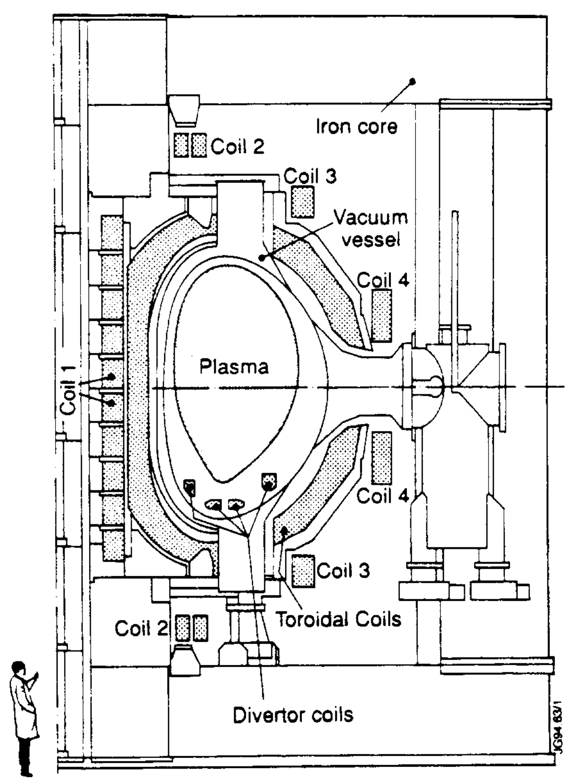


Fig.11: Cross-section of JET with pumped divertor

5.2. JET Facilities

JET has now a large variety of facilities available to tackle the key issues of the experimental programme to follow:

- The inertially cooled *pumped divertor*, with strike-point sweeping capability for long plasma pulses and high energy deposition on the target plates;
- A flexible set of *AC/DC converters*, supplied directly from the 400kV Grid, to permit a wide flexibility in plasma scenarios and performance;
- A *machine protection system*, that can calculate mechanical stresses on major components, thus allowing an expanded parameter operating space;
- *Machine and plasma control system* with a high degree of intelligence to produce and control the required plasma shapes and to actively control vertical instabilities;
- *Glow Discharge Cleaning* (GDC) with four graphite electrodes supplied at 5MHz.
- *Beryllium evaporation* by moving beyond the vessel wall four Be evaporators, heated by an internal graphite resistance;
- Up to 58MW of *injected power* into the plasma, namely:
  - Neutral Beam Injection (NBI), 16 PINI's (Plug in Neutral Injectors), 80kV-60A or 160kV-30A and up to 24MW of delivered power,
  - Radio Frequency (RF), 8 Ion Cyclotron Radio Frequency (ICRF) antennae assembled in pairs supplied through 16 tetrodes, 23-57MHz, and up to 24MW of delivered power,
  - Lower Hybrid Current Drive (LHCD), 24 klystrons, 3.7GHz, and up to 10MW of delivered power;
- *Fuel injection* by
  - Gas filling, puffing and density feedback;
  - Pellet Injection, multiple-pellet/pulse up to 0.6kms^{-1} and single pellet up to 4kms^{-1};
- *Diagnostics*, specially designed and/or modified for divertor plasma for measurement of all relevant plasma parameters in the divertor region with time and space resolution;
- *Active Gas Handling System* (*AGHS*), for D-T fuel injection, recovery and re-processing;
- *Remote Handling* facilities, including *IVIS* (In-Vessel Inspection System), to inspect the status of in-vessel components, when required by the development of the experimental campaign.

5.3. Operating Scenarios

In addition to the above facilities, flexibility in safely operating in a wide variety of plasma scenarios relies on *Operating Instructions* and on a *Plasma Protection System* to minimize the adverse effects of disruptions on the vacuum vessel, on the in-vessel components, and on experimental time. A proper plasma recovery following a disruption, may take several pulses and/or several hours of glow discharge cleaning and/or beryllium evaporation.

6. COMMISSIONING AND FIRST OPERATION

The awareness of having to deal with a completely new machine required a well prepared approach to the re-start of JET.

6.1. Readiness for JET Operations

The technical re-commissioning of the tokamak and of its major subsystems was prepared during the last six months of the shutdown by an 'ad-hoc' Group, which was given the following tasks:

- *Planning* the local commissioning with CODAS, the integrated commissioning, i.e. the subsystem commissioning on dummy loads and eventually commissioning from the central control room using the JET loads;
- *Machine active inspection*, based on well defined plans and procedures of inspection, following completion of the shutdown, in order to assure that power commissioning could proceed safely;
- *Machine protection and safety*, by verifying that all subsystems have adequate main and back up protection with particular attention paid to the newly installed equipment;
- *Operating Instructions*, setting the allowed machine operating parameters, following a careful analysis of the implication of the operating scenarios for the safety of the tokamak and its major subsystems.

6.2 Task Force for Re-start

A special task force was set up to take care of commissioning first without plasma and later with plasma. The key goal of this Task Force was to progressively bring all subsystems into operation and to produce a 2 to 3MA plasma, heated up to 10MW of additional power in PPCC control. The commissioning with plasma was performed in steps, i.e. gas breakdown, fast rise plasma, limiter plasma with several seconds of flat-top, divertor plasma with strike-point sweeping and finally 2 and 3MA divertor plasma heated up to 10MW with RF and NB injection.

The main problem encountered was with vertical stabilization because parasitic signal inputs to PPCC leading to FRFA delivering the maximum voltage (10kV) at the highest repetition frequency (~ 2kHz). This was corrected by modifying the internal control algorithm of the FRFA.

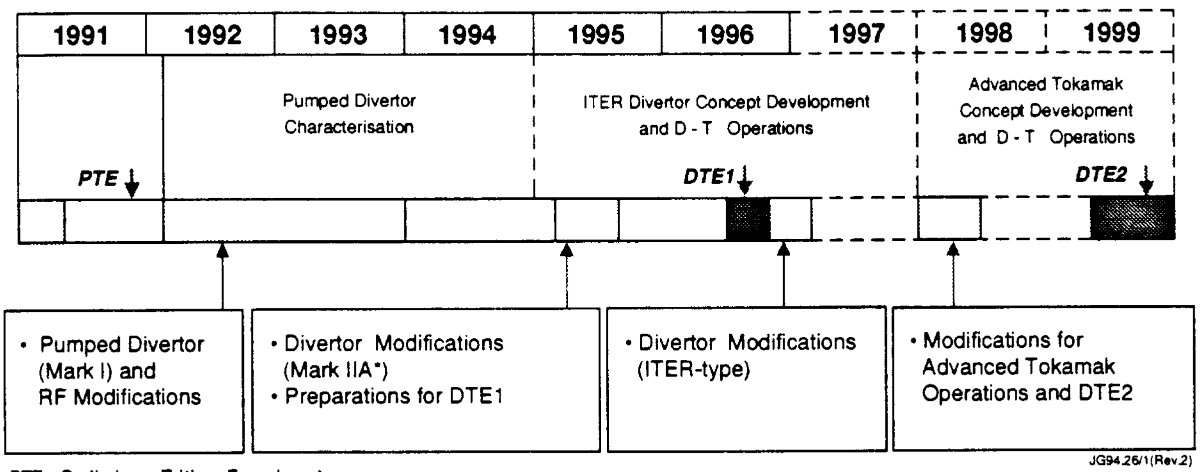
The physics experimental programme that started at the end of April 1994 included plasma discharges of up to 4MA with additional heating and X-point sweeping, and up to 20MW of combined heating delivered to the plasma [see Ref.2].

7. FUTURE DEVELOPMENT

The process is underway for extending JET beyond the scheduled end-date of 1996, to serve the needs of the European Fusion Programme, both as preparation inside the Team and in the administrative process necessary for obtaining all the required formal approvals.

7.1 JET Extension to 1999

Serious consideration is now given by JET's governing bodies to the extension of JET beyond 1996. The plan proposed by the JET Director is summarized in Table II.



PTE: Preliminary Tritium Experiment

DTE1: D-T Experiments with ≤ 2x10²⁰ neutrons in ≤ 4 months

DTE2: D-T Experiments with ≤ 5 x10²¹ neutrons in ≤12 months

* JET Council to decide in 1994 whether or not to install the Mark II Divertor

Table II: Plan for extension of JET to 1999

A significant extension of the JET programme has been evaluated and given high priority by the Joint Working Group set up by the FTSC and by the PC, and by the JET Scientific Council (JSC). A detailed proposal is now being considered by the JET Council for managerial approval. Its milestones are the installation of a Mark II divertor in 1995, a Deutrium-Tritium Experimental (DTE1) campaign at the end of 1996 followed by divertor modifications to test specific ITER concepts, and by a second D-T phase (DTE2) at the end of 1999.

As part of the JET extension assessment, the JSC has set up a Joint Group with the JET Team on the 'Technical Reliability of JET'. The JET machine design and performance, the power supplies, the control and protection systems and remote handling have been reviewed in great detail in relation to the present and future plasma scenarios envisaged. While the final report is due later in the year, the interim report notes that:...*the initial main conclusion of the Group is that only a small fraction of the life of JET's subsystems has been used.*

7.2 Divertor Development

The JET forward programme is focussed on divertor studies, progressively using configurations as close as possible to ITER requirements, since it is widely recognized that the power and particle exhaust may be the main obstacle remaining to be overcome for the development of a tokamak fusion reactor.

Although the Mark I divertor has already proven to work very satisfactorily, it has already been agreed that a more 'closed' divertor structure would lead to increased impurity and neutral particle retention, increased radiation and reduced power being conducted to the target plate [17]. A Mark IIA (Fig.12) divertor, which provides a closed configuration, is under construction. It would have a 'continuous target' design, be inertially-cooled, and may not require strike-point sweeping. The divertor structure would be no longer supported by the divertor coil casings. This would reduce the heat transfer to the divertor coils, which at present gives some concern for the fibre-epoxy insulation of the coils. Mark IIA is scheduled for installation in the second part of 1995.

For the proposed extension of JET, some preliminary consideration has been given to a gas box divertor Mark IIB(GB), which would consist of a large volume divertor closed at the entrance with a narrow baffle, below which the neutral particles can circulate freely throughout the divertor volume (see Fig.12). However, ITER design development may suggest more appropriate alternatives.

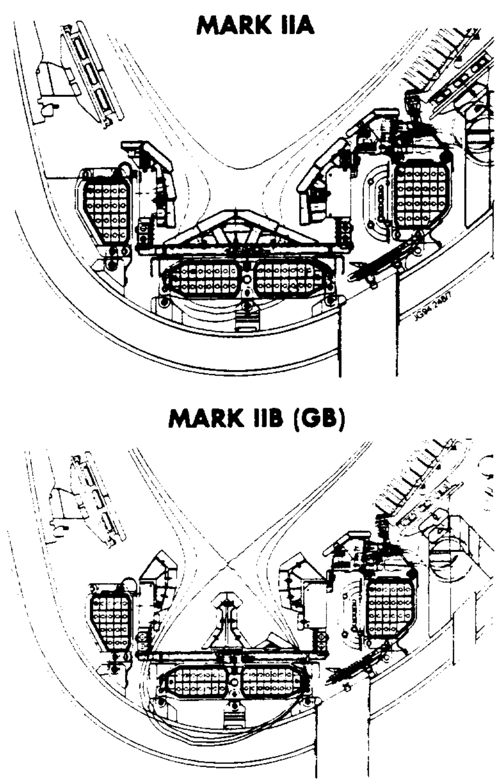


Fig.12: Schematics of Mark II and Mark IIB (GB) divertor configurations

7.3 Deuterium-Tritium Experiments

The objectives of DTE1 have been defined according to the following issues:

- demonstration of high fusion yield ($Q_{DT} \geq 1$);

- examination of H-mode threshold and ELM behaviour;
- long pulse, RF-heating scenarios;
- demonstration of reactor relevant remote handling techniques;
- demonstration of AGHS operation integrated with the JET machine.

All of these objectives can be accomplished with a total neutron budget of $2x10^{20}$ neutrons, which represents a small fraction of the total neutron fluence foreseen in the life-time of JET. However, it will take 18 months of cool down to allow manned work in the vessel. Therefore a remote handling plan is in preparation and is aimed at replacing the tile structure of the Mark IIA divertor, with a Mark IIB(GB), following the completion of DTE1.

Within the vessel, tile carriers will be handled and positioned by the Mascot IV manipulator mounted on the articulated boom transporter, using Octant No.5 port (Fig.13). Navigation and pre-positioning will be done automatically using *teach and repeat*. The required positioning of the boom, manipulator and cameras would be established using graphical modelling and

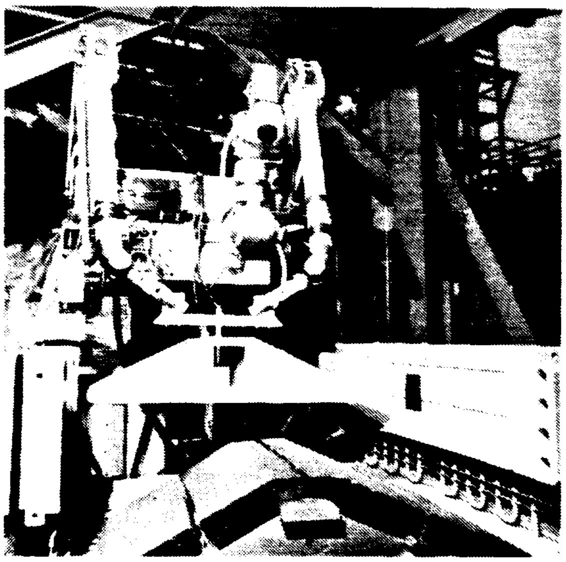


Fig.13: Mock-up tests for the remote removal of Mark IIA and for the installation of Mark IIB (GB) tile carrier modules

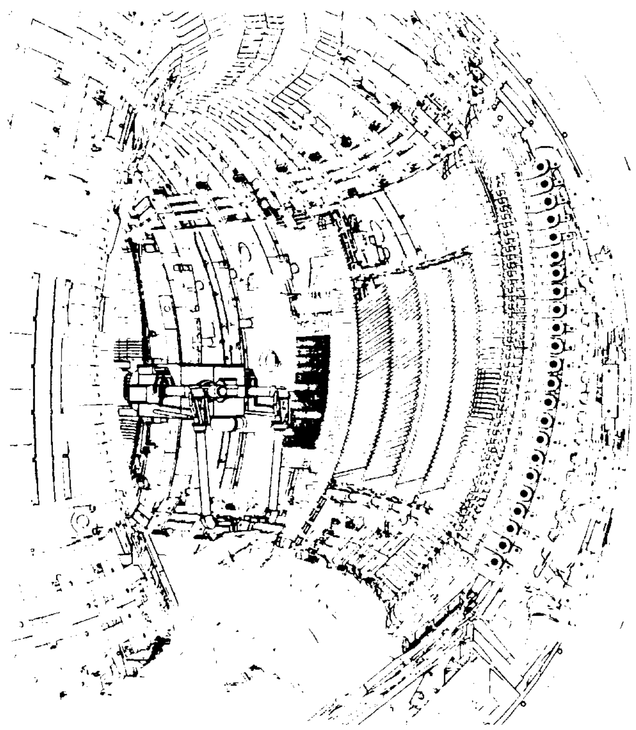


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Fig.14: Graphical modelling to establish access and 'teach'navigator file for remote replacement of Mark IIA tile carrier modules

verified in the mock-up facility. These positions would then be recorded in a teach file and repeated when doing the actual job (Fig.14). Bolting of the tile carriers would be performed by the manipulator, handling a ratchet wrench motorized for fast run in. A shorter version of the articulated boom would be used for transferring the tile carriers between the Torus Hall and the vessel through Octant No.1 port. For removal of the Mark IIA tile carrier there would be a similar sequence of events in reverse. This exercise will represent the first ever operation of its kind in a fusion device and will be of great interest to ITER.

To make the Deuterium-Tritium experimental campaign even more successful the powerful additional heating and current drive systems may allow exploration of advanced tokamak scenarios. Although this is not at present among the JET priorities, some work in this field would allow new tokamak concepts to be explored possibly leading to improved performance in D-T operation, thus making the study of the alpha-particle physics more substantial.

8. CONCLUSIONS

- Eleven years of JET operation have given essential contributions to the development of fusion research. This has been possible due to JET's original design and built-in flexibility, that has allowed the tokamak and JET major subsystems to be upgraded and modified to focus the experimental programme on the key reactor relevant physics issues;
- The ITER design concept and its parameters, calling for large plasma volumes and current, have been greatly influenced by the JET results, including single-null divertor and beryllium as first wall material;
- The JET divertor programme addresses the key issue of the pumped divertor for the control of power and particle exhaust, again with a flexible design, which allows a number of ITER relevant divertor concepts to be experimentally assessed;
- The JET forward programme, particularly if the extension of JET beyond 1996 is decided shortly, should be focussed to support the detailed design of the first prototype fusion reactor, ITER;
- Since JET has been originally designed for extended D-T operation, reactor relevant remote handling tools and a fuel reprocessing plant are now available, allowing plans for a step wise approach to D-T experiments with DTE1 and DTE2;
- The readily available additional heating and current drive facilities will facilitate establishment of high neutron yield D-T plasmas, for significant and ITER essential studies of alpha-particle physics.

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