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JET DESIGN, CONSTRUCTION AND PERFORMANCE

by

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1. INTRODUCTION

The Joint European Torus (JET) is the largest Tokamak experiment operating in the world. Its design was started in September 1973, by a team of engineers and physicists, assembled from throughout Europe at Culham, near Abingdon, United Kingdom. After the design work was completed, approval for construction was given in 1978 and the JET Joint Undertaking was established and the Construction Phase started in June 1978. The basic machine and its major subsystem were completed on time and within costs in June 1983, when the Operation and Development Phase began.

During the past six years major subsystems have been made progressively available, (such as the Neutral Beam and the Radio Frequency heating devices), machine performance was enhanced above design values and a highly succesful experimental programme has established JET as the world's leading fusion experiment.

The objective of the JET experiment is to obtain and study plasmas of hydrogen isotopes in conditions and with dimensions approaching those needed for a fusion reactor. In so doing, JET should prove scientific feasibility of controlled thermonuclear fusion and should secure the data-base for the design of NET (Next European Torus), the machine which should extend performance to full reactor grade plasma and prove the soundness of engineering and technology for the design and construction of a prototype fusion reactor, DEMO, (Demonstration Reactor), due to produce several hundred megawatt of electricity in a power station.

2. THE JET TOKAMAK

Since its discovery at the Kurchatov Institute in Moscow in the mid-1960's, the Tokamak has proven to be the most effective among the systems considered in magnetic confinement fusion for plasma production, confinement, heating and control. This is the reason why the Fusion Programmes in Europe and throughout the world, are mainly concerned with Tokamak systems.

It is the main objective of JET to operate with plasmas in near-reactor conditions. To this aim, four areas of research are involved.[1]

a) Plasma processes and scaling laws in regimes close to those needed in a thermonuclear reactor.
b) Methods of heating the plasma to temperatures approaching those required in a reactor by ohmic heating, neutral beam injection and radio frequency heating.
c) Control of plasma-wall interaction and influx of impurities in reactor relevant conditions.
d) Alpha particles production, confinement and subsequent plasma heating, produced as a result of fusion between deuterium and tritium atoms.

In order to minimise nuclear activation of the tokamak structures, the first three areas of study can be approached by using hydrogen, deuterium or helium plasmas, while the last one requires the use the thermonuclear mixture deuterium and tritium.

A tokamak is inherently a device that does not operate continuously, due to the fact that a DC plasma current must be maintained by transformer action. The operating time (pulse length) limited by mechanical stresses in the transformer winding, associated with the value of the pre-magnetising current, can however by quite long (several seconds in present experiments up to many minutes in a thermonuclear reactor) because the high electrical conductivity of the plasma at high temperature.

The main parameters of a Tokamak are the geometrical dimensions of the plasma ring, i.e. the major radius of the torus R_o, the radius (minor radius) of the plasma cross section a, the value of the toroidal field B_{To} (at R_o) and the plasma current I_p (or its associated poloidal magnetic field B_p).

In order to avoid the plasma ring becoming macroscopically unstable, the above four parameters cannot be chosen independently, and they must satisfy the so called Kruskal-Shafranov relation $B_{To}a/B_pR_o = q > 1$, where q is called stability safety factor and $R_o/a = A$ is called aspect ratio of the tokamak. The principle of the tokamak plasma confinement system is shown in Fig. 1.

The JET facility comprises the following major subsystems: the Tokamak Assembly; the Additional Heating Systems; the Power Supplies; the Plasma Diagnostics; the Control and Data Acquisition System. Moreover, for deuterium-tritium operation, Remote Handling Tools and a Tritium Plant are required.

In the following, the above systems will be described with the main emphasis on the Tokamak apparatus.

3. KEY DESIGN FEATURES OF THE JET TOKAMAK

An overall schematic view of the JET Machine is shown in Fig. 2, and a cross section is shown in Fig. 3. The principal parameters of JET are given in Table I. These illustrate the key features of the JET Tokamak.[1], [2], [3].

The toroidal coils, the vacuum vessel and the plasma exhibit an elongated ('D' shape) cross-section. This original choice made in 1973 was initially controversial, but has proven one of the most valuable assets for JET performance. There are no proposed tokamaks, big or small, being designed today, without the above features. The 'D' shape of the toroidal coils was chosen to reduce mechanical stresses. A coil in a magnetic field is subject to the stresses associated with the electromagnetic interaction between the coil current and its associated magnetic field. In a toroidal magnetic field, the coil tends to assume such a D-cross sectional shape, due to the magnetic field decreasing with major radius across the coil cross section. Therefore if a D shape is chosen, there is no bending moment in the plane of the coils and the only force is the centripetal force which pushes the coil toward the centre of the magnet. A vacuum vessel of similar shape allows the toroidal field volume used by the plasma to be maximized, as it permits a larger plasma current to be accommodated for a given B_{To} and q.

Plasma performance strongly depends on geometric dimension, magnetic fields and plasma currents. The choice of a large plasma volume, ~ 150 m^3, (larger by a factor of 2 or more than any other tokamak in the world), has been compromised with a relatively modest toroidal field (due to the trend of the magnetic field in a toroidal magnet, stress limitation prevents very high magnetic fields to be used in large volumes). It was also decided that a large plasma current should be made available and a value of 4.8 MA was chosen, more than five times larger than the current reached in any other tokamak before.

The D shape has allowed the elimination of a coil casing. Thus, thermomechanical stresses associated with the thermal expansion of the coils due to joule dissipation are eliminated, since the coils are free to expand in both the vertical and the radial directions. The only stress on the coils is therefore the shear stress on the electrical insulation associated with the overturning moment due to the interaction of the toroidal field current with the intersecting magnetic fields generated by the poloidal coil and by the plasma current. A mechanical structure to withstand this moment is therefore required.

The elimination of two out of three sources of stress on the toroidal coils in a tokamak, allows overall reduced stress, thus increasing reliability and access to the plasma (in between coils) for vacuum pumping, plasma additional heating and diagnostics.

Pulsed operation allows the machine to be designed in such a way that during the pulse the machine is in practice adiabatically heated and the cooling takes place between pulses, at a reduced cooling plant power depending on the ratio between operating time and interpulse time. In JET (as in any other experimental tokamaks), this time ratio is smaller than unity while in a reactor it should obviously be greater than unity.

For JET, an operating time of 20s at full performance followed by a pause of 600s has been selected. This has allowed the choice of copper coils, water cooled between pulses and energy storage has been required for the power supplies (however, for a reactor, superconducting coils may be used).

Since a tokamak is a pulsed machine, mechanical fatigue has been a major concern in the JET design.

4. .THE JET MACHINE [4]

The main components of a tokamak assembly are the vacuum vessel, the toroidal magnetic field coils, the poloidal magnetic field coils the iron core (optional) and the mechanical structure.

JET has a modular design with eight identical octants, each one composed of one eighth of the vacuum vessel, four toroidal coils (out of 32), and two-sixteenths of the mechanical structure. This modular approach can be also considered as a first step toward a fusion reactor design concept and it facilitates the replacement of one octant, when one of its components develops a serious fault.

4.1 Vacuum Vessel

The purpose of the Vacuum Vessel is to provide a clean environment for the plasma. Therefore, prior to filling with the hydrogen isotope gas, a vacuum level of up to 10^{-9} mbar is established in the vessel (at room temperature). In addition, the vessel inner wall represents a physical barrier to the plasma: Therefore, the vessel first wall must be designed both to maintain the plasma purity and to dissipate the energy released from the plasma (plasma radiation and transport losses). Finally, the vessel has been designed to eventually contain radioactive gas (tritium), and to cope with the high rate of neutron release (up to 10^{19} n/s) during the D-T operation phase.

Following JET modular design, the vacuum vessel is constructed by eight identical octants. Each octant consists of 5 rigid sectors, joined together by bellows sections (see Fig.4). The function of the bellows is two-fold: it provides a sufficiently resistive path for the current induced by the primary winding of the tokamak transformer (the vessel is electrically in parallel with the plasma) and it allows flexibility in joining the octants by lip welding on the inner wall (the U-joints are designed to allow up to 2-3 octant replacements during the life of the machine).

The vessel is double walled with stiffening ribs between the inner and outer structure, providing high mechanical rigidity of the octant and strength at reduced weight. In fact, the vessel withstands two types of forces: those due to the external atmospheric pressure and those associated with the current induced in the vessel during tokamak operation (up to 2,000 tonnes) when the plama current suddenly disappears in 10-100 ms (plasma disruptions).

The vessel interspace is designed to allow gas to flow, for controlling the vessel temperature (particularly the inner wall temperature) at the required level for operation (usually 300° C). However, the interspace has also an additional function: in cases of vacuum leaks, it can be pumped and the leak can immediately be controlled, even before repair. Finally, this double containment will be particularly suitable for D-T operation. The vessel material is Inconel (for the rigid sectors Nicrofer 7216LC and for the bellows Inconel 625), due to its mechanical properties and electrical resistivity. The volume of the vacuum chamber is ~ 200 m^3, with maximum dimensions of 4.3 m in height and 2.8 m in width.

The present configuration of the vacuum chamber is shown in Fig.5. Several features can be seen: the two water-cooled belt limiters to dissipate the energy released from the plasma; some of the eight radio frequency antennae for plasma heating; and graphite tiles covering 50% of the vessel wall (including the belt limiter, octant joints, inner wall, top and bottom). Carbon is a relatively low Z material (Z = 6), thus it allows overall plasma purity to be improved compared with an Inconel first wall. There are plans to progressively replace graphite with beryllium (Z =4) and preliminary experiments with beryllium evaporation and subsequent deposition on the vessel walls have shown substantially improved plasma performance due to the high plasma purity and to a reduction of plasma dilution (fewer non-hydrogen ions per unit volume), compared with graphite (carbon) walls.

The main parameters of the vacuum vessel are shown in Table II.

4.2 Toroidal Magnetic Field Coils

The purpose of the toroidal field coils is to produce the toroidal magnetic field which stabilize the plasma ring. The choice of dividing the magnet into 32 coils is a compromise between the need to access the plasma for vacuum pumping, additional heating and diagnostic equipment and the need to restrict the ripple of the magnetic field in the toroidal direction at the plasma edge.

Following the JET modular design, four coils are part of an octant. Each coil is composed of two pancakes each of twelve turns. The conductor is made of copper of 3900 mm^2 cross section with two channels for water cooling. To fit the coils within the chosen tokamak aspect ratio, the toroidal field coil are wedged on the inner side. Each turn is cooled separately and in the same direction to avoid temperature gradients between turns. Due to its size, only lengths up to 15 m of conductor could be extruded and this required a large number (approximately 1500) of brazing joints for the whole magnet.

Glass epoxy resin insulation was used and the copper bars were sand-blasted to improve mechanical attachment between copper and epoxy. The dimensions and the key features of a coil are shown in Fig. 6, and the coil parameters are summarised in Table III.

4.3 Mechanical Structure

The function of the mechanical structure is to withstand the forces to which the toroidal magnet coils are subject. Due to the JET design choice (see section 3), these forces are only the torsional forces generated by the interaction between the current flowing into the toroidal coils and the poloidal field. In the case of the sudden transfer of plasma to the vacuum vessel walls and rapid plasma current decay (i.e. a disruption), a torque up to 30,000 tonne m can be generated.

The structure has been designed as a metallic shell enclosing the toroidal field coils. The main components of the mechanical structure are shown in Fig. 7

According to the JET modular design, the external shell is split into eight octants, made up of an upper and a lower section. Only the external shell must resist the torsional load and therefore the inner part (inner cylinder) is thin, with the function of keeping the toroidal field coil straight. The high shear stresses at the shell octant joints are taken up by bolts and shear dowels. The collars and rings at the top and bottom of the shell, are to transfer the stiffness against torsional forces of the shell toward the centre of the machine. External shell and rings are made up of austenitic nodular cast iron, particularly suitable for mechanical fatigue operating conditions. The collar material is ferrite steel since it is part of the magnetic circuit. To reduce eddy currents in the vessel, octant and sectional joints are electrically isolated with fibre epoxy layers.

The toroidal field coils are only laterally supported by the mechanical structure, thus allowing vertical and radial expansion during temperature excursions (from room temperature to about 80°C) associated with coil heating during the pulse.

4.4 Poloidal Magnetic Field System

The duty of the poloidal field coil system is to produce and maintain the plasma current and to control plasma position and shape (see Fig. 3). The first function is performed by the P1 coils, which play the role of the primary windings of a transformer: by generating a high voltage across P1, an electric field is produced in the vessel, which breakdowns the gas, making it electrically conducting (i.e. a plasma), and thus allowing the secondary current (plasma current) to flow.

The iron core, although highly saturated in the centre, (due to lack of space and by choice of dimensions), allows a reduction in magnetizing currents by about 20% and also greatly reduces the magnetic fields outside the machine assembly (this is important for not interfering with diagnostic and neutral injection additional heating systems).

The position and shape of the plasma is controlled by the outer coil P2, P3 P4. The vertical field is produced mainly by the P4 coils, and counteracts the natural tendency of the plasma to expand. The radial field is produced by a combination of turns in the coils P2 and P3, and maintains the plasma vertical position relative to the equatorial plane of the machine. With an elongated plasma cross-section and an iron core, the JET plasma is vertically unstable: a fast feedback position control system is therefore essential. Other P2, P3 turns are used to control the shape of the plasma.

The poloidal coils are also made of hollow copper conductors, which are water-cooled and isolated with epoxy resin. A large flexibility in accessing coils and coil turns, facilitates the achievement of various plasma configurations.

The main parameters of the poloidal field coils system are given in Table IV.

4.5 Tokamak Assembly

The key stages of the sequence of machine assembly are summarised in the following paragraphs.
The bottom arms and vertical limbs of the iron core were first installed in the Torus Hall, with the bottom ring and collar of the mechanical structure. The bottom P2 and P3 coils are also located in their 'parking' position (see Fig. 8).

The machine octants were pre-assembled in the Assembly Hall. Each pair of toroidal field coils were placed in position inside one-sixteenth of the outer shell of the mechanical structure and placed on the octant assembly jig with one vacuum vessel octant . The vessel outer shell was thermally insulated to prevent heating the toroidal field coils when the vessel is at high temperature (see Fig. 9)

On completion of an octant, it was transferred to the Torus Hall and located in position (see Fig. 10). When all machine octants were in position the outer poloidal field coils were installed in their final positions, the iron core completed and the P1 coil was also installed in the centre of the machine. Finally, the vacuum vessel octants were welded together inside the vessel. The belt limiters, the RF antennae and the graphite tiles were also installed.

A recent view of the machine is shown in Fig. 11. Only the iron core, the (two) neutral beam systems, the RF antennae transmission lines and some plasma diagnostics are visible.

4.6 Machine Auxiliary Systems

The main machine auxilliary systems are the following:

- The vacuum system consists of turbomolecular pumps (3500 ℓs^{-1}), installed in the low magnetic field region (< 0.01T) by means of two large vessel extension ducts. They are backed by a system of roughing pumps;

- The baking system uses either CO_2 or He^4 as heat carrying gas, which is recirculated by radial turbocompressor and heated (or cooled) through a tube bundle heat exchanger;

- Wall conditioning (to remove wall impurities) is achieved by glow discharge cleaning: it consists of two graphite electrodes situated inside the vessel, coupled to an RF transmitter, where the vessel plays the role of the cathode of the discharge. The impurities are removed through chemical reactions of impurities with atomic hydrogen which lead to volatile components removed by the vacuum system;

- The gas introduction system allows the introduction of high purity gas, which is transformed into plasma in the machine. Various gases are used, including hydrogen and helium isotopes. The gas can be introduced as a puff (to first establish the discharge) and continuously, at the chosen rate. A plasma density feedback system actually controls the gas flow rate to provide a desired value of density.

- Machine systems apart from the vessel (e.g. toroidal and poloidal coil, belt limiters and RF antennae), are water cooled by demineralised water to site water heat exchangers with 50 MW cooling towers. The water cooling system services other major subsystems such as neutral beam injectors, power supplies etc.

5 TOKAMAK MAJOR SUBSYSTEMS

5.1 Site Layout

The JET site occupies an area of about 30.000 m^2 (see Fig. 12). The Torus Hall and the Assembly Hall occupy less that 10% of the area covered by JET facilities: Most of the site is required for the outdoor power supplies and for the power supply buildings.

From North to South, one can see the incoming 400 kV, the 400 kV/33 kV substation with the switchgear building (J4), the outdoor AC-DC power supplies for the additional heating system, the flywheel generators buildings (J3), the static unit building (J4), the North Wing with the DC power modulation equipment and the R.F. generators, the Assembly Hall and the Torus Hall; not shown are the West Wing with the Tokamak auxiliaries, the South Wing for the diagnostic equipment and finally the Control Building (J2), from where the machine is operated and further south the main office buildings indicated. The dominant building is the Torus Hall, which has been designed for D-T operation. The concrete walls are 2.8 m thick and there is an Access Cell between the Torus Hall and the Assembly Hall.

When JET starts operation with the thermonuclear fuel mixture (D-T), no access to the Torus Hall will be permitted, due to the neutron activation of the machine structures. All operations in the Torus Hall will be performed by remote handling techniques.

5.2 Additional Heating and Fuelling

Plasma in JET is heated by Ohmic Heating (i.e. by joule dissipation of the plasma current in the plasma itself), by Neutral Beams and by Radio Frequency systems.

Neutral Injectors [5] allow energy to be transferred to the plasma by means of beams of hydrogen isotope atoms at high energy. A neutral injector consists of: an ion source, where hydrogen isotope ions are accelerated by means of electric fields; and a neutralizer where, by charge exchange, a beam of neutral particles is produced, so as to allow penetration through the Tokamak's magnetic fields and reach the plasma (see Fig. 13). The beams are required to dissipate their energy close to the plasma centre. It has been evaluated for the JET dimensions and parameters (plasma densities ~ 5 - 10 x 10^{19} m^{-3} and temperatures > 5 keV), about 80 keV per nucleon are required. Therefore, 80 kV and 140 kV-160 kV are the potentials applied to the accelerating grid for hydrogen and deuterium beams respectively (160 kV will be used for tritium beams). Another important component of neutral injectors are the magnets required to deflect the non neutralised part of the ion beams onto ion dumps where their energy is dissipated and removed by cooling.

In JET there are two neutral beam assemblies, each composed of eight injectors. Together, these allow 20 MW of energy to be transferred to the plasma at 80 keV and 16 MW at 140 keV (see Fig. 11) These penetrate the plasma through Octant Nos. 8 and 4 vessel ports through specially designed rotary valves that can isolate the neutral injection system from the vacuum vessel.

For Radio Frequency heating [6], the choice was made to use Ion Cyclotron Resonant Frequencies (ICRF) in the range 25 - 55 MHz. Each ICRF unit consists of a radio frequency generator, whose main components are a pair of 2 MW tetrodes, the transmission line and the antenna which is located inside the vacuum vessel facing the plasma.

In JET there are eight antennae (see Fig. 5) each one supplied by two R.F. generators via two transmission lines: therefore each antenna is supplied with 4 MW of RF power. The voltage required is up to 60 kV DC. Taking into account coupling efficiency, up to 20 MW of R.F. power can be coupled to the plasma.

Plasma fuelling and density profile modification during a plasma pulse can be performed by means of a Pellet Injection system [7] (gas introduction and neutral beams also contribute). The present facility allows the firing into the plasma of deuterium pellets of 2.7, 4 and 6 mm diameters at a frequency of 5 Hz and at a speed of 1.5 km/s. Although this velocity is quite spectacular, it is not sufficient to deposit pellets at the plasma centre at high plasma temperatures, (as the pellet is vaporised and ionised before reaching the centre). An enhanced system is under development aimed at speeds of 5 km/s, which is the minimum required to fuel high performance plasmas expected in future machines.

5.3 Power Supplies [8], [9]

JET operates in pulses lasting for several seconds (typically 10 - 20 s, but pulses up to 30 s have been achieved at somewhat reduced performance); each pulse is followed by a 10 minute cooling interval.

There are basically four pulsed loads to be supplied by DC power: the toroidal magnet; the ohmic heating; the plasma control; and the heating systems (see Fig. 14). The energy required for a pulse approaches 10,000 MJ while the peak power is ~ 1000 MW, as the power peak demand for each load does not occur at the same time. In fact, the installed AC/DC rectifier power exceeds 1500 MVA. The proximity (~ half-mile) of the 400 kV British Grid with a high short circuit capacity (15-35 GVA) makes it possible for JET to satisfy more than 50% of its power and energy demand, directly from the grid, while the rest is supplied by two flywheel generators.

Extensive studies, conducted in co-operation with the U.K. Central Electricity Generating Board (CEGB), have shown that the effect of the disturbances, produced by the JET loads on the grid (voltage drops, reactive power swings, power steps, harmonic distortions) are less severe than anticipated. Therefore it has been possible to dedicate each generator to a load not requiring fast control (toroidal magnet and transformer coils), thus using cheap diodes instead of expensive thyristors as the main components of the rectifier system. All the thyristor units are instead directly fed by the grid through conventional transformers.

The Generator Hall is shown in Fig.15. The 8.5 MW asynchronous motors, which allows the energy to be stored in the generator's flywheel in between pulses are visible above the generator pits. Three 300 MVA (pulse rate), 400 kV/33 kV transformers supply three separate busbars, from where rectifier transformers are energised via 33 kV SF6 breakers. The toroidal magnet is supplied by a flywheel generator diode convertor system and by two x 114 MW thyristor rectifier units at total voltages up to 9 kV.

The primary windings of the tokamak transformer is supplied by the second generator via the Ohmic Heating Circuit, which modulate the voltage between 24 kV (for plasma breakdown) and 1-2 kV for the control of the plasma current. The key component of the ohmic heating circuit are the two DC current breakers, which, by opening a 40 kA DC current across resistors, generate the transient high voltage required for gas breakdown and plasma formation.

The radial plasma control coils and the plasma shaping coils are supplied by a system of identical Thyristor Rectifiers Units, (1.3 kV x 40 kA each) which can be configured according to needs. The vertical plasma control coils are supplied by a Back to Back Thyristor Unit ($\pm$ 2.3 kV x $\pm$ 3 kA). The main parameters of the magnet power supplies are given in Table V.

The plasma position and current control systems use magnetic probes and flux loops as sensors and analog electronics for measurement of position. Digital controllers are now used except for the plasma vertical stabilisation. The control voltages are transmitted to the relevant power supplies. As a result, the plasma current, plasma position and cross-sectional shape can be controlled.

Both neutral injectors and radio frequency generators are supplied by controlled rectifier units of similar design. Due to the high DC voltage required (up to 160 kV), rectification with thyristors would prove most expensive. An original scheme has been devised with a 'cascade' of two transformers, the first reduces the voltage from 33 kV down to ~ 1 kV where a thyristor bridge system ('star point controller') at low voltage controls the input voltage to the next transformer, thus allowing diode bridges to be installed at its secondary high voltage side. Frequent transient short circuits occur in the neutral beam ion sources that must be cleared in less that 1 ms, and the voltage should be re-applied (to allow the pulse to continue) in less than 20 ms. This function is performed by the protection unit, the main component of which is a tetrode which can switch-off with a delay time of 10µs. Auxiliary power supplies are provided for the other circuits of the injector, i.e. filament, arc, gradient grid and magnet. The power is taken to the injectors by SF6 filled flexible transmission lines (see Fig.16).

The neutral beam power supplies can be configured for 140-160 kV, by connecting in 'series' two 80 kV units where provision was made in the design for such high voltage configuration.

5.4 Diagnostics [10]

A set of integrated plasma measurement systems (diagnostics) is necessary to learn how to proceed toward fusion conditions and to study plasma in such conditions. About 40 diagnostic devices are installed in JET. The most important ones are those for measuring, with accurate time and space resolution, plasma densities (n) and temperatures (T), which allow the key parameter to be derived, i.e. the fusion product of $n.\tau_E.T$ (where τ_E is the energy confinement time), plasma impurities, neutron rates, etc. An overall view of the comprehensive JET diagnostics system is shown in Fig. 17.

A common requirement for operating of diagnostics in the D- T phase, is that they can withstand the neutron environment. For this reason the key JET diagnostics have electronics and other key functional components installed away from the machine (> 50 m), outside the Torus Hall and beyond the radiation barrier.

As for all Torus Hall equipment, diagnostic components in the Torus Hall must be neutron induced radiation and tritium compatible and must be designed for remote handling maintenance and repair/replacement.

This represents a major departure from previous tokamak diagnostic designs and a significant contribution to the experience needed for the design of diagnostics required for a Next Step device and for a thermonuclear reactor.

5.5 Nuclear Systems

The ultimate JET objective of operation with the thermonuclear fuel (D- T) in near ignition conditions, has been one of the main constraints since the early JET design phase.

As previously mentioned, the Tokamak building has been designed according to nuclear standards. The main access to the tokamak is from the basement, for busbars, additional heating, cooling and cables, while for the diagnostics it is through the Torus Hall wall and ceiling. Each access is specially designed to prevent direct viewing of the machine, to 'contain' neutron streaming.

The Remote Handling programme [11] has followed the JET development with continuous interaction. A large variety of remote handling tools and procedures have been prepared: the overall pattern of the remote handling equipment is shown in Fig. 18. There are main transporters such as the 150 tonne crane and the articulated boom, fixing tools like the antenna grabber and the tile gripper and multi-purpose servomanipulators. The effectiveness of such tools has been already proven, since they are currently used for in-vessel work as the most effective way of performing the necessary operation as compared to a manual approach, even if the low radiation level in the vessel does not require their mandatory use. The articulated boom operating in the vessel for the installation of a section of the belt limiter is shown in Fig. 19.

The JET Active Gas Handling Plant [12] can be regarded as a meaningful prototype of a fusion fuel recycling system. It will receive all gases from the torus vessel, neutral injectors and pellet injectors; the gases of hydrogen isotopes (D, T) are purified (i.e. He^4, the product of the D-T nuclear reactions and other impurities are removed), and D and T are then separated out. Static cryogenic system have been chosen rather than mechanical pumps or chemical beds. Double containment for safety is also provided by the bakable double-walled vacuum vessel. The plant capacity is for a through put of 30 g of tritium per day.

5.6 Control and Data Acquisition System (CODAS) [13]

The purpose of CODAS is two-fold: to perform the control of the machine and its subsystems on the one hand and on the other hand, to collect, organise and store both the engineering and physics data.

This is achieved by an integrated system of a network of microcomputers (16 bit, ND-100 and 32 bit, ND-500 Norks Data computers) This computer system is connected to an IBM-CRAY computing facility for off-line data storage and analysis of the experimental results. There are 34 such microcomputers, which are divided into three networks; the on-line network (for day to day operation); the off-line network (for program and system development); and the data base network (for data base applications). Protection of JET subsystems is performed by hardware protection of each subsystem unit backed up by the Central Interlock and Safety System (CISS), which provide additional interlocks by means of programmable logic controllers. During a pulse, the key machine parameters and physics data are displayed in real time in the two control rooms (one for machine operation and one for plasma diagnostic measurements). A view of the machine control room is shown in Fig. 20.

6. OPERATION PERFORMANCE AND FUTURE DEVELOPMENT

6.1 Operation and Performance

The significant experimental results obtained with JET so far, have been made possible not only by the choice of the main tokamak parameters during the design phase, but also by the intelligent exploitation of its in-built flexibility, which has allowed enhancement of the plasma performance toward reactor relevant conditions. [14], [15]

A major example is the following. Experiments conducted in a wide range of plasma conditions have shown that the plasma current and the magnetic limiter configuration both play a key role in counteracting the decay of the energy confinement time, while increasing the plasma temperature. It has been possible to modify the electromagnetic system (tokamak magnet and associated power supplies) to allow both a substantial upgrading of the capability of the tokamak transformer and an active control of the magnetic limiter configuration (X-point) at high plasma currents.

As a result of this development JET now operates at 7 MA (higher than ~ 40% the design value of 4.8 MA) in the material limiter configuration (when the plasma is in contact with the belt limiter) and at up to 5 MA in X-point configuration (not even foreseen in the original design). The experimental results obtained during 1988 have permitted the quantification of the significant effects of this upgrading. [16], [17]

Another important achievement has been to show quasi-steady state plasma condition for 20 s at plasma temperatures $\geq$ 6 keV, with R.F. heating, while ohmicaly only heated discharges have been obtained for 30 s at 3 MA of plasma current. No other tokamak in the world can sustain a plasma for more than a few seconds. These results are of paramount importance for the Next Step (NET/ITER) where plasma discharges of several hundred seconds are required, while even longer discharges are obviously required in a thermonuclear reactor.

6.2 Further Development

The use of the JET design flexibility is far from being exhausted. A number of new engineering developments are underway: the most important are related to the reduction of the content of impurities and to their control. Impurities have two adverse effects on plasma performance: they increase plasma losses (radiation) and they reduce the amount of ions that can fuse (dilution), thus reducing the fusion product ($n\ T\tau_E$)

The first measure has been to replace carbon (Z = 6) by beryllium (Z = 4) as the plasma facing material. This has been achieved by means of beryllium evaporators which allow the vessel wall to be sprayed with a material at lower Z as compared to carbon and by replacing carbon tiles with beryllium tiles on the belt limiter and on other crucial segments of the vessel walls and in addition, by replacing R.F antennae nickel screens with beryllium screens. The first measure (beryllium evaporators) has already been implemented and has shown a typical increase of a factor of two of plasma performance. The long term measure would be the mounting of a divertor inside the vacuum vessel, which should allow the plasma impurities to be pumped away. If this system will be implemented, the choice will be made to operate JET at high performance only in single X-point magnetic configuration.

Other developments are also under way, such as the installation inside the vessel of eight saddle coils, (supplied DC by up to 3 kHz power supplies) to control radial plasma disruptions. Moreover, a 10 MW Lower Hybrid current Drive (LHCD) system should allow control of the plasma current profile which should have favourable influence on the microscopic stability of the plasma , aiming at increasing the MHD safety factor q at the plasma centre.

7. CONCLUSIONS

Although not achieved simultaneously, the values of the fusion parameters (ion temperature T_i and electron temperature T_e > 10 keV, density $n > 10^{20}$ m^{-3} and confinement time τ_E > 1s) have reached thermonuclear values. The fusion parameter ($nT\tau_E$) is only a factor of 12 short of reactor requirements and four times below breakeven, $Q_{th} = P_{out}/P_{in} = 1$, the goal of JET in D-T.

Present JET results allows the size and configuration of the Next Step machine to be defined. Moreover, the achievement of long pulse operation, suggests that there are no inseperable obstacles for such a device to operate in pulses for several minutes.

It can be stated that JET has already fulfilled two of its four objectives. The fulfilment of the third objective concerning the reduction of impurities relies on the success of the machine development at present underway.

A suitable method to further reduce impurity levels and to control impurity influx must be demonstrated before experiments in D-T must be effectively performed, to complete the data-base required to finalise the design of the Next Step Tokamak. This is the goal of a phase for JET presently under consideration.

It is important to mention that one of the key reasons for the success of JET has been the efficient co-operation with the European Industry. In addition, the construction of JET, achieved on time and to cost, can be considered a major joint achievement of the JET Joint Undertaking and of the European Industry.

Thus, JET is not only providing the data base in physics and engineering for the Next Step Tokamak, but will also provide an established relationship and a method of working with the European Industry, which should even increase its role in the Next Step following JET on the road to developing a fusion reactor.

Acknowledgements

This paper has been compiled largely using as main source of information the January 1987 issue of Fusion Technology, dedicated in full to the JET design, construction and early operation. Therefore, I would like to thank all my colleagues who are the authors of the various contributions reported in the references below.

A special thank you is due to my colleagues Dr B. Keen and Dr. B. Green for reading the manuscript and for the suggestions and comments made to improve this paper.

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TABLE I

Main Parameters of JET

PARAMETER	DESIGN VALUES	UPGRADED VALUES	ACHIEVED VALUES
Plasma Major Radius (R_o)	2.96 m	2.96 m	2.5 - 3.4 m
Plasma Minor Radius (a)	1.25 m	1.25 m	0.8 - 1.2 m
Toroidal Magnetic Field at R_o (B_{To})	3.45 T	3.45 T	3.45 T
Magnetic Flux Swing	34 Wb	> 40 Wb	40 Wb
Plasma Current	4.8 MA	7 MA	7 MA
Pulse Length at 3 MA	20 s	> 30 s	30 s
Pause Time	600 s	600 s	600 s
Neutral Beam Power	20 MW	20 MW	21 MW
ICRF Power	15 MW	18 MW	18 MW

TABLE II

Main Parameter of the Vacuum Vessel

Major radius of the toroidal vacuum vessel (mm)	2960
Internal diameter of chamber (equatorial plane) (mm)	2630
Internal daimeter of chamber (perpendicular to equatorial plane) (mm)	4180
Wall thickness (mm)	12-16-18
Volume of vacuum chamber (m^3)	~ 200
Internal wall area of equivalent torus (with smooth surface) (m^3)	220
Bellows material	Inconel 625
Rigid sector material	Nicrofer 7216LC

TABLE III

Main Parameter of the Toroidal Magnetic Field Coils

Number of coils	32
Total number of ampere turns (MA)	51
Magnetic field at 2.9-m radius (T)	3.45
Current (kA)	67
Resistive power (MW)	280
Energy dissipated per pulse (GJ)	5.5
Dimensions of each coil (vertical/horizontal) (m)	5.68/3.86
Number of turns/coil	24
Copper cross section/turn (mm^2)	3900/2700

TABLE IV

Main Parameters of the PF Coil System

	Coil-Type Number						
	1	2	3				4
Field Type	Magnetizing	Vertical	Radial	Magnetizing	Vertical	Radial	Vertical
Number of Coils	8		2		2		2
Weight (tonnes)	90		28		84		158
Outer diameter (m)	2.17		4.84		8.37		10.91
Conductor size (mm^2)	54 x 39.5	82 x 35	35 x 35	82 x 35	35 x 35	35 x 35	82 x 35
Number of turns	568	40	32	20	120	40	122
Maximum Current (kA)	40	45	20	40	20	20	25
Maximum Voltage to earth (kV)	20	20	3	20	20	3	20

TABLE V

Main Parameters of the Magnet Power Supplies

SYSTEM	PARAMETER	VALUES
Power grid (CEGB supergrid, 800 m from JET Site)	AC voltage Short circuit level Peak power Energy/pulse	400 kV 15 - 35 GVA 575 MW 15,000 MJ
Flywheel generator convertor (2 units, vertical shaft, diode rectifiers, 6 pulses)	Load DC voltage DC current Power Energy/pulse	6.0 kV 67 kA 400 MW 2,600 MJ
Static unit (2 units, thyristor rectifiers, 12 pulses)	Load DC voltage DC current	1.7 kV 67 kA
Vertical field and shaping amplifiers (6 units, thyristor rectifiers, 12 pulses)	DC load voltage DC current	1.0 kV 40 kA
Radial field amplifier (2 units, 12 pulses)	DC load voltage DC current	± 2.0 kV ± 3.0 kA

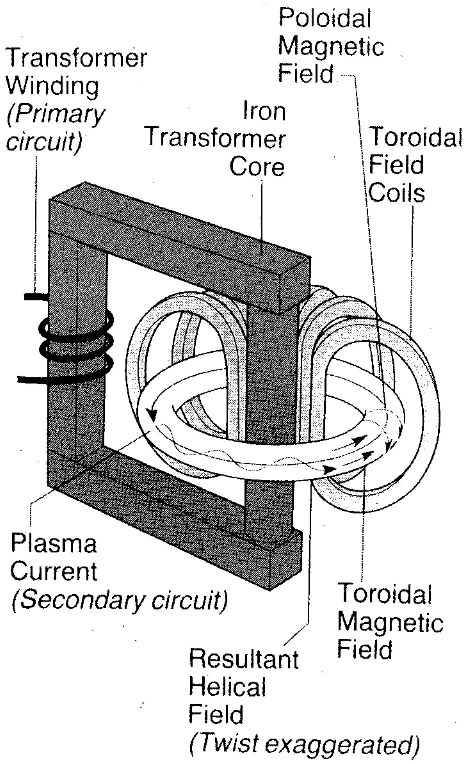

Fig. 1 Principle of a tokamak plasma magnetic confinement system.

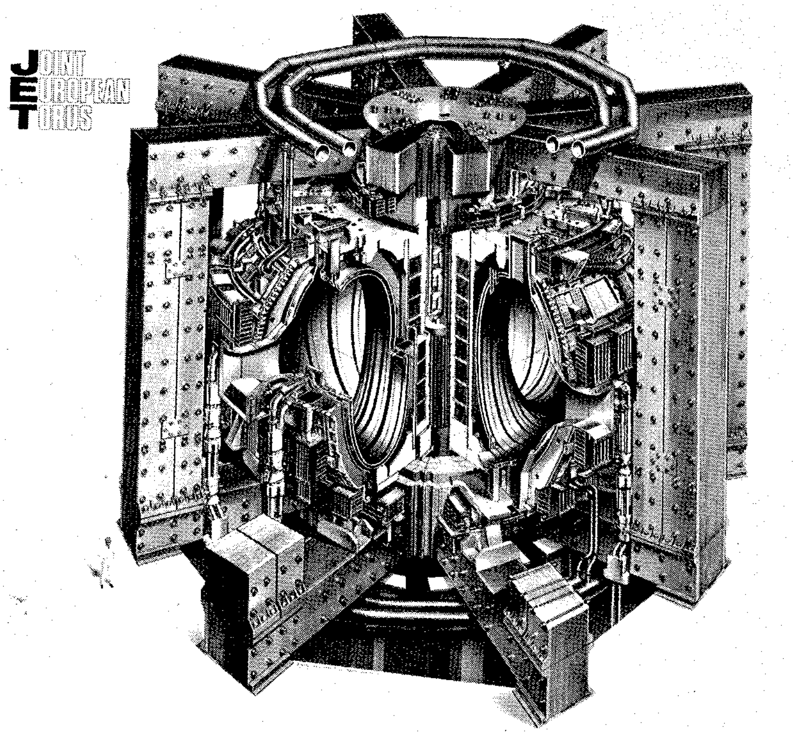

Fig. 2 Schematic view of the JET Tokamak showing its main components and key features.

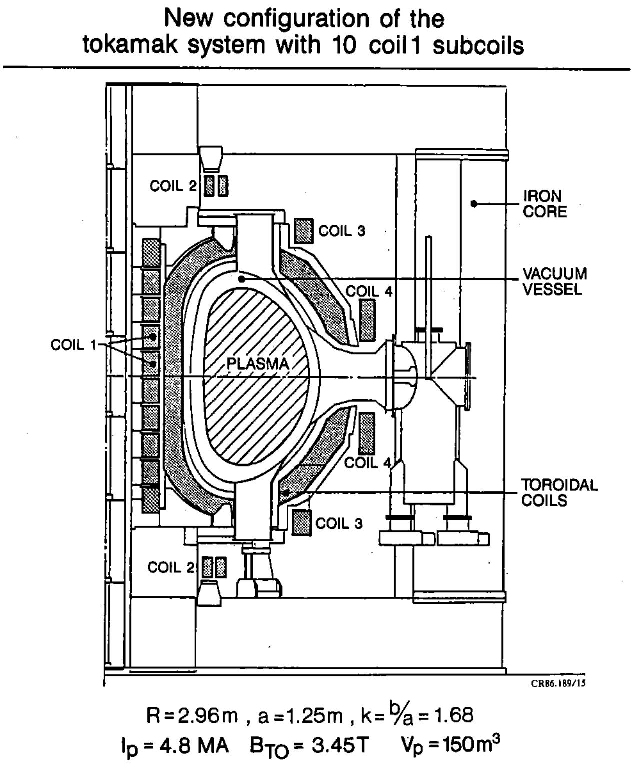

Fig. 3 Cross-section of the JET Tokamak emphasizing the 'D' shape.

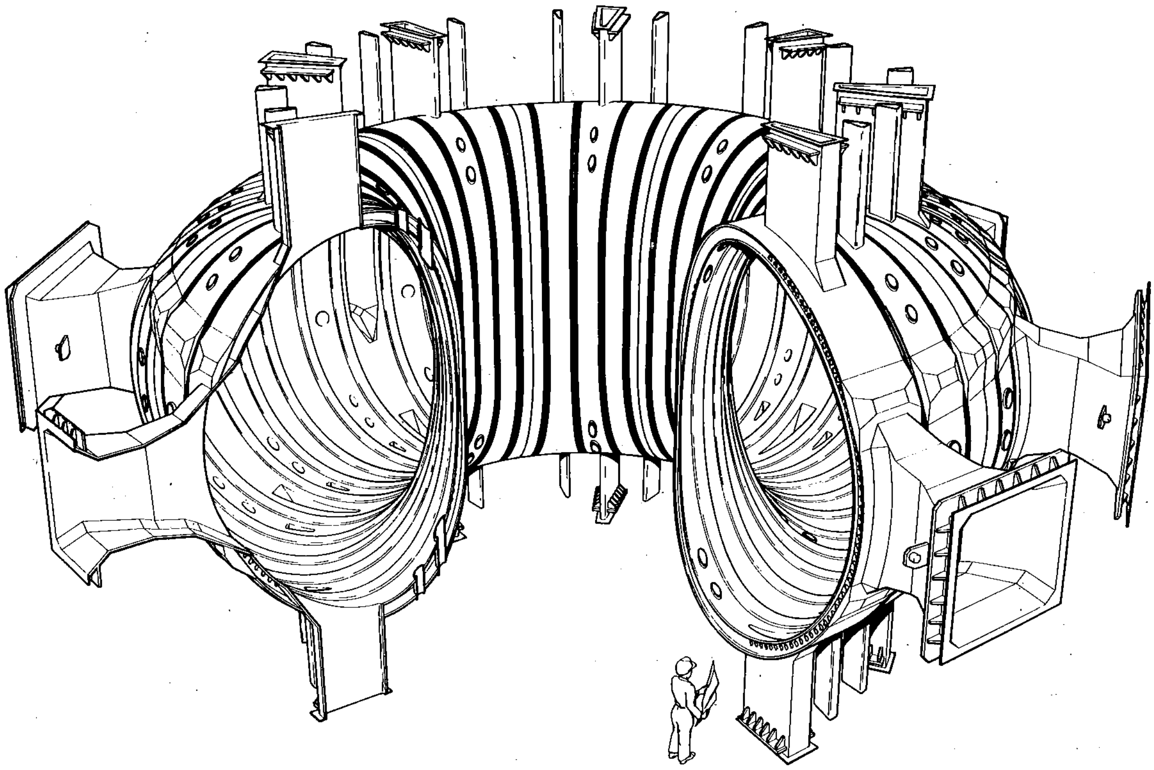

Fig. 4 The JET Vacuum vessel design showing the double walled octant structure with rigid sector and bellows sections.

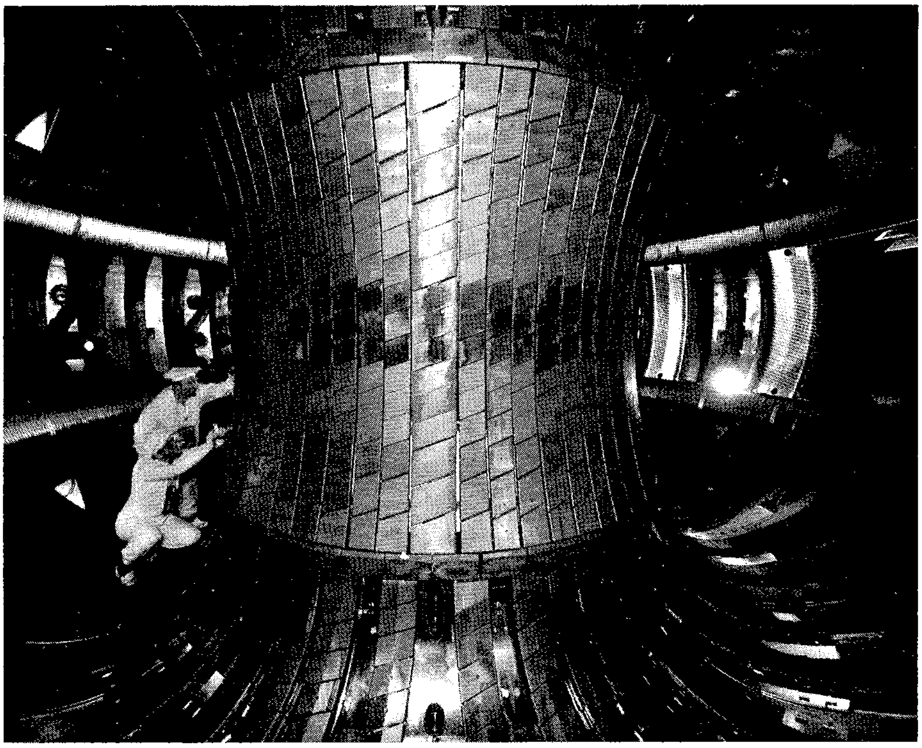

Fig. 5 Present configuration of the vacuum vessel inner walls (June 1989), showing belt limiter, RF antennae and graphite tiles.

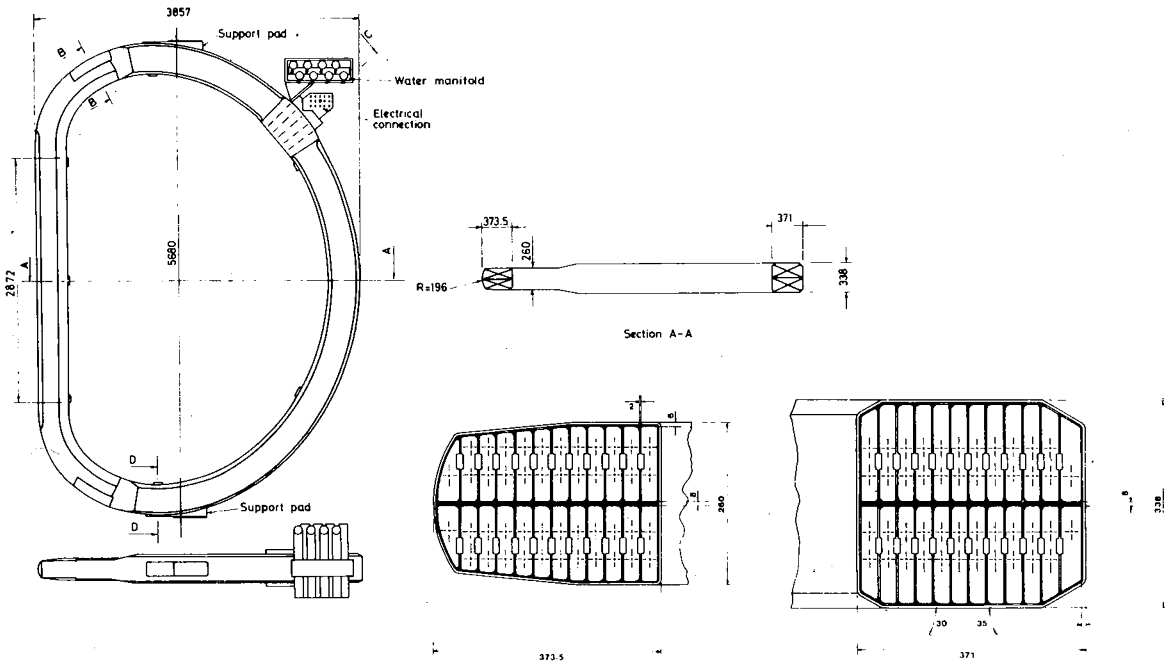

Fig. 6 The key design features of JET toroidal field coils.

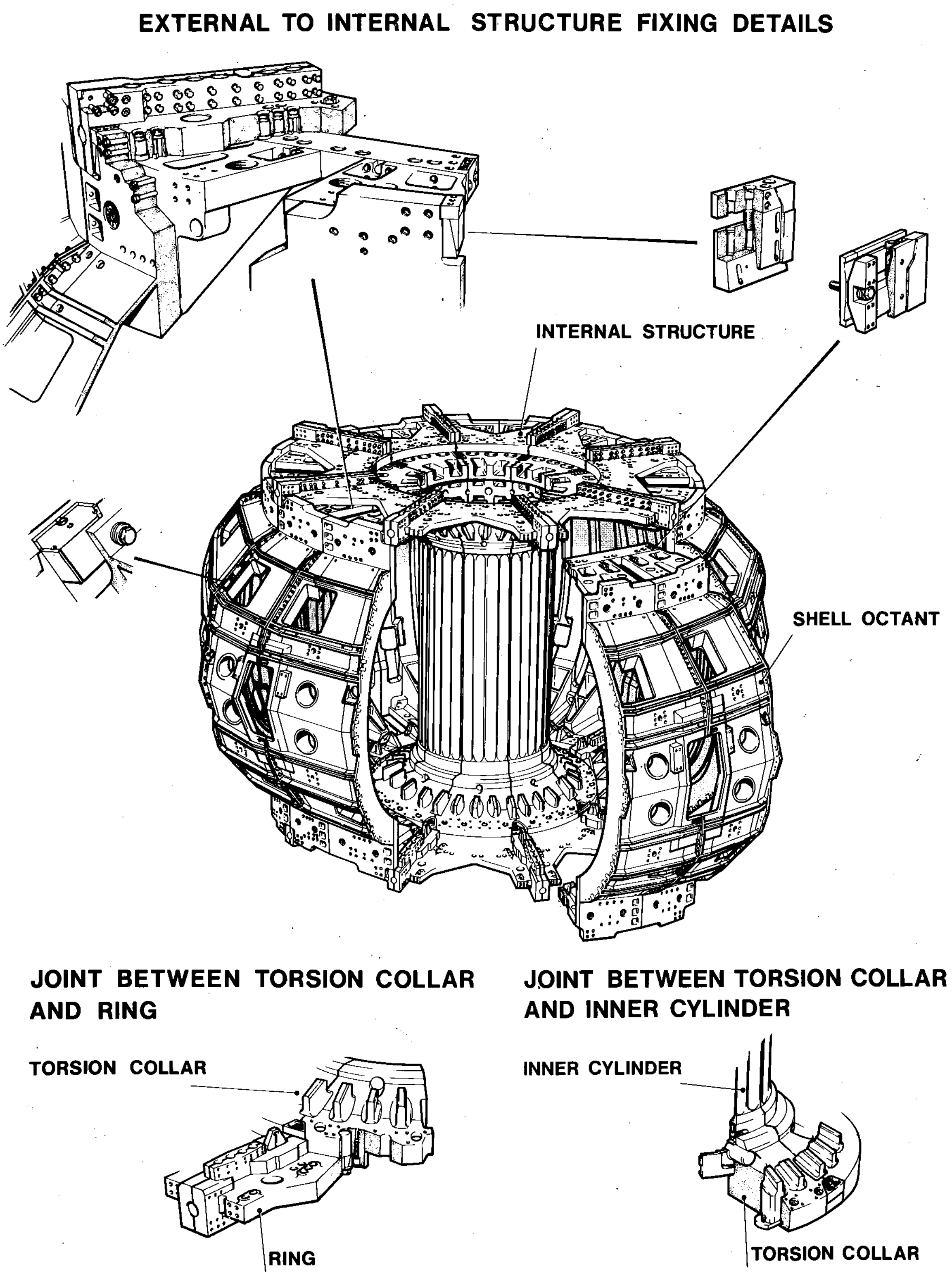

Fig. 7 Overall view of the main components of the JET mechanical structure.

Fig. 8 Early stage of the JET Tokamak assembly: iron core and outer poloidal coils.

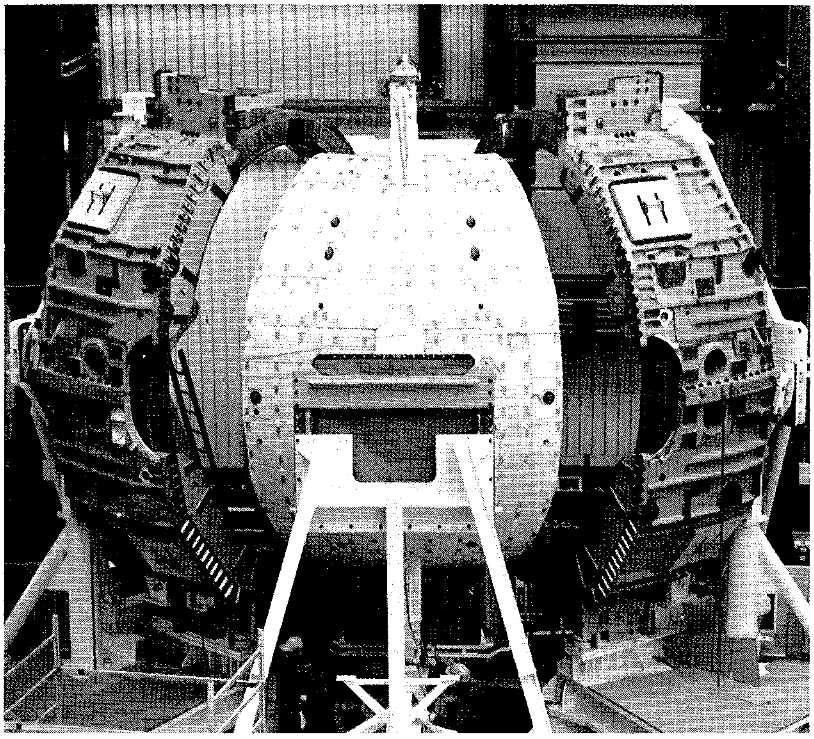

Fig. 9 An octant on the assembly jig.

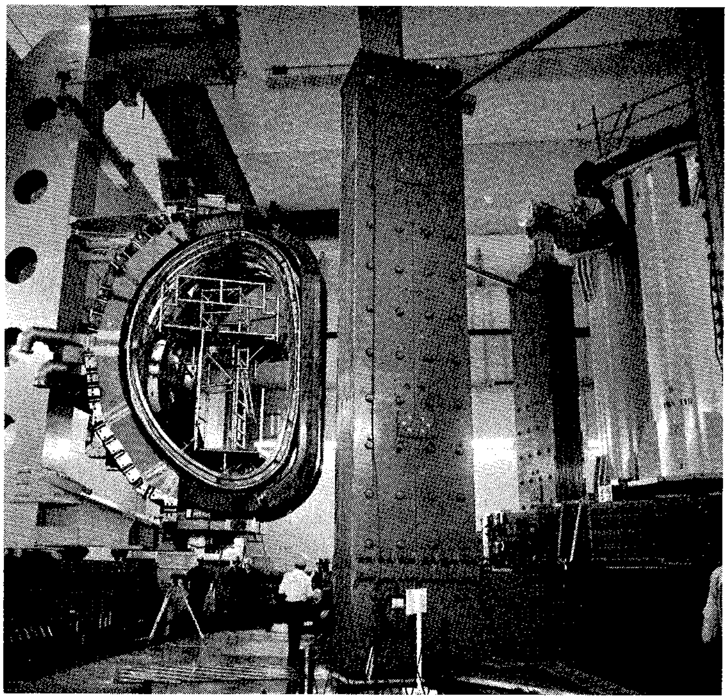

Fig. 10 An octant fully assembled is located in its final position.

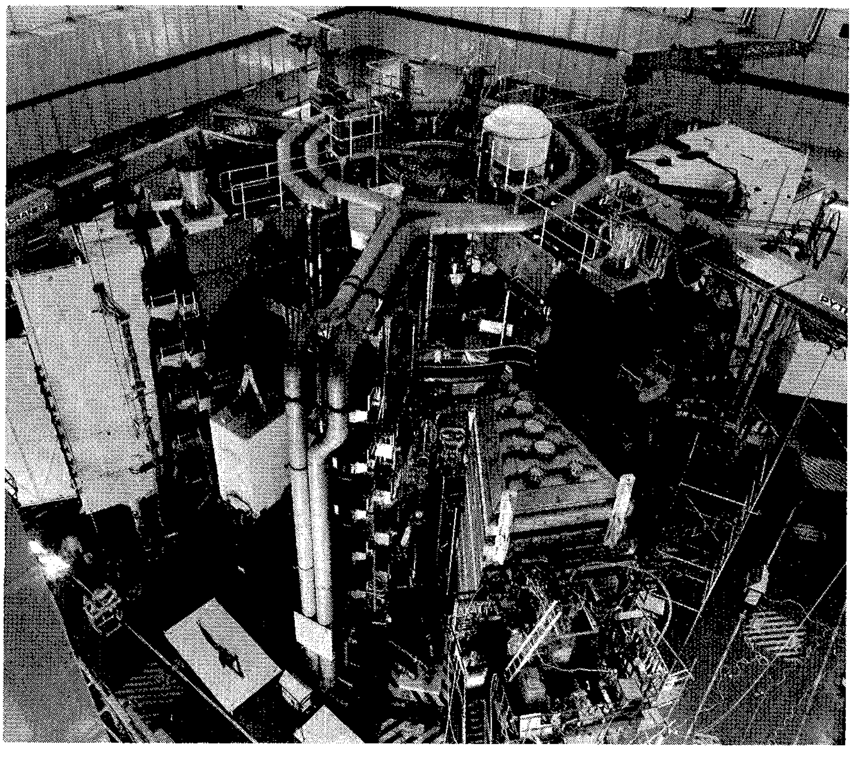

Fig. 11 An overall view of the JET Tokamak at June 1989.

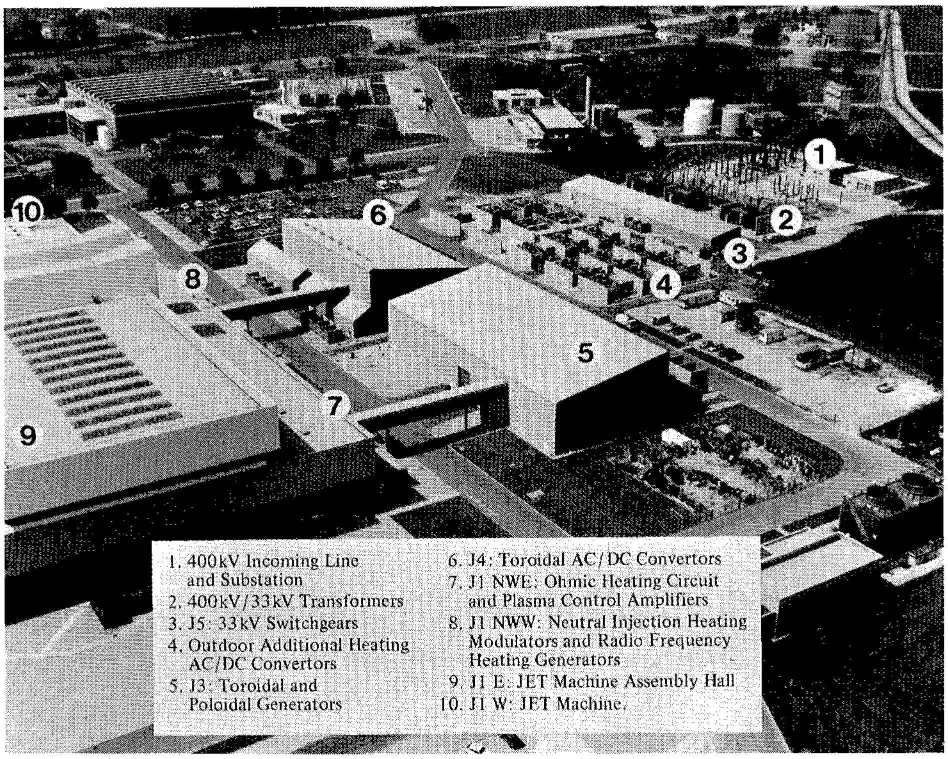

Fig. 12 The JET Site layout showing power supplies outdoor areas and building and the main JET buildings.

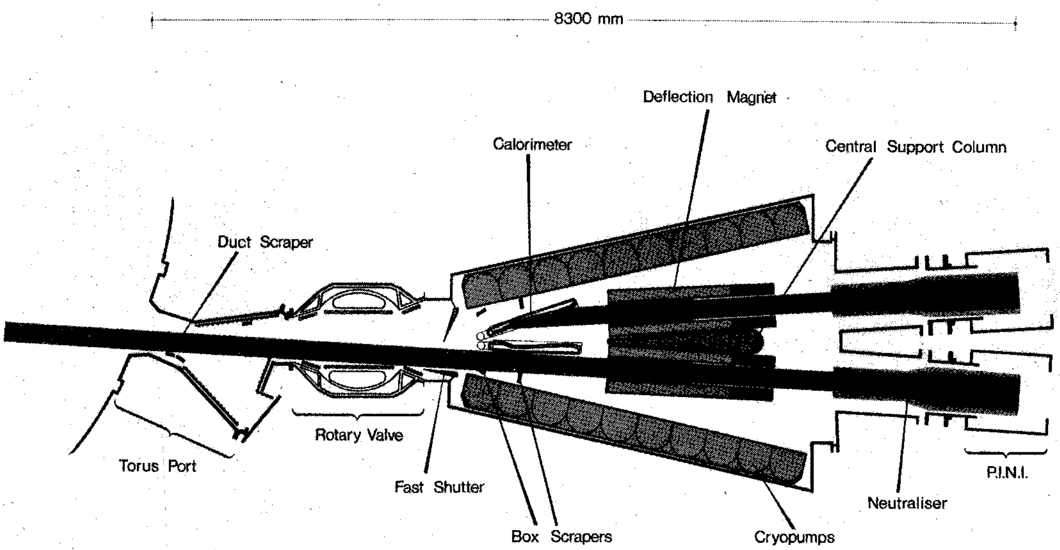

Fig. 13 Schematic of a Neutral Beam Injector.

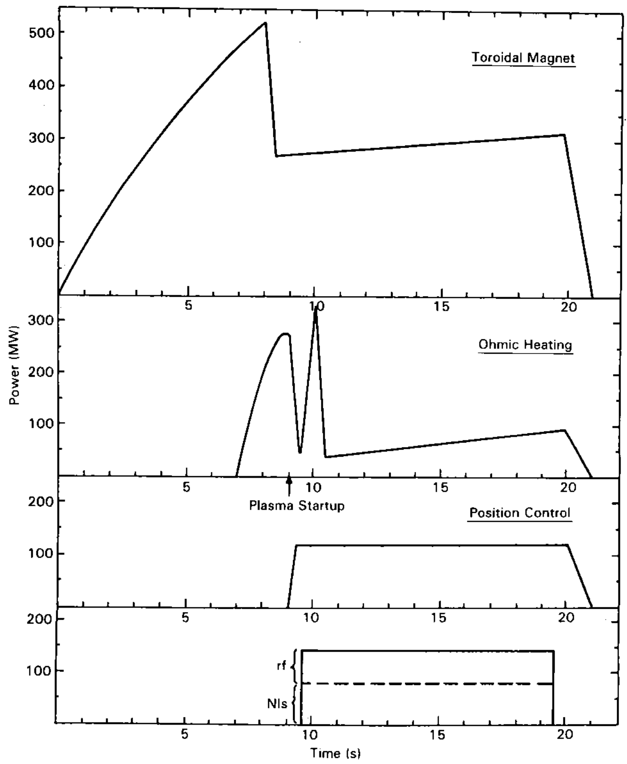

Fig. 14 A typical power requirement profile for the four main JET pulsed loads.

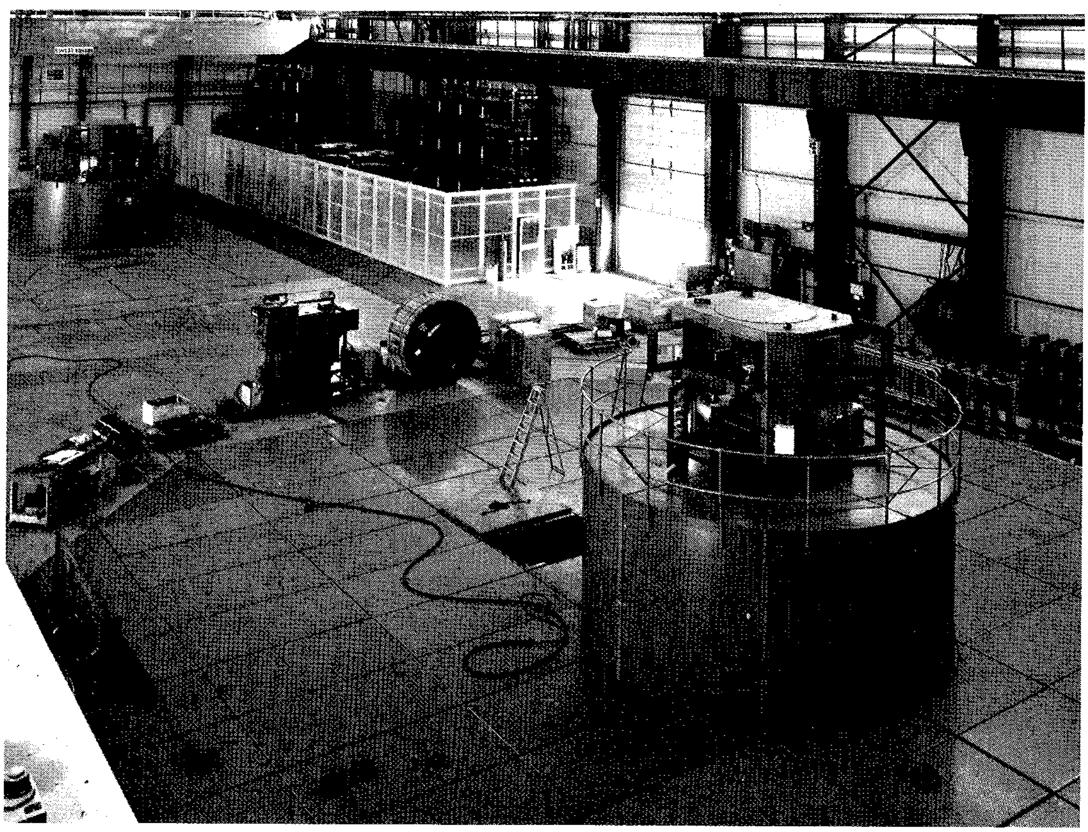
Fig. 15 The Flywheel Generator Hall.

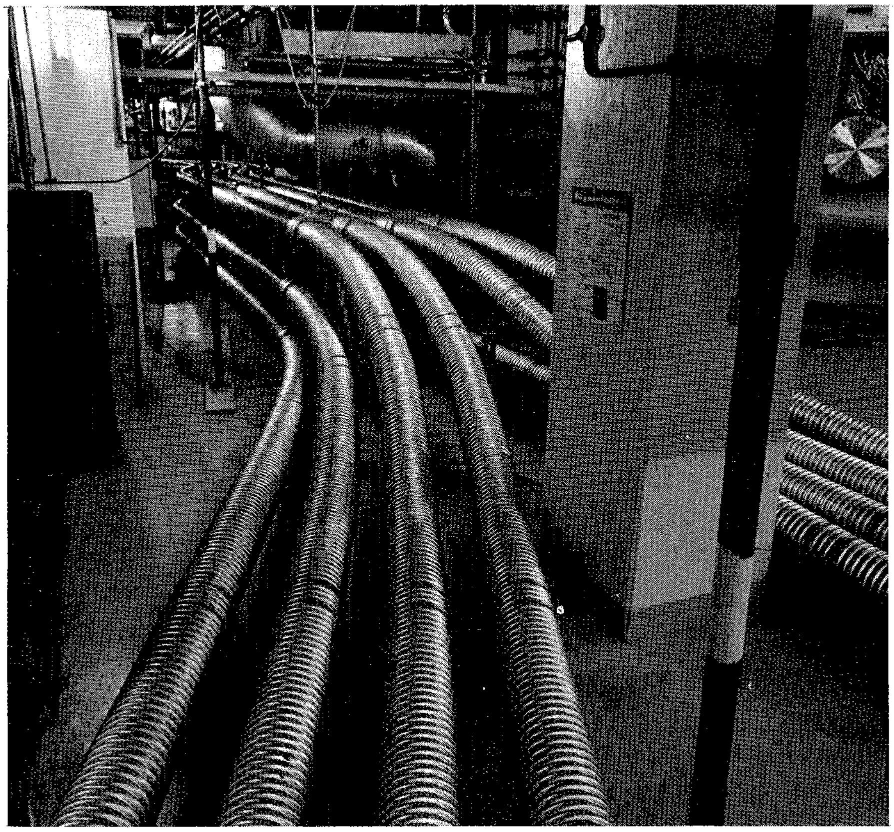

Fig. 16 The SF6 insulation power transmission lines to the neutral injectors run through the JET building basement.

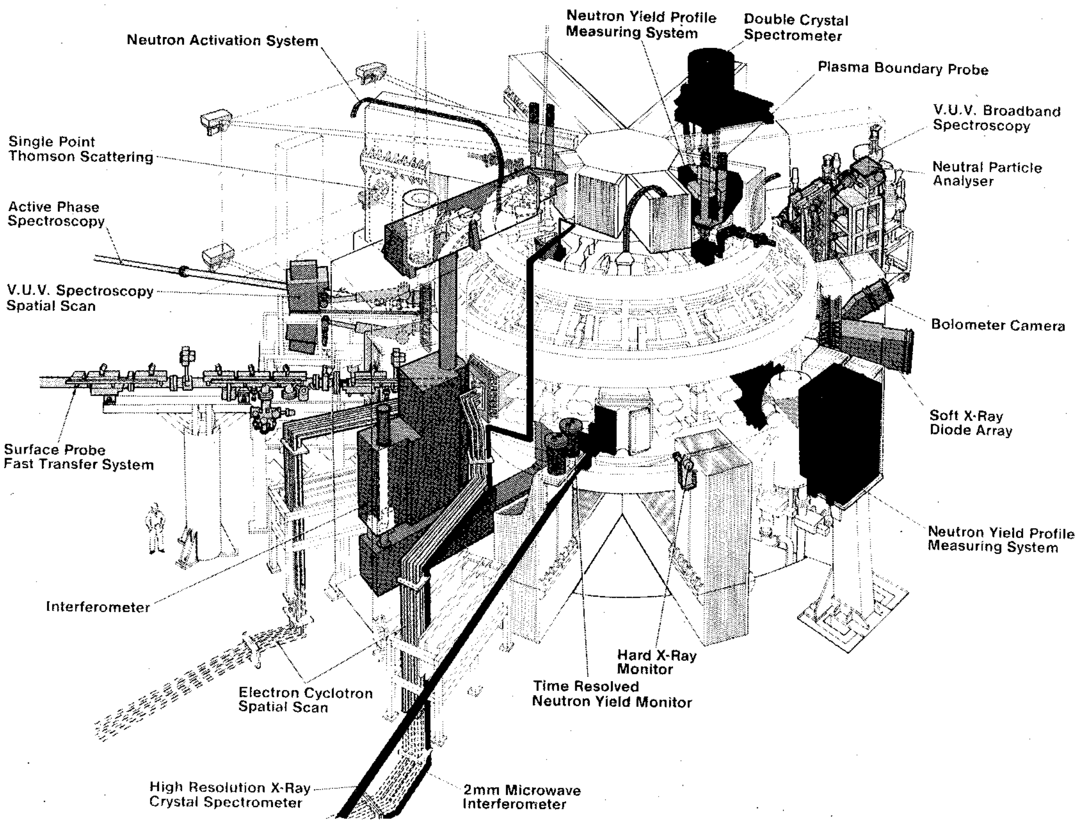

Fig. 17 Overall view of the schematic assembly of the main JET diagnostics.

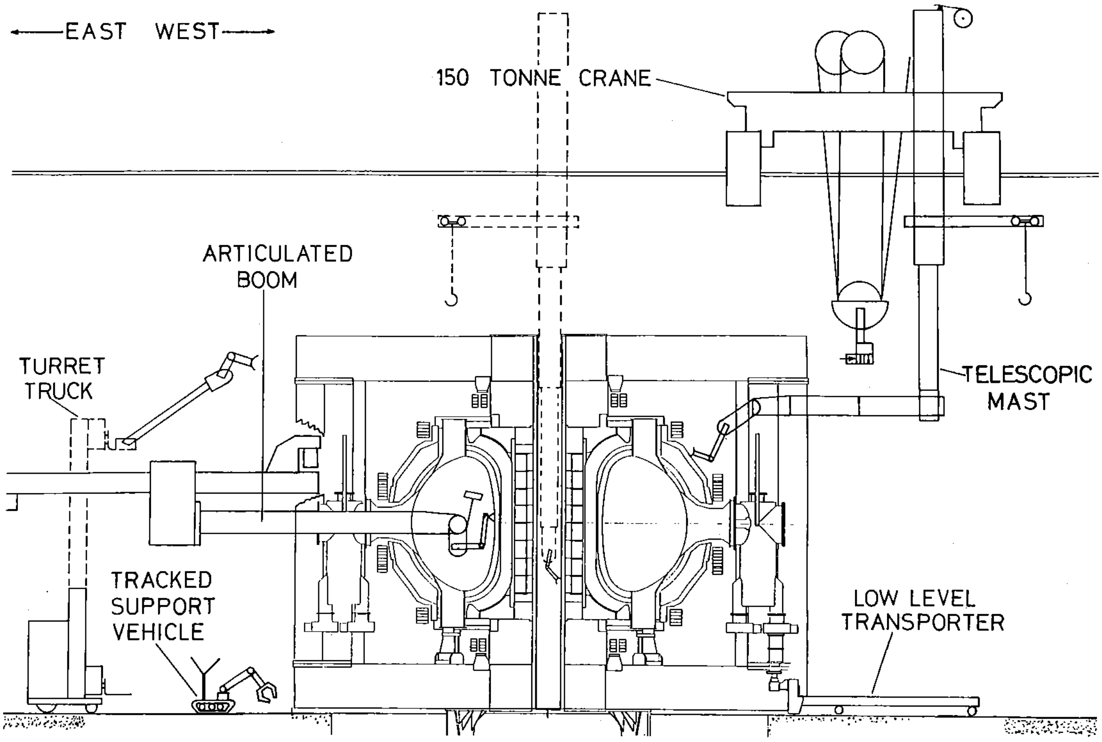

Fig. 18 The overall pattern of remote handling.

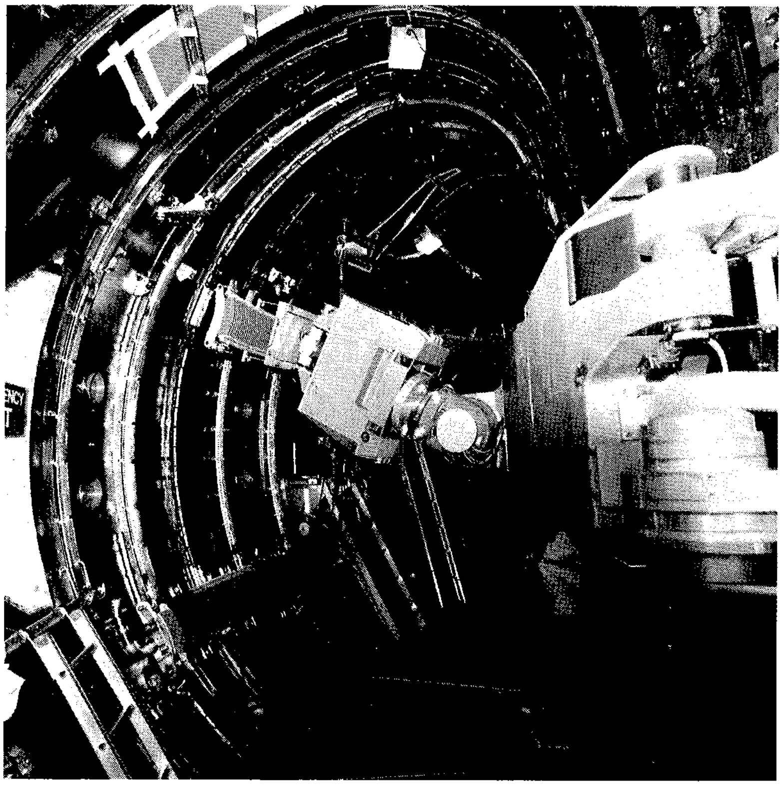

Fig. 19 One of the main remote handling transporters, the articulated boom in operating in the JET vacuum vessel.

Fig. 20 Overall view of the JET Machine Control Room showing control panels and viewing screens.