JET–P(97)49

Current Engineering Issues and Further Upgrading of the JET Tokamak

The JET Team

presented by E Bertolini.

JET Joint Undertaking, Abingdon, Oxfordshire, OX14 3EA,

December 1997

ABSTRACT

The design of the Joint European Torus was conceived with inherent flexibility to accommodate modifications and upgradings to match the evolving requirements of the physics programme, while preserving basic machine structure. The first major upgrading was to increase the plasma current capability from 4.8 to 7MA in limiter configuration and from 3.0 to 5MA in X-point configuration. The second change was the progressive covering of the vessel walls with low-Z materials such as graphite and beryllium. The most recent major modification was to make JET into a pumped divertor machine. Three divertors are being tested in sequence (Mark I, II, IIGB), in support of the ITER design. JET is operating at present with Mark II both in D-D and in D-T. Thus, the installation of Mark IIGB will be performed using only remote handling techniques. Divertor plasmas are more vertically unstable, and so a new plasma control system had to be designed and implemented. The engineering instrumentation of the machine has been upgraded, for machine protection and to monitor and study new phenomena such as sideways vessel displacements, caused by plasma disruptions. An in depth reassessment of the toroidal coils, of the mechanical structure and of the vessel is in progress. This includes finite element calculations and mechanical tests on samples and on two toroidal field whole coils, to evaluate the machine capability to operate at higher toroidal field (from 3.4T to 4.0T) and operation at 3.8T has been undertaken already. In the early phase of the 1997 D-T campaign ~13MW of fusion power have been produced.

1. INTRODUCTION

JET started operation in June 1983, as the largest tokamak experiment of the coordinated fusion research programme of the European Union. The global objective of JET is to produce and study plasmas of thermonuclear grade, in configurations suitable for extrapolation to a reactor, such as ITER (International Thermonuclear Experimental Reactor). This led to the early choice of machine parameters with D-shaped toroidal coils, vacuum vessel and plasma cross-section . Great flexibility and suitable stress margins were included in the original JET design, to allow modifications and/or upgrading of the machine to follow the evolving requirements of the physics programme [1,2].The most straightforward way to develop fusion performance is to upgrade machine parameters, to increase the additional heating power, and to explore new plasma regimes. Indeed, two major interventions took place between 1986 and 1989. The first, involved increasing the plasma current capability from the design value of 4.8MA to 7.0MA in limiter configuration and from 3.0MA to 5.0MA in the X-point configuration. This work involved revising the design of the toroidal and poloidal coils, of the mechanical structure and of the vacuum vessel, by detailed finite element computer modelling and calculations, and fatigue tests on the prototype toroidal coil. Moreover, new power supplies had to be provided and plasma control also had to be improved, to cope with the enhanced vertical instability of the plasma ring. This upgrading was generated by initial experimental results, which showed a sharp decay of the energy confinement time with heating. This could be counteracted by increasing the plasma current and by setting up an X-point configuration, which allowed H-modes to be established [3]. The second change required progressively covering the inconel vessel walls with low-Z materials, i.e. graphite tiles (Z=6) and later beryllium tiles (Z=4), supplemented at first, by wall carbonisation and later by beryllium evaporation [4]. This intervention was prompted by the need to substantially reduce Z_{eff}.

The full implementation of these measures, led to the development of the hot-ion regime, showing a spectacular increase in overall plasma performance. The fusion triple product improved from 0.12 to $0.9x10^{21}m^{-3}$skeV and the equivalent energy gain Q_{DT} increased from 0.01 to 1.07. In separate pulses, an ion density of n_D~$4x10^{20}m^{-3}$, an ion temperature T_D~30keV and an energy confinement time τ_E~1.8s, were obtained. Finally, a $Z_{eff} \leq 2$ and a dilution factor $n_D/n_e \geq 0.9$ were reached. This level of global plasma parameters was sufficient to perform the first ever D-T experiment; which in spite of using a mixture far from optimum (11% T- 89%D), achieved a peak fusion power of 1.7MW, with >50% of thermalised neutrons, and a fusion energy of 2MJ [5].

The additional heating power has been progressively increased up to 50MW (20MW of Neutral Beam Injection, 20MW on Ion Cyclotron Resonance Heating and 10MW of Lower Hybrid).

2. JET WITH A DIVERTOR

These results fulfil, to a large extent, the JET original objectives. However the high performance could only be maintained for about ~1s, limited by a combination of MHD instabilities and accumulation of impurities in the X-point region. Active control of the impurities was therefore required, and it has been achieved by installing a pumped divertor, to control impurity levels, particle and energy exhaust, and enhance plasma energy confinement in H-mode[6]. This required the third major upgrading of the JET tokamak. The original features of the JET design allowed the basic structure of the machine to be maintained (i.e. toroidal and poloidal coils, mechanical structure and vacuum vessel). In fact, the necessity to install the divertor coils inside the vacuum vessel, produced a loss of ~25% of the plasma volume, but it allowed plasma currents up to 6MA to be accommodated.

Three divertor configurations (Mark I, Mark II and Mark II Gas Box), have been designed, with progressively more closed configuration to enhance particle and impurity retention in the divertor chamber and to increase the amount of plasma energy released by radiation (Fig. 1). The results of JET divertor studies are of great importance to finalise the ITER divertor design [7].

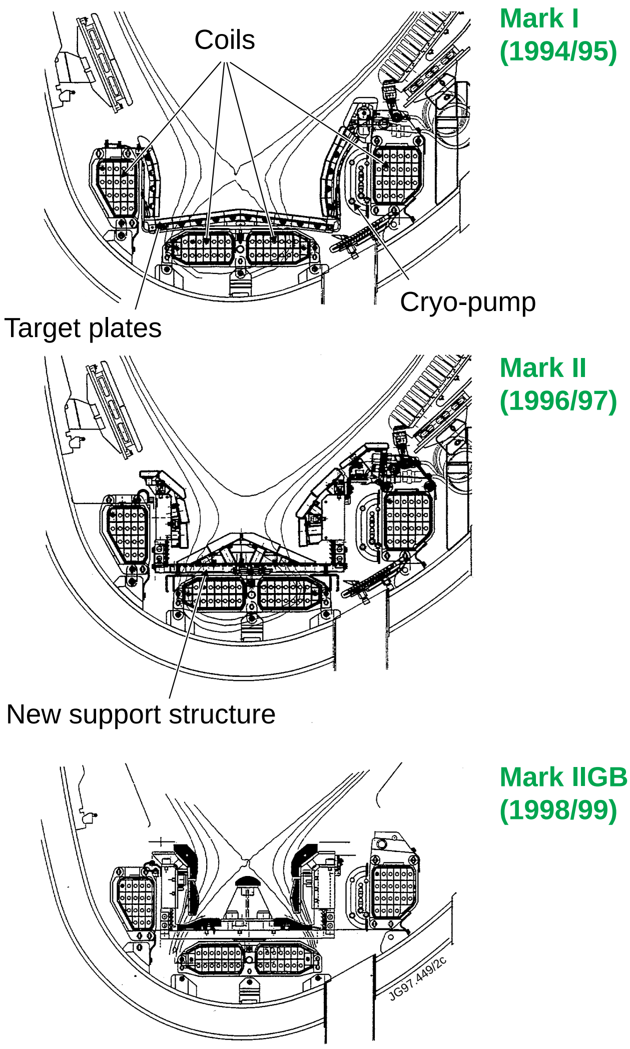

Fig. 1. The three JET divertor configurations, showing divertor coils, target plate arrangements and cryopump

3. SOME KEY ENGINEERING ISSUES

Mark II, now operational, has been designed to limit the modifications required to install any new divertor configurations at a later stage. It consists of an inconel water cooled continuous tray, which acts as a support structure for the tile carriers.

Extensive exploration of the 'high performance regimes', hot-ion and optimised shear regimes, shows already a global performance identical, if not above, that achieved without divertor (when 25% more plasma volume was available).

3.1. Replacement of Mark II with Mark IIGB by Remote Handling

JET has developed RH (Remote Handling) concepts and tools as part of its original design, to meet the requirements of unmanned interventions following extended D-T operation. The first major RH exercise will be in early 1998 when the Mark II tile structure will be replaced by the

Mark IIGB structure. The sequence of operations has been studied in great detail, and it has been tested in a full size vessel mock up (In-Vessel Training Facility). Moreover, 20% of Mark II tile carriers have been installed by remote handling. Within the vessel, the tile carriers are handled and positioned by the Mascot IV servo-manipulator mounted on an articulated boom transporter, using Octant No.5 port (Fig. 2) [8,9].

Fig. 2. Installation of Mark II divertor tile carriers by remote handling techniques

3.2. Plasma Control

Due to the enhanced vertical instability and to the complexity of JET divertor plasmas, a new PPCC (Plasma Position and Current Control) was conceived, designed and installed [10]. PPCC is based on the control of the plasma boundary in real-time by means of poloidal coil currents and plasma-vessel wall gap control.

The handling of divertor plasmas requires a continuous upgrading and modification of PPCC. Its flexibility and accuracy have been enhanced for the Mark II experimental campaign (Fig 3), by re-designing both systems: Shape Control (SC) and Vertical Stabilisation (VS).

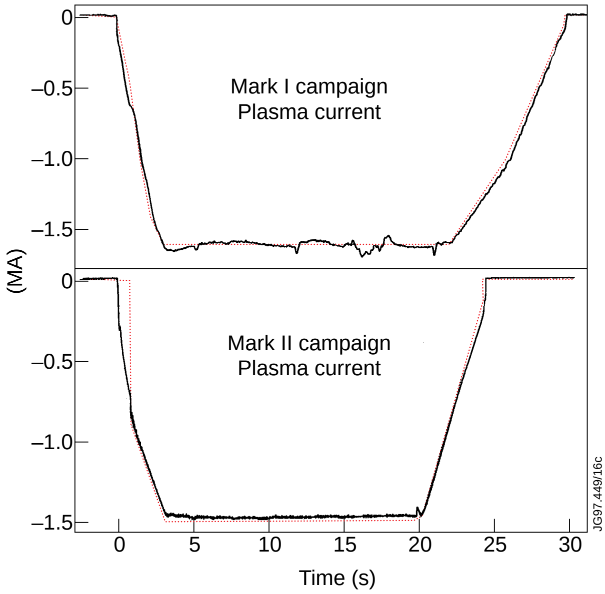

Fig. 3. Modifiations to the Plasma Position and Current Control with Mark II show better plasma current control accuracy

Recently a new first wall protection system, WALLS (Wall Load Limiting System), has been implemented to prevent thermal overloading of the first wall graphite tiles: the system can estimate the energy deposited locally in 21 different positions, during the pulse and terminate the pulse as required. WALLS relies on the interconnection of three control systems: plasma shape, position and current control and the real-time input power control [11].

3.3. Plasma behaviour and structural components

Vertical instabilities and disruptions have been experienced by JET since the early phase of operation at levels not foreseen in the original design. The JET vacuum vessel is relatively flexible, so these events showed up first as vessel movements. Therefore, measurements of displacements and forces, together with more robust vessel supports were progressively implemented, providing a satisfactory monitoring and control of the events. However, divertor operation highlighted the presence of new phenomena. It was always assumed that the vertical forces were essentially toroidally symmetric. It was surprising to notice, in the early phase of divertor operation (Mark I, May 1994), that some vertical disruptions gave rise to non-axisymmetric forces. Moreover, a major vertical disruption occurred in February 1995 and produced a displacement of the whole vessel sideways by 5.6mm [14]. However, by reviewing the vertical disruptions of the past, it was noticed that sideways displacements did occur, even without a divertor. Extensive work, still underway, has been performed on JET and other tokamaks, where similar events were experienced. Although a way to prevent them has not been found, some understanding has been developed [13,14].

Sideways displacements are recorded only when plasma kinks lock long enough in the same toroidal position, so that a net radial sideways impulse can build up. The sideways force acting on the vessel is, in the worst asymmetric events, of the same order of magnitude as the vertical force (~1 MN). It was found that the initial main restraining action came from magnetic damping: when the vessel is moving in the toroidal magnetic field, eddy currents are induced on its surface which tend to keep the flux constant inside the torus. Later, these decay due to the electric resistance of the vessel. The largest sideways displacement observed so far, is ~7mm. The upper boundary of the vessel sideways displacements scales with the product of the plasma current and the toroidal field (Fig. 4). To reduce the amplitude of the sideways displacements, new and much more effective restraining supports are being designed.

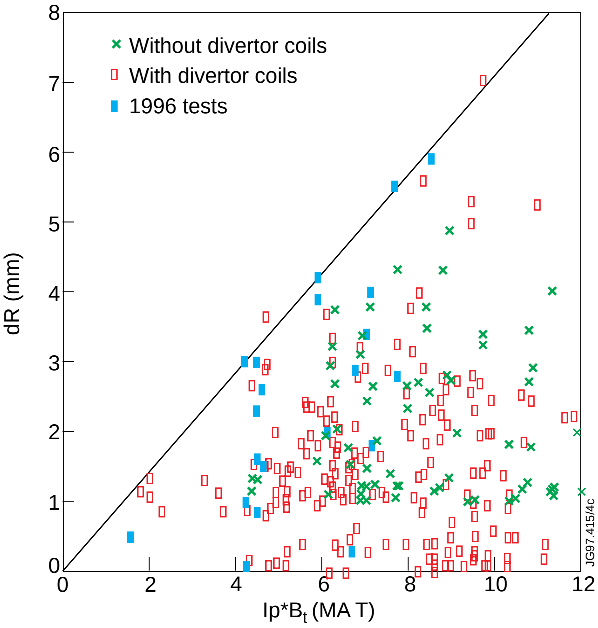

Fig. 4. Scaling of the upper boundary of the vessel sideways displacements with toroidal magnet field and plasma current

4. UPGRADING FOR OPERATION AT 4T TOROIDAL MAGNETIC FIELD

JET operates at the lowest toroidal magnetic field, among the large tokamaks (3.4T). Plasma performance would greatly benefit by an increase of the magnetic field. It was decided to embark on a major re-assessment of the basic machine (i.e. the toroidal coils, the mechanical structure and the vacuum vessel) to prove that the toroidal field could be safely increased to 4T, with a plasma current up to 5.0MA, without replacing or modifying any of the subsystems.

4.1. The toroidal coils

JET has 32 toroidal field (TF) copper coils, each made up of two adjacent pancakes with 12 turns each. Each coil is subject to a net inward force which is reacted along the straight section by the inner poloidal coils (P1): both tension and the inward force are proportional to B_T^2. The lateral forces on the coils, caused by the interaction of the coil current with the poloidal magnetic field, are supported by the mechanical structure in 12 positions along the coil (Fig. 5). The areas of main concern are the collar teeth, because a full support could not be installed due to the lack of space, and the ring tooth, due to the limited stress capability of the teeth bolts. The coils were originally water cooled. Between 1989 and 1992, three of these coils had to be replaced with available spares, as these developed inter turn electrical shorts generated by water leaks. Thus the water coolant was replaced with freon. The status of the coils is continuously monitored, and since then coils are in good electrical and mechanical condition [19].

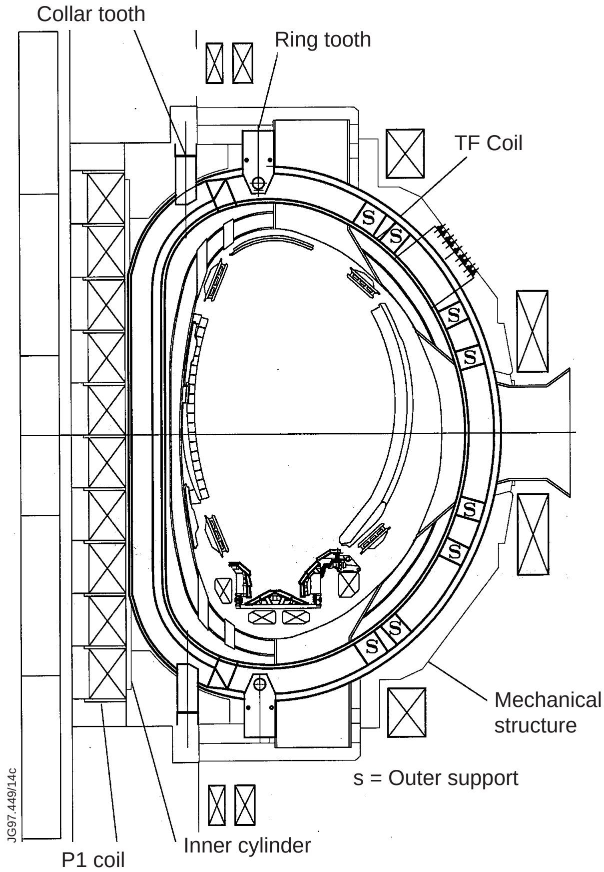

Fig. 5. Cross section of the JET tokamak showing supports of the toroidal coils

4.1.1. The MAXFEA code

The main computational tool used to assess forces is the MAXFEA (Maxwell Finite Element Analysis) code, developed at JET and now widely used, both for assessing operating scenarios and for design purposes (ITER) [20]. The 2D component of MAXFEA is a finite element plasma equilibrium code, to solve Maxwell's equations combined with the Grad-Shafranov's equation. In addition, the toroidal magnetic field is calculated by a 3D Biot Savart magnetic code. Input to the code are plasma parameters and TF and PF coil currents, while the main outputs are the out-of-plane and the in-plane forces acting on the TF coils. Subsequently, by using the ABAQUS code, forces and stresses on the coils and on the supports are calculated.

The forces on the mechanical supports of the TF coils and the stresses on the coils reach maximum values in equilibrium and they decay during disruption, with the exception of the forces on the ring teeth. Therefore, allowable stresses are set for a large number of pulses (>50,000). Since the number of large disruptions is limited and their number can be controlled, the allowable can be set for a much reduced number of pulses (range~1,000).

Forces and stresses have been calculated for the reference combination of 4.0T, 5MA, for JET main operating scenarios, in equilibrium and following a disruption event (hot-ions, optimised shear and ELMy H-modes). The main stresses on the coils and on the supports are still lower than the allowable stresses considered so far, with the exception of the ring teeth, following a disruption event, in the hot-ion mode (Table I).

TABLE I
Stresses on TF Coils and forces on supports calculated with MAXFEA and ABAQUS beam model

	HOT ION 4T/5MA			
	Equillibrium	Disruption	Allowable value	Unit
Shear stresses in the insulation				
τ Interpancake	11.5	8.0	15	MPa
τ Interturn (Peak) Collar tooth TOP	10.9	5.5	13	MPa
τ Interturn (Peak) Collar tooth BOTTOM	6.7	8.7	13	MPa
Tensil stresses in copper				
σ Total (nose)	117	102	128	MPa
σ Member + Bending (Brazed joints)	71	74	74	MPa
σ Membrane (Brazed joints)	60	60	58	MPa
Forces at the MS teeth				
F Collar tooth, top	–321	163	350	kN
F Collar tooth, bottom	197	257	350	kN
F Ring tooth, top	468	649	675	kN
F Ring tooth, bottom	609	–717	675	kN

JG97.449/17c

4.1.2. Mechanical tests

It has been decided to sacrifice the first faulty coil (Coil 3.1). The coil was cut in slices at various positions (Fig. 6) with the unprecedented objective of repeating tests on samples extracted from an extensively used coil, that are usually made before manufacturing to validate the fabrication process [15]:

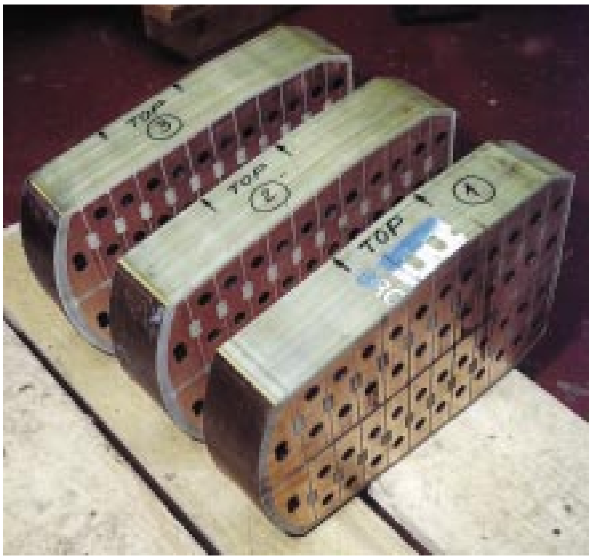

Fig. 6. Cuts of one of the replaced toroidal coils for visual inspection and for extracting test samples

a) careful micro-examination (x32 to x200) of the status of the glass- fibre epoxy resin and bonding, including the uniformity in the composition fibres-epoxy: the inspection indicates that the mixing of fibre and epoxy is good and that the micro voids are typical of a new coil in size and number;

b) cut out samples, to measure the fatigue shear stress capability of the bonding both at 20^{o}C and at 70^{o}C: the results showed practically no ageing of the epoxy, i.e. the shear stress capability measurements obtained with the coil samples and the ones obtained with the manufacturing samples are quite close (Table II);

TABLE II
Tests on Shear Fatigue Strength in MPa of TF Coil Insulation (*)

Years of Test	at 20°C	at 70°C
1977 (Static)	20 – 60	16 – 24
1997 (Static)	21 – 50 (**) 18 – 54(***)	25 – 40(**) 28 – 36(***)
1997 (Fatigue)	26(***) 20,000cy	20(***) 20,000cy

(*) Each Test performed on several samples
(**) Performed at ENEA Laboratories (I)
(***) Performed at Oxford Brookes University (UK)

c) extract a number of brazed joints from the coil, to perform fatigue tests: the fatigue tensile strength of the brazed joints also compare satisfactorily with the ones obtained with the manufacturing samples (Table III).

TABLE III
Tests on Shear Fatigue Strength in MPa of TF Coil Insulation (*)

Years of Test	Test Static	Fatigue Tests(*)	
		20,000 cycles	250,000 cycles
1979	176 – 200	–	150(*)
1997	155 – 210	150	141(**)

(*) No failure of ant sample
(**) Predicted

Moreover a second faulty coil and the spare new coil still available were tested to compare deflections under forces simulating the actual load in operation. The stiffness of the used coil is only ~8% lower than the one on the spare coil (Fig. 7).

All these tests are not fully completed. However, the results obtained so far support increasing the magnetic field to 4T.

4.1.3. FE computer modelling and analysis

Preliminary tests to measure the G on coil samples shows a G~1200MPa at 70°C. These results are very important, as a lower G would reduce the shear stress in the glass-fibre epoxy resin. Therefore, more accurate tests to measure G, using the Iosipescu method will be performed. In fact, the coils were designed assuming a much larger G~4000MPa.

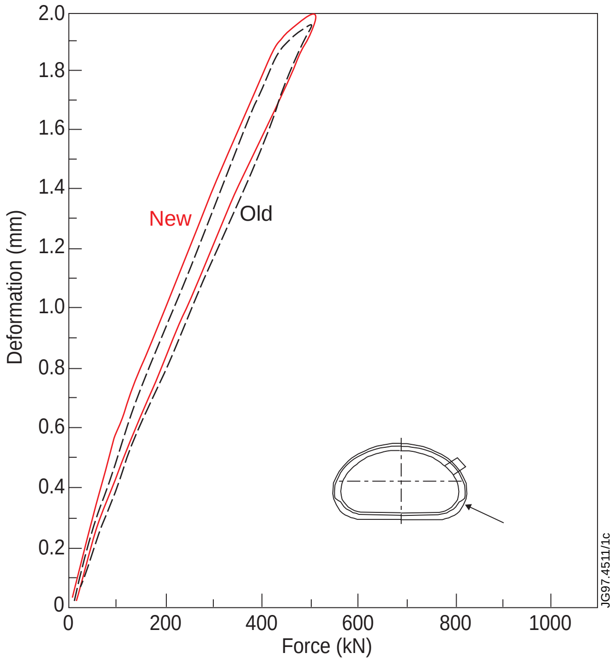

Fig. 7. Comparison of force deformation between a new and an used toroidal coil, showing a difference in stiffness of ~8%

A hybrid brick-beam model has also been developed and validated: the coil is modelled with bricks (fine mesh) for a sufficient length in the collar region (~600mm), while the rest of the coil is modelled with beams (coarse mesh). This model is the main tool to calculate the stress distribution in the collar region. The small size mesh in the collar region provides an high resolution stress picture, while the beam elements reproduce accurately the global stiffness of the coil and provide realistic boundary conditions for the bricks (Fig. 8). The ability of the model to simulate the properties of the glass-fibre epoxy separately from the copper properties resulted in more realistic predictions of the peak epoxy shear stress. In addition, by using a lower G value (1,200MPa), the calculated peak stress reduces further. Results indicate that a reference reaction force of 500kN would correspond to a peak interturn stress of ~10MPa and a peak interpancake stress of 7MPa, which are both very safe.

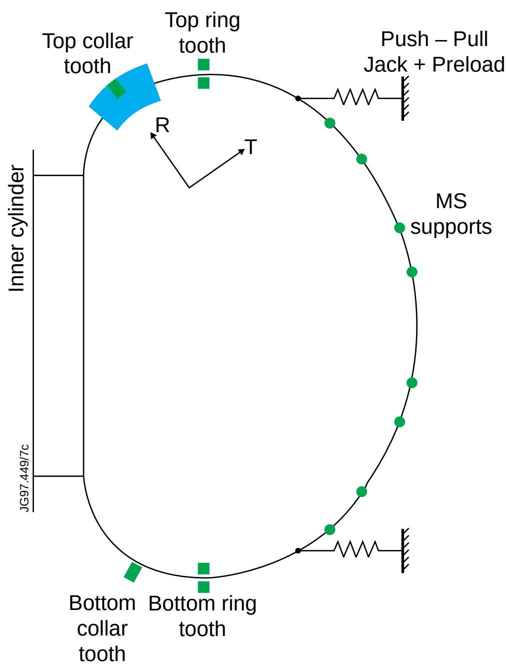

Fig. 8. Schematic of the hybrid brick-beam toroidal coil model to calculate stress distribution in the collar region

Another area of concern is the interpancake transition element. Several model have been developed to determine the copper and the butt brazed joint tensile stresses, and the epoxy shear stresses up to 4T. These models gave consistent results, showing acceptable stresses.

Finally, a safety factor >2 has been calculated for the mechanical structure, while the most critical areas, the collar and the ring teeth, have been assessed in great detail. A fatigue limit of ≥50,000 pulses has been found for the ring teeth material (in equilibrium), while ≥1,000 disruptions can be accepted for the ring teeth bolts for 4.0T, 5.0MA[24].

4.2. The vacuum vessel

Unlike the TF coils, where forces and stresses mainly arise in equilibrium, in the vacuum vessel major forces and stresses arise as a consequence of plasma disruptions. Therefore, the allowable stresses can be set for a limited number of pulses.

During disruptions the vacuum vessel demonstrates essentially two kinds of oscillatory movements: a rolling motion at ~15Hz around the toroidal axis and a sideways motion at ~5Hz. The former is caused mainly by the vertical forces generated by the currents induced in the vacuum vessel during VDE, which can be as high as several MN and scale with I_p^2 [25,26]. A lumped-parameter model and a FE model are used to assess the effects of this force. The first allows evaluation of the large scale dynamic structural response and an estimate of the forces that have caused the observed displacement. The FE model allows detail modelling of the geometry of the vessel and its supports and it is used to calculate the stress distribution in critical areas.

The sideways motion of the vessel, is caused by a net radial force. The maximum force and the consequent displacement scale essentially as $B_T I_p$ (see Fig. 4), and it is transferred from the plasma to the vessel by eddy and halo currents.

The most critical area is predicted to be the weld joining the Main Vertical Ports (MVPs) with the outer vessel wall. Fatigue testing of the material and detailed computer analysis have given the relationship between measured displacements, applied loads and number of cycle to failure.

4.2 Power supplies

The TF coils are supplied by a series connection of a flywheel generator with diode converters and two static units with thyristor converter, supplied directly from the 400kV Grid. More energy and voltage are required to produce and sustain a 4.0T magnetic field for about 10s. A detail assessment of the power supplies indicated that basically only the transformers of the static units and the cable and thyristor protection fuses, had to be replaced. Therefore, two new thyristor transformers with higher secondary voltage capability, have being purchased and will be installed by Summer 1998. However, JET is operating already at higher field, namely 3.8T with 5s flat top.

5. CONCLUSIONS

The main conclusions are summarised as follows:

- The flexibility and the engineering margins of the JET original design have allowed implementation of modifications and upgradings as suggested by the findings of the experimental programme.
- Early upgradings and modifications have been the increasing of the plasma current up to 7.0MA, the establishing of the X-point magnetic configuration up to 5.0MA, and the covering of the first wall either with CFC tiles or with Beryllium tiles.
- In this configuration, the main plasma parameters , temperature T, density n and energy confinement time t_E, individually exceeded the reactor values while in combination, a Q_{DT} equivalent exceeding unity was achieved; moreover the first ever controlled thermonuclear experiments in D-T, produced 1.7MW of fusion power.
- The need for an active control of the impurity influx, required an axisymmetric pumped divertor to be included, and this was installed without replacing any of the major components of the machine.
- Further progress in performance can be achieved by increasing the toroidal magnetic field to 4.0T; extensive work, both experimental and computational has been performed indicating that plasma scenarios at 4.0T, 5.0MA are feasible and safe, without modifying the machine.
- Due the non-availability of the new TF power supplies, JET operation is limited, at present to 3.8T.
- A world record value of 12.9MW of fusion power has been reached already in the hot-ion mode scenario.
- If JET is extended beyond 1999 and funds made available, the additional heating power could be substantially increased and new machine configurations could be implemented and tested.
- Fifhteen years of operational history clearly indicate that JET is an experimental tool that can be rejuvenated, to meet the progress in understanding of fusion physics for several years to come.

ACKNOWLEDGEMENT

This paper summarises some key engineering work performed by the JET Team. However, I wish to especially thank my colleagues that have provided me with written material, have discussed with me several relevant subjects and have given valuable help in the editing of the paper: A. Kaye, B. Keen, J. Last, F. Milani and the staff of the Engineering Analysis Group, S. Papastergiou, M. Buzio, P. Miele, P. Noll and V. Riccardo.

REFERENCES

[1] M. Keilhacker and the JET Team, JET D-T experiments and their implications for ITER, this Symposium

[2] M. Huguet et al, The JET machine: design, construction and operation of major subsystems, Fusion Technology 11 (1987) 43.

[3] A. Tanga et al, Experimental studies in JET with magnetic separatrix configuration, Nuclear Fusion 1 (1987) 65.

[4] M. Keilhacker and the JET Team, The JET H-mode at high current and power levels, Nuclear Fusion 1 (1989) 159.

[5] The JET Team, Fusion energy production from deuterium-tritium plasma in the JET Tokamak, Nuclear Fusion 32 (1992) 187.

[6] E. Bertolini (for the JET Team), JET with a pumped divertor: design, construction, commissioning and first operation, Fusion Engineering and Design 30 (1995) 53-66.

[7] J. Jacquinot (for the JET Team), Features of JET plasma behaviour in two different divertor configurations, Proc. of the 16th IAEA Fusion Energy Conference, Montreal (Canada) 7-11 Oct. 1996.

[8] M. Pick (for the JET Team), JET's latest technical and scientific results and engineering development, Proc. of the 12th Topical Meeting on Technology of Fusion Energy, Reno (Nev, USA) 16-20 June 1996.

[9] A. Rolfe, Remote handling on fusion experiments, Proc. of the 19th Symposium of Fusion Technology, Lisbon, 16-20 September 1996.

[10] M. Lennholm et al, Plasma vertical stabilisation at JET using gain adjustment, Proc. of the 17th Symposium on Fusion Engineering, San Diego (Cal, USA) 6-10 October 1997

[11] F. Milani, JET, private communication, September 1997.

[12] E. Bertolini et al, Engineering analysis of JET operations, Proc. of the 16th Symposium on Fusion Engineering, Champaign (Ill, USA) 1-5 October 1995.

[13] R.S. Graenitz et al, Nuclear Fusion, vol 36, No 5 (1996) 545.

[14] V. Riccardo et al, Asymmetric vertical displacement events at JET, Proc. of the 17th Symposium on Fusion Engineering, San Diego (Cal, USA) 6-10 October 1997.

[15] J. Last et al, Analysis and tests in support of upgrading the JET toroidal field to 4 Tesla, Proc. of the 15th International Conference on Magnet Technology, Beijing (China) 20-24 October 1997.