JOINT EUROPEAN TORUS JET

JET JOINT UNDERTAKING

ANNUAL REPORT 1991

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EUR-JET-AR14

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Preface

1991 was a highly successful year for the project. By the end of the year, JET had completed Phase III of the planned operational programme on Full Power Optimization studies. This marked the fulfilment of two of the main objectives set out in the original JET design, namely, demonstrating effective heating methods in near reactor conditions and the scaling of plasma behaviour as parameters approach the reactor range. Substantial progress has also been made towards achieving the other major objectives of studying plasma wall interactions in near reactor conditions and studying alpha-particle production, confinement and consequent plasma heating. The first experiment with a deuterium-tritium fuel in a tokamak is testimony to this.

The scientific results obtained in 1991 showed that improved plasma performance had been obtained with the new facilities available to JET. Energy confinement times of more than one second have been consistently obtained. In the high mode of plasma energy confinement (H-mode), this containment has been maintained throughout a pulse length of 18 seconds. There was a further increase in the triple fusion product of plasma density, plasma temperature and confinement time, obtained simultaneously, and "breakeven" conditions for the plasma have been achieved with deuterium plasmas. The simultaneous conditions inside JET have consequently approached within a factor of five those required in a fusion reactor. JET is the only machine in the world to have achieved this.

Experiments on high performance fusion plasmas using a deuterium-tritium fuel mixture were carried out in JET in November 1991 producing, for the first time, a significant amount of power form controlled nuclear fusion reactions. Experimental performance had reached the stage at which it was justified to introduce up to 10% of tritium into the machine, although ultimately about 50% tritium will be used in a reactor. The peak fusion power generated reached about 1.7 Megawatts in a pulse lasting for 2 seconds, giving a total energy release of about 2 Megajoules. This was a major step forward in the development of fusion as a new source of energy, and should permit extrapolation to a Next Step device which should demonstrate ignition in a routine way.

However, these results were obtained in a transient situation, limited by the influx of impurities. Plasma impurities still present a problem for steady-state operation which is required for a fusion reactor. The prolongation of JET to the end of 1996 has been approved and new experiments are planned with a pumped divertor configuration which will address these problems in operation conditions close to those of a Next Step tokamak. JET has now entered a major shutdown period when the necessary machine changes to undertake these studies will be implemented.

Also, JET has carried out preliminary studies on very long pulse operation which is essential for reactor. Using inductive drive, stable conditions have been maintained in the JET apparatus for periods of up to one minute with a 2MA plasma current. An alternative technique for maintaining long pulse operation is to use alternating current (AC) operation. JET has demonstrated reliable AC operation at plasma currents up to 2MA, in which two plasma cycles have exhibited equivalent parameters.

On the wider international front, JET has been involved in the preparatory work for the setting up of the International Thermonuclear Experimental Reactor (ITER) Project, an international Next Step, between four partners; the European Community, USA, Japan and the Russian Federation. The ITER Conceptual Design Activities have been completed and the ITER Engineering Design Activities (EDA) are being set up. Within the draft agreement for the ITER-EDA, supporting physics and technology programmes and some design work will be undertaken by the four partners under Task Agreements. The JET programme is expected to make unique and essential contributions to the EDA.

Due to severe budgetary constraints, the Project must continue to achieve economies whenever possible, during coming years, to ensure successful completion of its programme. This will involve restricted scope of its programmed, reduced staffing levels and intensified economies in operational expenditure.

The remarkable success of JET is a tribute to the continuing support which the Project obtains from the European Commission, the Host Organisation, (the UK Atomic Energy Authority) and Associations within the European Fusion Programme. I am greatly indebted to my colleagues on the JET Council and to the members of the JET Executive Committee and the JET Scientific Council for their assistance and guidance. In particular, I would like to thank those members who retired from those Committees during the past year, for their sustained commitment.

I wish to congratulate Paul-Henri Rebut, the Director of the Project, his Associate Directors and all JET staff for their outstanding achievements during the past year. JET must continue, in spite of budgetary constraints, to progress along the path towards simulating the operation of a fusion reactor and providing results which are crucial to the preparation of the Next Step. I have no doubt that the Project will successfully meet the challenges that lie ahead of it between now and the end of 1996, when the Project is scheduled to end, and in so doing, will help to ensure that nuclear fusion will become an important source of energy for mankind.

P. Fasella
Chairman of the JET Council
May 1992

Introduction, Summary and Background

Introduction

The Joint European Torus is the largest project in the coordinated fusion programme of the European Atomic Energy Community (EURATOM), whose long term objective is the joint creation of safe environmentally sound prototype fusion reactors.

The Statutes setting up the JET Project include a requirement for an Annual Report to be produced which:

> *'... shall show the current status of the Project, in particular with regard to timetables, cost, performance of the scientific programme and its position in the Euratom Fusion Programme and in the world-wide development of fusion research.'*

This report is designed to meet this requirement. It provides an overview of the scientific, technical and administrative status of the JET programme, which is intended to be comprehensible to the average member of the public. Where appropriate, descriptive sections (in italics and boxed) are included to aid the reader's understanding of particular technical terms used throughout the Report.

A more detailed and comprehensive description of the technical and scientific aspects of the JET Project for the same period can be found in the JET Progress Report.

Report Summary

The Report is essentially divided into two main parts:

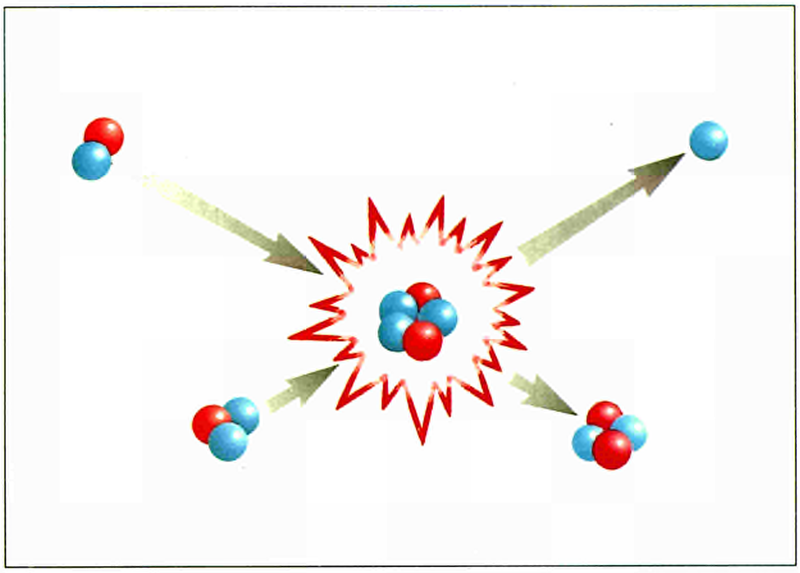
- the scientific and technical programme of the Project;
- the administration and organization of the Project.

The first part of the Report includes a brief general introduction, provides an overview of the planning of the Report and sets the background to the Project. This is followed by a description of JET and the Euratom and International Fusion Programmes, which summarise the main features of the JET apparatus and its experimental programme and explains the position of the Project in the overall

Nuclear Fusion

Energy is released when the nuclei of light elements fuse or join together to form heavier ones. The easiest reaction to achieve is that between the two heavy isotopes of hydrogen-deuterium and tritium.

Most of the energy released in this reaction is carried away by a high speed neutron. The remaining energy goes to the alpha-particle (helium nucleus, ^4He) which is also produced in the reaction. In a fusion reactor, a jacket or blanket around the reactor region would slow down the neutrons, converting their energy into heat. This heat could be extracted to raise steam for conventional electricity generation.



Fuels

As deuterium is a common and readily separated component of water, there is a virtually inexhaustible supply in the oceans of the world. In contrast, tritium does not occur naturally in any significant quantities and must be manufactured. This can be achieved by using reactions that occur between neutrons formed in the fusion reactions and the light metal lithium.

Therefore, although the fusion reactions occurring in a reactor will be between deuterium and tritium, the consumables will be deuterium and lithium.

Fusion Reaction $D + T \rightarrow {}^4He + n$

Tritium Breeding Reactions $^6Li + n \rightarrow T + {}^4He$

$^7Li + n \rightarrow T + {}^4He + n$

There are sufficient reserves of lithium available to enable world electricity generation using fusion reactors, to be maintained at present levels, for several hundreds of years.

Euratom programme. In addition, this relates and compares JET to other large fusion devices throughout the world and confirms its pre-eminent position in fusion research.

The following section reports on the technical status of the machine including: technical changes and achievements during 1989; details of the operational organisation of experiments and pulse statistics; and progress on enhancements in machine systems for future operation. This is followed by the results of JET operations in 1991 under various operating conditions, including ohmic heating, radio-frequency (RF) heating, neutral beam (NB) heating and various combined scenarios in different magnetic field configurations; the overall global and local behaviour observed; and the progress towards reactor conditions. In particular, the comparative performance between JET and other tokamaks, in terms of the triple fusion product, shows the substantial achievements made by JET since the start of operations in 1983. This section concludes with a discussion of future scientific prospects. The scientific part of this Report concludes with a description of the proposed future programme of JET until its planned conclusion.

The second part of the Report explains the organisation and management of the Project and describes the administration of JET. In particular, it sets out the budget situation; contractual arrangements during 1991; and details of the staffing arrangements and complement.

Background

In the early 1970's, discussions were taking place within the European fusion research programme on a proposal to build a large tokamak fusion device to extend the plasma parameters closer to those required in a reactor. In 1973, agreement was reached to set up an international design team which started work in the UK later that year, and by the middle of 1975, the team had completed its design for a very large tokamak device.

On 30th May 1978, the Council of Ministers of the European Communities decided to build the Joint European Torus (JET) as a Joint Undertaking of the European Fusion Programme. To implement the Project, the JET Joint Undertaking was originally established for a duration of 12 years, beginning on 1st June 1978.

It was decided that the device would be built on a site adjacent to the Culham Laboratory, the nuclear fusion research laboratory of the United Kingdom Atomic Energy Authority (UKAEA), and that the UKAEA would act as Host Organisation to the Project. Figure.1 shows the site of the JET Joint Undertaking at Culham, near Oxford in the U.K.

The Members of the JET Joint Undertaking are Euratom, its Associated Partners in the framework of the Fusion Programme, including Sweden (NFR) and Switzerland, together with Greece, Ireland and Luxembourg, who have no Contracts of Association with Euratom.

Eighty per cent of the expenditure of the Joint Undertaking is borne by Euratom. As the host organisation, the UKAEA pays ten per cent, with the remaining ten per cent shared between Members having Contracts of Association with Euratom in proportion to the Euratom financial participation in the total costs of the Associations.

The Project Team is formed mainly by personnel drawn from the Associated Institutions, although some staff are assigned on a secondment basis from the Institutions and the Directorate General of the Commission responsible for Science Research and Development (DGXII).

In July 1988, the Council of Ministers agreed the prolongation of the JET Joint Undertaking to 31st December 1992. A further proposal to prolong JET to 31st December 1996 was submitted to the Council of Ministers in October 1990. At its meeting on 19th December 1991, the Council of Ministers adopted Decisions concerning the Euratom Fusion Programme in the period to the end of 1994 and a modification to the JET Statutes which prolonged its statutory lifetime by four years until 31st December 1996. The extension will allow JET to implement the new Pumped Divertor Phase of operation, the objective of which is to establish the effective control of plasma impurities in operating conditions close to those of the Next Step. This programme of studies will be pursued before the final phase of full D-T operations in JET.

Conditions for Fusion

Fusion reactions can only take place if the nuclei are brought close to one another. But all nuclei carry a positive charge and therefore repel each other. By heating the gaseous fuels to very high temperatures, enough energy can be given to the nuclei for the repulsive force to be overcome sufficiently for them to fuse together. In the case of the deuterium-tritium reaction, temperatures in excess of 100 million degrees Kelvin are required - several times hotter than the centre of the sun. Below 100 million degrees, the deuterium-tritium reaction rate falls off very rapidly: to one-tenth at 50 million degrees, and 20,000 times lower at 10 million degrees.

A reactor must obtain more energy from the fusion reactions than it puts in to heat the fuels and run the system. Reactor power output depends on the square of the number (n_i) of nuclei per unit volume (density) and the volume of gas.

Power losses must also be kept to a minimum acceptable level by holding the hot gases in thermal isolation from their surroundings. The effectiveness of this isolation can be measured by the energy confinement time (τ_E) - the time taken for the system to cool down once all external forms of heating are switched off.

In a fusion reactor the values of temperature, density and energy confinement time must be such that their product ($n_i\tau_E T_i$), exceeds the figure of $5\times10^{21}m^{-3}$skeV. Typical values for the parameters that must be attained simultaneously for a reactor are:

Central ion temperature, T_i	*10-20keV*
Central ion density, n_i	*$2.5\times10^{20}m^{-3}$*
Energy confinement time, τ_E	*1-2s*

The temperature is expressed as the average energy of the nuclei (1keV is approximately equal to 10 million degrees K).

Fusion Reactor

In a fusion reactor a lithium compound would be incorporated within a blanket surrounding the reactor core so that some neutrons can be utilised for manufacturing tritium. The tritium produced would then be extracted for use in the reactor.

The blanket would also provide the means of utilising the energy carried away from the reactions by the neutrons. As the neutrons are slowed down within the blanket, its temperature would rise thus enabling steam to be raised so that electricity could be generated in the conventional manner.

Ultimately, it is hoped that the conditions would be reached to enable a reactor to be built utilising the deuterium-deuterium reactions below:

$$D + D \rightarrow {}^3He + n$$

$$D + D \rightarrow T + p$$

In this case there would be no need to manufacture tritium and a virtually inexhaustible reserve of energy would become available.

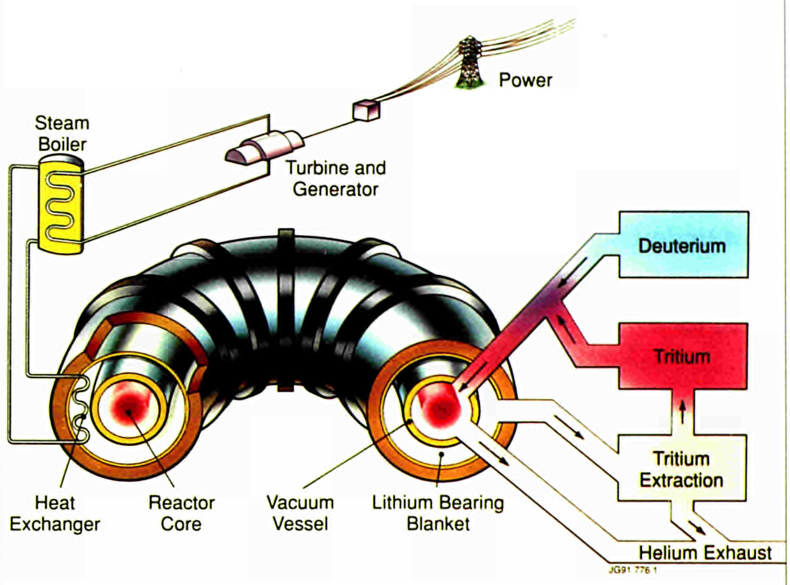
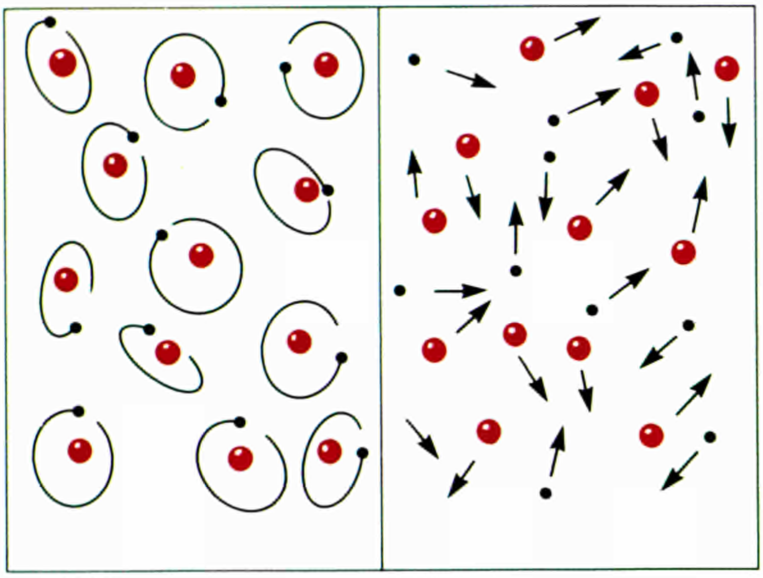


Fig.1: The site of the JET Joint Undertaking, near Oxford in the United Kingdom.



Plasma

As the temperature of the fuel is increased, the atoms in the gas become ionised, losing their electrons, which normally orbit around the nuclei. The mixture of positively charged ions and negatively charged electrons is very different from a normal gas and is given a special name - PLASMA.

The fact that a plasma is a mixture of charged particles means it can be controlled and influenced by magnetic fields. With a suitably shaped field it should be possible to confine the plasma with a high enough density and a sufficiently long energy confinement time to obtain net energy gain.

The configuration that has so far advanced furthest towards achieving reactor conditions and on which most data is available is the TOKAMAK, originally developed in the U.S.S.R.

Objectives of JET

The original decision of the Council of Ministers in 1978 states that the JET Joint Undertaking's mandate is to:

> *'... construct, operate and exploit as part of the Euratom fusion programme and for the benefit of its participants in this programme, a large torus facility of tokamak-type and its auxiliary facilities in order to extend the parameter range applicable to controlled thermonuclear fusion experiments up to conditions close to those needed in a thermonuclear reactor.'*

The principal objective of JET is to enable the essential requirements of a tokamak reactor to be defined. To implement this, it was necessary to create and study plasma in near-reactor conditions.

There are four main areas of work:

1. the study of scaling of plasma behaviour as parameters approach the reactor range;
2. the study of plasma-wall interaction in these conditions;
3. the study of plasma heating;
4. the study of alpha-particle production, confinement and consequent plasma heating.

In addition, JET is pioneering two of the key technologies that will be required in subsequent fusion reactors. These are the use of tritium and the application of remote maintenance and repair techniques.

JET, Euratom and other Fusion Programmes

The Joint European Torus

JET uses the tokamak magnetic field configuration to maintain isolation between the hot plasma and the walls of the surrounding vacuum vessel. A diagram of the JET apparatus is shown in Fig.2 and the principal design parameters are presented in Table I.

The toroidal component of the magnetic field on JET is generated by 32 large D-shaped coils with copper windings, which are equally

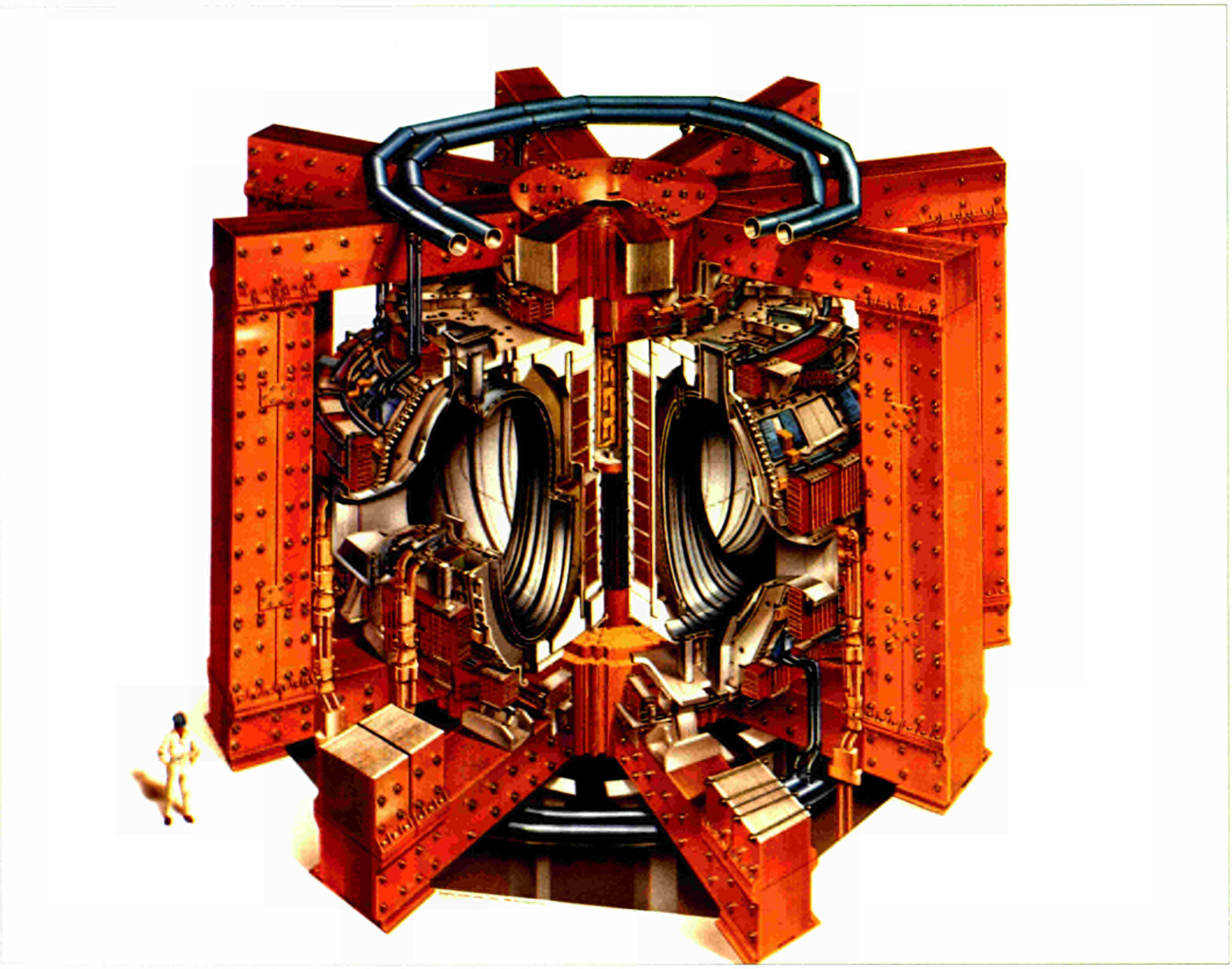


Fig.2: Diagram of the JET apparatus.

Table I: Original Design Parameters of JET

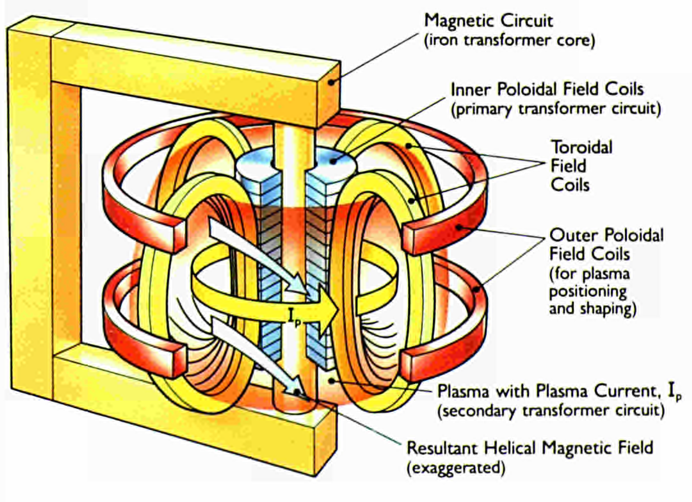
Plasma minor radius:	
horizontal	1.25m
vertical	2.10m
Plasma major radius	2.96m
Flat-top pulse length	20s
Weight of the iron core	2800t
Toroidal field coil power (peak on 13s rise)	380MW
Toroidal magnetic field at plasma centre	3.45T
Plasma current:	
circular plasma	3.2MA
D-shape plasma	4.8MA
Volt-seconds available to drive plasma current	34Vs
Additional heating power	25MW

spaced around the machine. The primary winding (inner poloidal field coils) of the transformer, used to induce the plasma current which generates the poloidal component of the field, is situated at the centre of the machine. Coupling between the primary winding and the toroidal plasma, acting as the single turn secondary, is provided by the massive eight limbed transformer core. Around the outside of the machine, but within the confines of the transformer limbs, is the set of six field coils (outer poloidal field coils) used for shaping and stabilising the position of the plasma inside the vessel.

During operation large forces are produced due to interactions between the currents and magnetic fields. These forces are constrained by the mechanical structure which encloses the central components of the machine.

Magnetic Field Configuration

The tokamak magnetic field configuration is built up from three components. The first of these is produced by a set of coils around the minor circumference. These coils produce the toroidal magnetic field around the major axis of the machine. The second component (poloidal field) is produced by a large current caused to flow through the plasma by transformer action. The combination of these produces a helical magnetic field which keeps the plasma away from the vessel walls. The final component is generated by a set of hoop coils, which is used to shape and stabilise the position of the plasma.



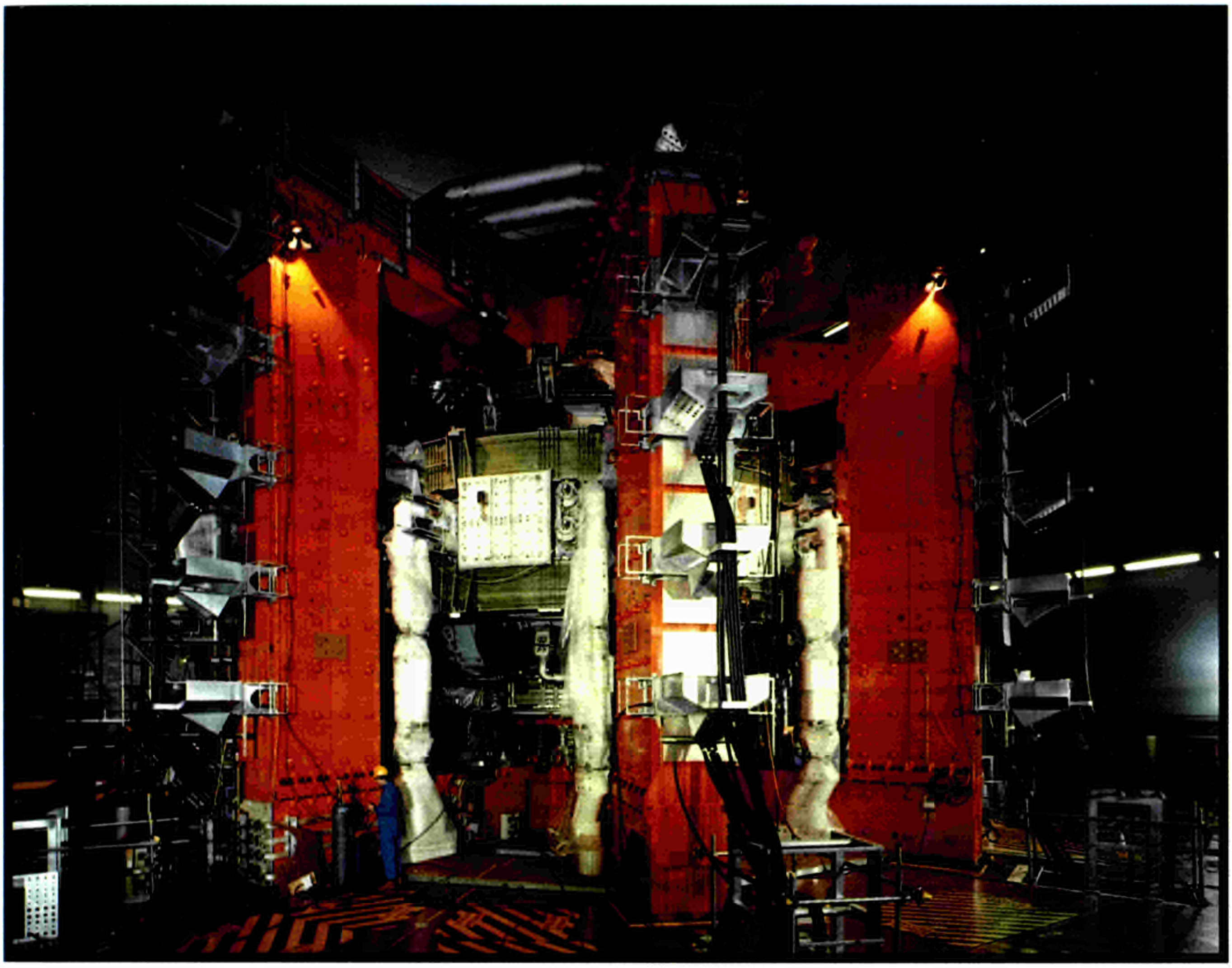


Fig.3: The JET experimental apparatus photographed in May 1983

The use of transformer action for producing the large plasma current means that the JET machine operates in a pulsed mode. Pulses can be produced at a maximum rate of about one every ten minutes, with each one lasting up to 60s. The plasma is enclosed within the doughnut shaped vacuum vessel which has a major radius of 2.96m and a D-shaped cross-section of 4.2m by 2.5m. The amount of gas introduced into the vessel for an experimental pulse amounts to less than one tenth of a gram.

The construction phase of the Project, from 1978 to 1983, was completed successfully within the scheduled five year period and within 8% of projected cost of 184.6 MioECU at January 1977 values.

The first plasma pulse was achieved on 25 June 1983 with a plasma current of 17000A lasting for about one tenth of a second. The JET Tokamak is shown in Fig.3 just prior to the start of operation in June 1983. This first phase of operation was carried out using only the large plasma current to heat the gas. In 1985, the first additional heating system, employing radio-frequency heating, came into operation and during 1991 reached 22MW of power into the plasma. The neutral beam heating system was brought into operation in 1986, and exceeded its design capability in 1988, with 21.6MW of power injected into the torus.

Experiments have been mainly carried out using hydrogen or deuterium plasmas, although during 1991, experiments were carried out in helium-3 and helium-4 and a preliminary experiment was performed using up to 10% tritium in deuterium. In the final stage of the programme, it is planned to operate with deuterium-tritium plasmas so that abundant fusion reactions occur. The alpha-particles liberated from the reactions should then produce significant heating of the plasma. During this phase of operation, the machine structure will become radioactive to the extent that any repairs and maintenance would have to be carried out using remote handling systems.

Heating

Initial production and heating of the plasma is produced by the large electric current flowing in the plasma itself (ohmic heating) used to generate the poloidal magnetic field.

The heating effect of this current is reduced as the plasma gets hotter as the electrical resistance of the plasma decreases with increasing temperature. Therefore, it is necessary to provide additional means of heating if the temperatures needed for a reactor are to be reached.

Two main additional heating methods are in general use:

(1) Neutral Beam Heating: In this method, a beam of charged hydrogen or deuterium ions is accelerated to high energies and directed towards the plasma. As charged particles cannot cross the magnetic field confining the plasma, the beam must be neutralised. The resulting neutral atoms cross the magnetic field and give up their energy through collisions to the plasma, thereby raising its temperature.

(2) Radio Frequency Heating: Energy can be absorbed by the plasma from high power radio-frequency waves. The frequency of operation is chosen to be close to that at which the ions or electrons orbit or gyrate in the magnetic field.

The Community Fusion Research Programme

All fusion research in Europe is integrated into one single Community Programme. Successive programme decisions by the Council of Ministers have described the Community Fusion Programme as "a long-term co-operative project embracing all the work carried out in the Member States in the field of controlled thermonuclear fusion". The long-term objective of the Programme is the joint creation of safe, environmentally sound prototype fusion reactors.

The Commission of the European Communities is responsible for the implementation of the Fusion Programme. It is assisted in this task by the Consultative Committee for the Fusion Programme (CCFP), composed of national representatives. The programme is implemented principally through Associations with organisations in the Member States (and Sweden and Switzerland), the JET Joint Undertaking, the Agreements concerning the Next Step after JET (be it NET in the European frame or ITER in a wider international frame), and cost-shared contracts in the countries having no Associations. The Joint Research Centre of the Community, through its own programme, conducts research on specific aspects of fusion technology in close coordination with the Fusion Programme.

This Community approach has led to extensive collaboration between the fusion laboratories. For example, most of the Associations undertake work for other Associations. The Associations are partners in JET and NET and undertake work for them through various types of contracts and agreements. The Community Fusion Programme has built across Europe a genuine scientific and technical community of large and small laboratories, readily able to welcome newcomers, and directed towards a common goal. Indeed, two non-Member States, Sweden and Switzerland, are fully associated with the Programme and enjoy the same rights and responsibilities as Member States.

A step by step strategy towards a Prototype Commercial Reactor is envisaged, including after JET, an Experimental Reactor (Next Step) and a Demonstration Reactor (DEMO). As far as the Community is concerned, the results from JET and from the medium-sized devices in the Associated Laboratories allow the definition, with confidence, of the plasma size and the plasma parameters of the Next Step, the major physics goal of which is to achieve self-sustained thermonuclear burn of a deuterium-tritium plasma and its control during long pulse operation. The Next Step should demonstrate the safe operation of a device which integrates important technologies of a fusion reactor, and should test components and subsystems essential for a fusion reactor. The Next Step should provide the basic data for the engineering of a demonstration fusion reactor (DEMO) capable of producing significant amounts of electricity while taking due account of environmental constraints.

In the frame of the Third Framework Programme of Community Activities in the field of Research and Technological Development (1990 to 1994), the recent Council Decision (19th December 1991, OJ No. L375 of 31 December 1991) adopting the 1990-94 Fusion Programme states that the first priority objective of the Programme is to provide the scientific and technological base, to establish the environmental and safety criteria, and to prepare

TABLE 2: Large Tokamaks operating around the World

Machine	Country	Minor radius a(m)	Elongation κ	Major radius R(m)	Plasma current I(MA)	Toroidal field B(T)	Input power P(MW)	Start Date
JET	EC	1.20	1.8	2.96	7.0	3.5	36	June 1983
JT-60	Japan	0.90	1.0	3.0	3.2	4.8	25	April 1985
TFTR	USA	0.85	1.0	2.50	3	5.2	32	Dec. 1982
Tore-Supra	France	0.70	1.0	2.4	1.7	4.5	23	April 1988
T-15	USSR	0.70	1.0	2.4	2.0	4.0	-	Jan. 1989
DIII-D	USA	0.67	2.0	1.67	3.5	2.2	20	Feb. 1986
ASDEX-U	Germany	0.5	1.6	1.65	(2)	3.9	(15)	April 1991
FT-U	Italy	0.31	1.0	0.92	1.6	8.0	-	Dec. 1988

TABLE 3: Plasma Parameters of Large Tokamaks

Parameter		JET (H-mode)	TFTR (Hot-ion Mode)	DIII-D (VH-mode)
Electron Temperature	T_e(keV)	11	12	6
Ion Temperature	T_i(keV)	19	30	7
Central ion density	n_i($\times 10^{19}$m^{-3})	2.9	6.7	8.0
Effective charge	Z_{eff}	1.8	2.5	1.2
Confinement time	τ_E(s)	1.8	0.16	0.23
Neutral beam power	P_{NB}(MW)	15	25	12.6
Fusion product ($\times 10^{20}$m^{-3}keVs)	$n_i T_i \tau_E$	>9	3.2	1.3

industry for the construction of a Next Step tokamak device. For this purpose, a large fraction of the 1990-94 activities, including those on JET and within the Associations, will be in support of the Next Step. This is also the reason why, together with the Programme Decision, the Council adopted a Decision on a prolongation of the JET Joint Undertaking to the end of 1996, in order to establish reliable methods of plasma purity control in conditions relevant to the Next Step.

The scientific and technical achievements of the Community Fusion Programme place Europe at the forefront of world fusion research. JET, which is the leading fusion experiment in the world, has made substantial progress towards demonstration of the scientific feasibility of fusion. The first Tokamak discharges in deuterium-tritium fuelled mixtures on JET in November 1991, leading to the production of fusion power in the megawatt range for a few seconds, constitute an important step forward in fusion research. A contribution towards this success has been due to research carried out in the Associated Laboratories such as the discovery of the H-mode on ASDEX (IPP Garching, FRG) and developments in plasma heating systems (AEA, Culham Laboratory and CEA, France).

Currently, expenditure on fusion research through the Community budget is running at the rate of ~200 MioECU per year. When funding by national administrations and other national bodies is taken into account, expenditure on fusion from all sources in Europe exceeds 450 MioECU per year. About 1750 professional scientists and engineers are currently engaged in fusion research in Europe.

The leading position of the Community Fusion Programme continues to make Europe an attractive partner for international collaboration. For example, bilateral Framework Agreements have now been concluded with Japan, USA and Canada. Similarly, one is under negotiation with the Russian Federation. There are also eight specific Implementing Agreements in the framework of the International Energy Agency (OECD), including cooperation among the three large tokamak facilities (JET in Europe, JT-60 in Japan and TFTR in the USA).

However, the most far-reaching development in international collaboration is in connection with the International Thermonuclear Experimental Reactor (ITER) Project. Following initiatives taken at the highest political level since 1985, the European Community, Japan, the USA and the USSR agreed in early 1988 to participate on an equal quadripartite basis in the joint development of a conceptual design for an experimental reactor of the tokamak type. After the ITER Conceptual Design Activities (CDA) were concluded successfully at the end of 1990 (as expected), a period of negotiations followed in 1991, which culminated in the acceptance "ad referendum" in November of the draft for an Agreement establishing the Engineering Design Activities (EDA) of ITER. The design will be carried out by the home teams and Joint Central Teams located in three internationally-staffed co-centres in Garching in the EC, Naka in Japan and San Diego in the USA. The ITER EDA, due to last six years, should start as soon as the Agreement has been signed by the four parties (in which the former USSR is replaced by the Russian Federation). The ITER EDA will be conducted under the auspices of the IAEA, as was carried out for ITER CDA.

Large International Tokamaks

Plasma current is an important factor in determining the confinement of the plasma and, in particular, of the energetic fusion products, the alpha-particles. On a worldwide basis JET is the machine with the largest plasma current by far; followed by DIII-D, General Atomics, San Diego, USA; JT-60 at JAERI in Japan; and TFTR at Princeton Plasma Physical Laboratory (PPPL), USA. Table.2 sets out an overview of these tokamaks with their main parameters and starting dates. Other operating devices with plasma currents in excess of 1MA are also included.

Other smaller tokamaks exist throughout the world or are under construction, each dedicated to specific tasks of studying different aspects of physics, engineering and heating of plasmas. In addition, these large tokamaks have also been designed with specific tasks. For instance, in Tore Supra, France, the main magnetic field is created by superconducting coils. The various machines are also designed to test various heating systems. Ion

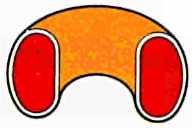
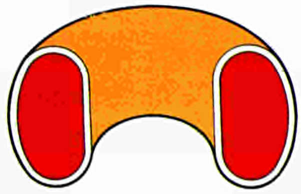
		TFTR	JET	ITER
				
Minor radius	a	0.85	1.25m	2.15
Major radius	R	2.48m	2.96m	6.0
Elongation	κ	1.0	1.8	2
Toroidal Field	B	5.0T	3.45T	4.85
Input Power	P	30MW	36MW	30-200MW
Fusion factor	Q_{DT}	0.25	1.1	30 - Ignition
Plasma current	I	3	7	22

Fig.4: Operating parameters of three large tokamak designs

Cyclotron Resonance Frequency (ICRF) and Neutral Beam (NB) heating are now commonly applied in excess of 10MW each. Electron Cyclotron Resonance Heating (ECRH) of several MW has been applied in some devices. Non-inductive current drive by means of various heating methods include Lower-Hybrid Current Drive (LHCD). LHCD has also undergone preliminary tests in JET to complement the ICRF and NB heating systems.

JET, JT-60 and DIII-D can produce magnetic field configurations, that have open magnetic surfaces within the vacuum vessel near the edge of the plasma. The plasma is then defined by a magnetic limiter and not by a material limiter in contact with the confined plasma. The magnetic limiter configuration, with its high shear, not only leads to higher edge temperatures and improved global confinement, but also permits better particle and impurity control.

1991 has seen a breakthrough in fusion research, in which JET carried out preliminary experiments using up to 10% tritium in deuterium plasmas. The tritium concentration in the discharges was kept low in order to ensure machine activation was maintained at an acceptably low level. For the first time, a significant amount of energy by fusion of deuterium and tritium nuclei was produced in a controlled thermonuclear device. Two Megajoules of energy was achieved with a peak power of 1.7 Megawatts in discharges operating in the hot-ion H-mode: a high confinement regime at relatively low density at which the ion temperature is significantly higher than the electron temperature. The confinement was ~1 second, and the ion temperature was ~20keV, with the electron temperatures of ~10keV in the plasma centre. The actual fusion amplification factorachieved, Q_{DT}, was 0.15. With an optimum tritium concentration (about 60% tritium), this pulse would have produced 5 Megawatts of power and a nominal Q_{DT} of 0.45.

The performance of fusion devices is generally characterized by a parameter termed the triple fusion product $n_i \tau_E T_i$. Here n_i is the density of the hydrogen isotope in the centre of the device, τ_E the energy confinement time and T_i the central ion temperature. Break-even in energy production is defined by $Q_{DT} = 1$, which corresponds to a triple fusion product of ~$10 \times 10^{20} m^{-3}$skeV for typical radial profiles of temperature and density in steady state conditions. Ignition occurs at values of the fusion product of 5-8 times higher, when the alpha-particle energy production balances the power losses of the plasma.

In preparation for the preliminary tritium experiment, JET has obtained high performance with the hot-ion mode in deuterium plasmas. The highest fusion product achieved was ~$10^{21} m^{-3}$skeV in conditions that would have been equivalent to a $Q_{DT} = 1.14$, which exceeds break-even conditions. This is within a factor of ~5-8 of that required in a fusion reactor (see Table 3). In addition, TFTR has reached ion temperatures of ~35keV, and a fusion product value of ~$3 \times 10^{20} m^{-3}$keV (when corrected for dilution). Projections for the designed non-thermal ion distribution in this device with higher power density, suggest Q_{DT} values of 0.4-0.7.

However, in both machines these results are still of a transient nature and have not yet been sustained in steady state. Both TFTR and JET have carried out extensive studies on the termination of the good confinement period. This termination occurs through a combination of various effects of which the most obvious is a sudden high influx of impurities (the so-called impurity 'bloom'). This 'bloom' is thought to be related to an overheating of the target plates or limiters, which catch the energy escaping from the plasma. Both machines have been able to make improvements, by machining and redesigning the target plates. During 1991, JET has further improved the performance of its target plates by a factor of two. The new divertor phase of JET's operation should improve this performance much further.

Further improvements in plasma confinement have been attained in DIII-D, which has discovered the so-called VH mode of operation. This high confinement regime shows improvements up to ~3.5 times the normal (or L-mode) confinement time. This confinement also exists in JET in the hot ion H-modes in high-beta discharges. This high confinement then leads to high ion temperatures and neutron rates. In addition, TFTR has achieved good confinement in the so-called 'supershots', in which conditions most favourable for high Q_{DT} are created.

Another important parameter for fusion is the so-called plasma beta which is the ratio of the pressure exerted by the plasma to the pressure of the magnetic fields. The highest toroidal beta of 11% has been obtained in the DIII-D tokamak. The toroidal beta is important for the economic aspects of a reactor. Typical values for the next generation devices like ITER, the International Thermonuclear Experimental Reactor, are expected to be around 5%. A number of tokamaks have carried out studies on plasmas with high pressures and large beta values. Very good confinement has been observed compared with normal (L-mode) values. Improvements of up to a factor of 4 have been obtained in the ratio of measured confinement times over the L-mode values. On TFTR, it has been observed that the confinement is independent of plasma current. All devices find that the pressure gradient driven current, (the so-called bootstrap current) drives most of the plasma current. This then leads to inverted profiles of the safety factor, q, in the plasma centre and parts of the plasma are close to a region of enhanced stability which should be beneficial for confinement at high beta.

Both JET and TFTR have reached high values of the triple fusion product. Fig 4 shows the main parameters and schematic diagrams of the cross-section of both these devices, together with the proposed next step device (ITER). However, JET has attained breakeven conditions transiently, with a calculated Q_{DT} value exceeding unity during one energy confinement time. The transient nature of the good fusion conditions is at present limited by local overheating of sections of the wall surfaces. This leads to a high influx of impurities, which terminate the good plasma conditions. A new phase is scheduled for JET, starting in 1992, to demonstrate a reactor relevant solution to this problem, that should lead to a sustained production of a few Megawatts of alpha-particle heating by 1996.

An extensive exchange of information, equipment and scientists exists within the worldwide fusion community. In particular, a collaboration agreement between the large tokamak groups in EEC, USA and Japan was signed in 1986. Subsequently, this culminated in the setting up of a collaboration between the EEC, USA, Japan and the Russian Federation (then USSR), under the auspices of the International Atomic Energy Agency (IAEA), to perform a conceptual design of an International Thermonuclear Experimental Reactor (ITER). The main objective of the ITER project is to define a single concept that should demonstrate the scientific and technological feasibility of fusion. The four ITER partners have now negotiated an agreement on the Engineering Design Activity (EDA), and the draft agreement has been submitted to Governments for ratification. Under this agreement, the parties will jointly conduct activities to establish engineering design and construction planning for a device capable of controlled ignition and extended burn. The JET programme is expected to make unique and essential contributions to the ITER EDA.

Technical Status of JET

Introduction

The present technical status of JET is described in the following three sections:

- The first section outlines details of the developments and improvements implemented during the shutdown at the beginning of 1991 and the technical achievements during the main operating period of 1991. In particular, there is some emphasis placed on the special preparations made for the preliminary tritium experiment, which took place in November 1991;
- Machine operations during the operating periods are summarized in the second section;
- The third section sets out the main details of continuing technical developments on equipment for future installation.

Technical Achievements

During early 1991, the machine was already in a scheduled shutdown. This shutdown had started in November 1990, and the main tasks were to repair and strengthen the mechanical supports for in-vessel wall protection tiles and to install target plates at the top and bottom of the vessel, for use during single and double-null X-point configurations.

In 1990, operation had been disrupted and somewhat limited due to damage to wall protection tiles. The problem was attributed to the large forces generated by the so called "halo" currents, which flow during plasma vertical displacement events and disruptions. At the end of the 1990 experimental period, 49 wall tiles had been dislodged and had fallen or were projecting into the vessel, thus restricting the size of the plasma that could be produced. During the shutdown from November 1990 to April 1991, remedial action was taken. Critical welds were systematically strengthened, cantilevered mountings were eliminated and weak mechanical supports were modified and stiffened. These measures which involved a considerable amount of work on a large number of components were remarkably successful insofar as no further damage to tile support mechanisms were observed during the 1991 campaign.

The values of "halo" currents could not be measured during the 1990 campaign. Installation of relevant current measuring devices was therefore one of the priority tasks of the shutdown. Some tiles were modified and electrically connected to the vessel wall through resistive elements, permitting measurement of the current flowing from the tile to the wall. These instrumented tiles allowed measurements at various poloidal and toroidal locations, which confirmed that the total poloidal "halo" currents flowing in the vessel walls could reach about 20% of total plasma current.

The main task of the shutdown was the installation of the X-point target plates. Each target plate consists of 48 inconel sectors firmly attached to the vacuum vessel. These sectors provide a dimensionally accurate base for fixing the tiles. Using specially designed assembly jigs, it was possible to align the sectors with an absolute accuracy of ±1.5mm. To allow a comparative assessment of carbon and beryllium as target plate material, carbon fibre composite (CFC) tiles were installed at the top target plate, and beryllium tiles at the bottom.

A disadvantage of beryllium as a target plate material is its low melting temperature making it prone to surface damage and irregularities.The beryllium tiles were carefully machined to eliminate leading edges, which would be nearly normal to magnetic field lines and therefore receive a high power density. As a result, melting at leading edges was not observed during 1991. However, this shaping resulted in a considerable reduction in effective area of the target plates.

When machine operation resumed in June, only the beryllium target plate tiles had been specially shaped. The carbon target plate tiles, although well aligned, had their leading edges and fixing holes edges exposed to high power fluxes. Operation with carbon target plates soon revealed that their performance was limited by local hot spots at leading edges. This was not observed with the specially shaped beryllium tiles. Therefore, a decision was taken in Summer 1991 to machine the carbon target plate tiles in a similar fashion to allow a meaningful comparison of the two materials. Installation took place in a short shutdown during the first half of September.

Operation of the machine and associated systems showed good reliability during 1991. The availability of systems was generally very high and fault finding and repairs were carried out speedily with a minimum of lost operation time. This must be credited to ever improving fault reporting and maintenance organisation systems. Credit must also be given to the motivation and dedication of staff who had to carry out trouble-shooting and repairs outside normal operational hours.

During 1991, a number of new systems were brought into operation and improved the performance of the machine. The Reactive Power Compensation system was commissioned smoothly and integrated rapidly into routine operation. It makes use of vacuum switches to switch capacitator banks and therefore reduce the reactive power consumption. The system operated reliably and demonstrated its usefulness in keeping the JET power consumption within limits specified in contracts with the Electricity Companies.

The poloidal field power supplies and, in particular, the vertical poloidal field system, were modified to allow the running of two successive plasma pulses with the plasma current flowing in opposite directions.This is the so-called mode of AC (alternating current) operation, and is relevant for future machines and reactor designs. The poloidal shaping circuit which controls the plasma elongation, was modified to increase its current capability from 40 to 50kA. This allowed double null X-point operation with X-points well inside the vessel, and at a plasma currents up to 4MA. A disruption detector which monitors the amplitude of MHD modes and triggers pulse termination at a defined threshold was implemented. With this system, the plasma elongation and current were ramped down in advance of the actual disruption, and the severity of the disruption, and, in particular, the intensity of forces acting on the vacuum vessel, were reduced.

The second half of 1991 was dominated by work relevant to preparation, execution and clean-up of the Preliminary Tritium Experiment. The physics, technical and safety aspects of this experiment required a project wide effort and coordination. Two Task Forces were set up for the experiment. A Physics Task Force was charged with selection and optimisation of the discharge type, choice of tritium injection method and, preparation of simulation codes and diagnostics. The Technology Task Force was responsible for design, construction and commissioning of new systems for the introduction of tritium and recovery of tritiated exhaust gases; preparation of all safety reports and documentation; to obtain necessary authorisations; and for installation and commissioning of the necessary radiological protection instrumentation. Additional tasks included the procurement of tritium and provision for its safe storage on the JET site. Preparations were also made for the safe handling of tritiated components and wastes.

Power Supplies

The electric power to the JET device during an experimental pulse is counted in hundreds of megawatts.

An agreement with the Generating Boards allows up to 575MW of pulse power to be taken directly from the 400kV grid, which after transformation down to 33kV is fed to the JET loads through a system of circuit breakers.

Two flywheel generators are used to provide the peak power for the toroidal magnetic field coils and ohmic heating circuit. Each of the generators has a rotor 9m in diameter weighing 775 tonnes. Between pulses, 8.8MW pony motors are used to increase the speed of rotation. When power is required for JET pulse, the rotor windings are energised and the rotational energy of the flywheel is converted into electrical energy. On slowing down from the maximum speed of 225rpm to half speed, the generators can reach deliver 2.6GJ of energy with a peak power output of 400MW.

In-Vessel Components

Among the new wall elements which were introduced into the vessel during the scheduled shutdown were the toroidally continuous X-point target plates, at the top and the bottom of the vacuum vessel. These replaced the provisional target plates used during 1990 which consisted of toroidally spaced energy dumps initially devised as wall protection. The new X-point target plates were each made up of 48 cylindrical segments attached to the rigid sectors of the machine covering the top and bottom over a poloidal arc of 1 m length. Fig.5 shows the upper target plate after installation. The accuracy of the mechanical alignment between two adjacent sectors was better than ± 1 mm. The active areas of the lower plates were covered with beryllium tiles, whilst the upper plates were covered with carbon fibre reinforced graphite.

The beryllium tile assemblies were designed to optimize the power handling capability of the system, whilst reducing the probability of producing hot spots leading to influxes of carbon or beryllium impurities (i.e. the so-called impurity 'bloom'). The design required the machining of each tile so that leading edges were eliminated. This ensured more even power loading of the tiles. An additional slope was machined onto all tiles at dump plate segment edges to allow for steps from sector to sector. This allowed for the maximum expected manufacturing and installation tolerances of the torus and dump plate segments.

The graphite fibre tiles were initially not shaped in this intricate way, but had flat surfaces, which were slightly raised in some areas to compensate for the cylindrical shape of the target plate segments. During operation, carbon "blooms" occurred at low incident power and re-shaping was required. The original graphite tiles on the upper target plate were replaced by modified tiles during the September shutdown.

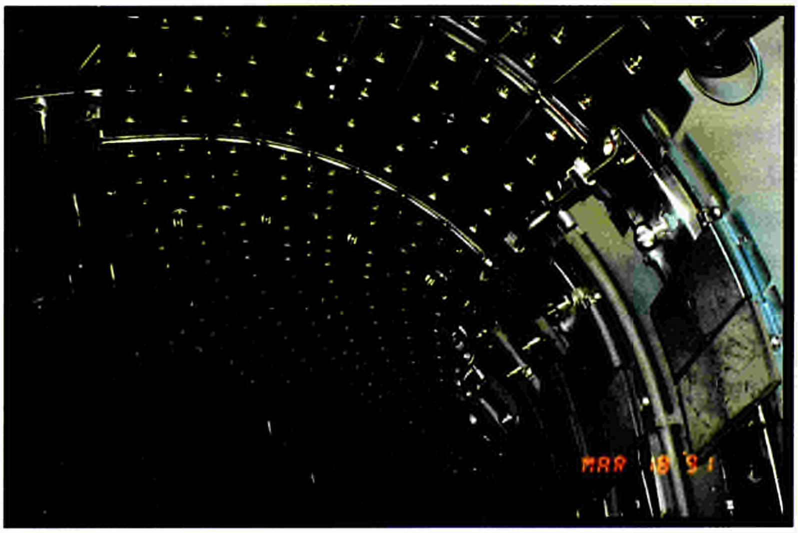


Fig.5: Upper target plate after installation;

Strengthening of Mechanical Supports

Problems experienced with the wall protection during 1990 were related to vertical plasma movement during disruptions and loss of position control. A fraction (up to 25%) of the total current was transformed into poloidal currents (halo currents), which upon contact with the wall, were then transferred to the vessel structure, including the wall protections. This is a new phenomenon, which had not been observed earlier and was related to the improved purity of the plasma. The wall components were originally designed to withstand eddy current forces due to the decay of the poloidal magnet field, but were not strong enough to cope with the forces created by the interaction of these halo currents with the toroidal field, did a series of component failures and damage to in vessel components occurred.

During the shutdown, strategically located and instrumented tiles to measure these currents were installed in the vessel. With the resulting measurements, analysis provided improved knowledge of the nature and magnitude of plasma "halo" induced currents in first wall components.

Based on this understanding, remedial action was taken to reinforce the wall protection in the affected areas. Welds were strengthened, fasteners improved, weak mechanical supports were modified and cantilevered supports were eliminated. Experience during subsequent operation proved that these modifications were successful. No further wall damage was observed.

Fig.6 shows the inside of the vacuum vessel before operations resumed in 1991. The upper and lower X-point target plates are the main new features. Adjacent are mushroom shaped wall protection tiles. These are made of carbon fibre material and designed to take the forces resulting from "halo" currents. Some are instrumented. The plasma facing material of the belt limiters was also changed, with beryllium on the upper belt and graphite on the lower one.

Faulty Toroidal Field Coil

As reported in the 1990 Report, a faulty toroidal field coil was discovered in Octant No.3 during 1989. The faulty coil was changed during January 1990. Inspections revealed that the cause of the fault was a water coolant leak into the coil insulation and subsequent electrical leakage through the low resistivity path of the water. During 1990, the water coolant was replaced with an organic fluid (CFC) with much improved dielectric properties.

Subsequently, a further interturn fault was detected in Toroidal Field Coil No:4.2 in Octant No:4. This fault had no effect on operation and has been monitored since discovery to detect any

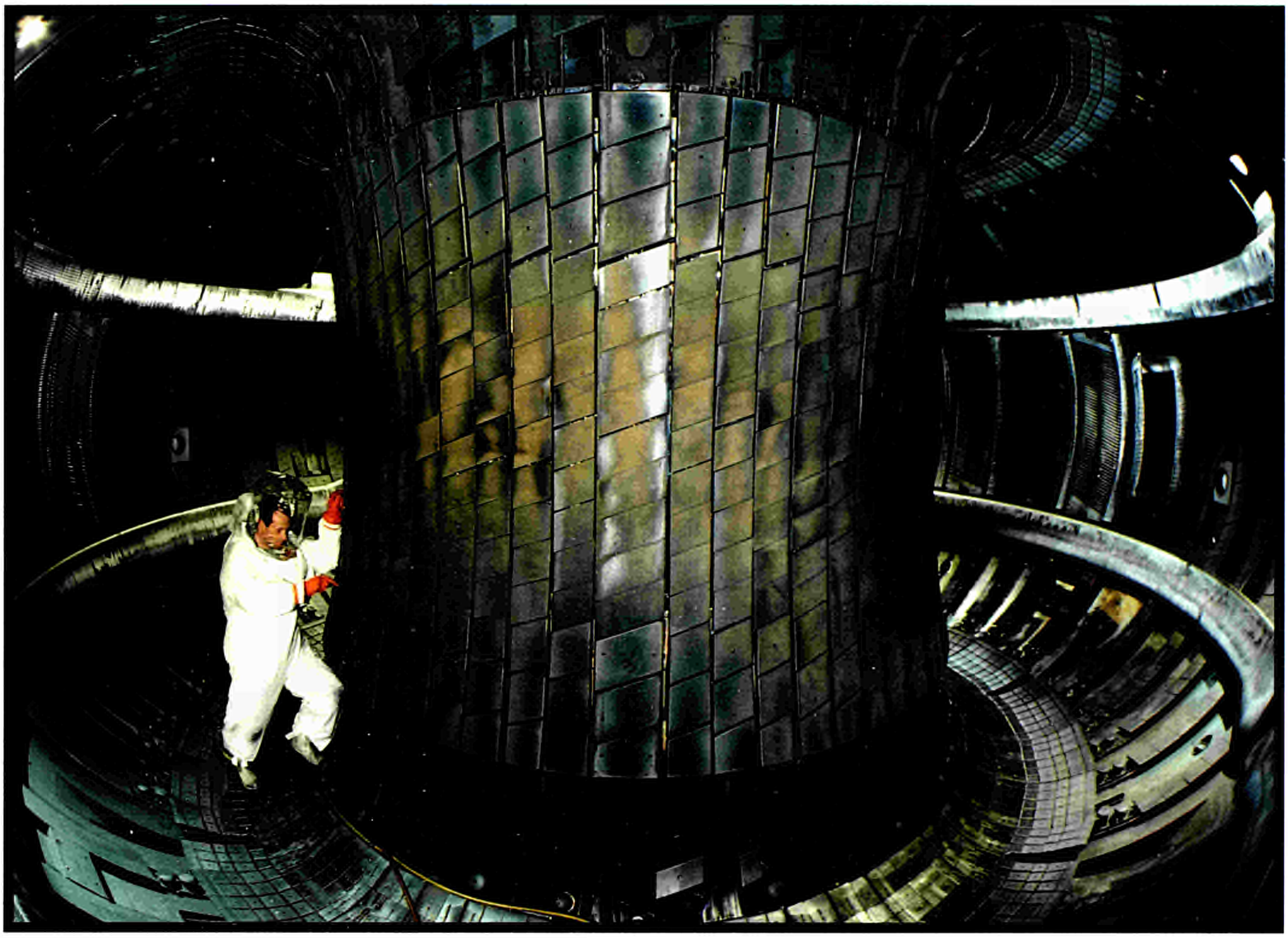


Fig.6: Inside the vacuum vessel before operations in 1991;

deterioration. This was carried out by measuring the fault ampere-turns during a standard toroidal field pulse and was made each day. Fig.7 shows the peak fault current throughout 1990 and 1991. The peak fault current increased after each field polarity reversal but then settled. This behaviour is not understood. The overall trend showed a slight increase towards the end of 1991, due to frequent polarity changes.

No faults were detected in the other toroidal field coils. These have been monitored similarly during operation and also by measurements at the coil terminals during shutdowns. This operation is intended to examine the effect of magnetic field ripple on the plasma confinement. Toroidal Field Coil No:4.2 will be changed during the 1992 shutdown. A spare coil is being prepared for insertion into Octant No.4.

Plasma Control

Despite attempts to improve vertical stabilisation, disruptive instabilities frequently lead to loss of vertical position and, consequently, to larger vertical plasma displacements (which can be up to 1m). The associated vertical force acting on the vessel

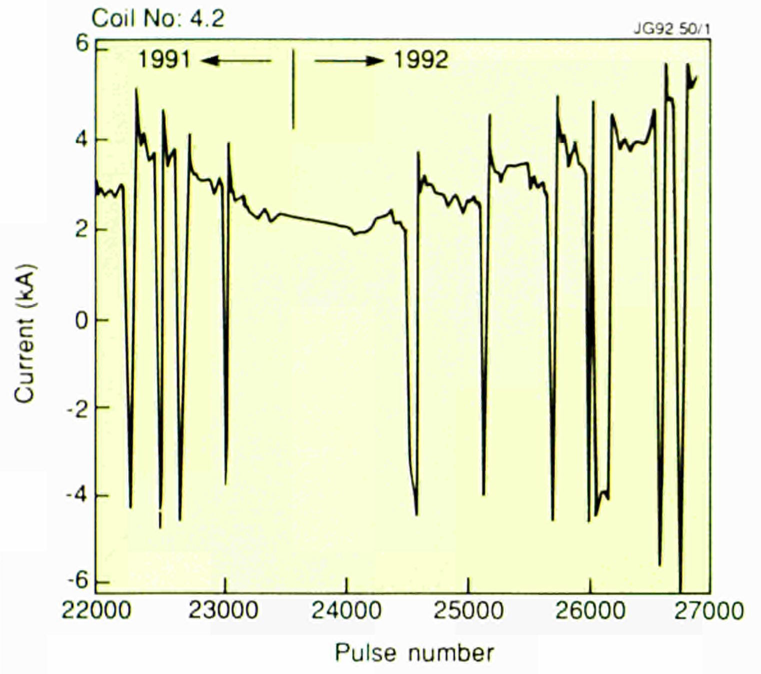


Fig.7: Peak fault current in the TF Coil No:4.2 during 1990 and 1991;

is large at high plasma current and when large shaping currents are applied, such as in single or double-null configurations. The vertical instability produces potentially dangerous forces at in-vessel elements such as protection tiles.

The systems for plasma current and shape control have been modified to reduce the current in the transformer coil and the current flowing through the shaping coils as quickly as possible when a disruption precursor, is detected by the plasma fault protection system. In most cases, the delay between the precursor signal and the disruption (typically, several 100ms) permits a substantial reduction of these currents. In turn, the destabilising force at the plasma is greatly reduced at the time of the disruption and the vertical position is much better maintained than without this current reduction. As a result, the vertical force on the vessel can be reduced by a factor of ten in most disruptions.

Neutral Beam Heating

The successful operation of both the Octant No:4 and Octant No:8 neutral beam injection systems was maintained throughout 1991, with high levels of both reliability and availability. Prior to the start of 1991 operations, the Octant No:8 system was converted to 140kV injection voltage to bring both injection system to 140kV operation in deuterium, with a total power of 15MW.

The major focus of the work carried out on the Neutral Beam (NB) heating system during 1991 was related to the Preliminary Tritium

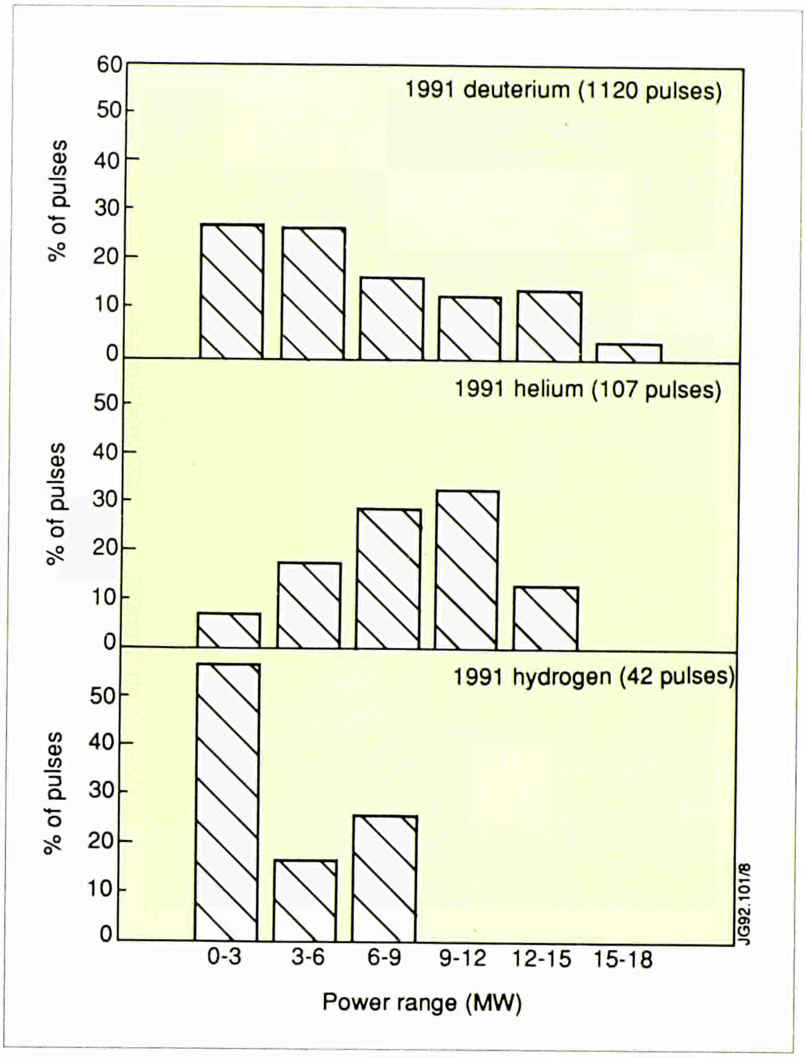


Fig.8: Distribution of injected power for the various beam species used throughout 1991;

Neutral Beam Heating

The two JET neutral beam systems have been designed for long (~10s) beam pulses. They have the unique feature that each injector consists of eight beam sources in a single integrated beamline system connected to the torus. The first beam sources have been designed to operate at accelerating voltages up to 80kV and for 1990 one system was substituted with units capable of operating up to 140kV. In addition, this box was also converted to operate with helium (He^3 and He^4) beams during 1990. In the D-T phase, one unit will be converted for operation with tritium at 160kV.

Each system is connected to the torus by a long narrow duct through which up to 10MW of power can be directed.

Experiment (PTE) for which neutral beam injection was used, not only to heat the plasma using fourteen beams of deuterium, but also to inject tritium fuel using two tritium beams. This was also the first time that multi-ampere energetic tritium beams had been produced at the megawatt power level and was a considerable technical achievement. Further, it demonstrated the importance and flexibility of the JET system, which had already been used to inject multi-megawatt beams of H, ^2D, ^3He and ^4He.

Due to the requirement to maximise the tritium fuelling of the plasma, two of the JET Positive Ion Neutral Injectors (PINIs) (consisting of a plasma generator and ion accelerator) were used in their four-grid 80kV configuration. At this beam energy, the maximum conversion efficiency of the primary tritium ion beam to energetic neutral atoms was ~72%. Details of this successful experiment are described in the section on the Preliminary Tritium Experiment.

In order to minimize the neutron production and thus the activation in the machine, the neutral beam system was converted to operate with ^3He. For the Octant No:4 injector, this involved re-establishing the operation achieved in 1990 and was extended to higher extraction voltage (125kV was the limit for the 1990 operations). For the Octant No:8 injector, ^3He operation was commissioned for the first time and successfully operated up to 145kV extraction voltage. Routine operation was established on both injectors and extensive characterisation of the beamline operation was carried out. At the end of this phase, injected powers of up to 13.5MW at extraction voltages of 135-145kV were achieved. This was a considerable improvement over the 125kV limit reached in 1990. In addition, several of the best conditioned PINIs were successfully operated up to the 155kV limit.

An overview of the power distribution for injected pulses in D^0, $^3He^0$ and H^0 operation is shown in Fig.8. Due to the high reliability maintained throughout 1991, the distribution of power reflects the demands of the experimental programme and not a shortfall in performance.

Radio Frequency Heating

The ion cyclotron resonance frequency (ICRF) heating system is used for highly localized heating of the JET plasma. The system is also used for current drive studies which are in the initial stage. The wide frequency band (23 to 57MHz) allows variation in the heating position as well as the minority ion species which is resonant with the wave (H or ^3He at present, D in the future D-T phase). The heating system is composed of eight units, each driving an antenna installed between the belt limiters in the toroidal vessel. Each unit is made of two identical sub-units, sharing a common high voltage power supply and a common low power RF drive. The original design power was rated at 15MW for 20s. However, the output state of each sub-unit has been upgraded to 2MW each instead of the original 1.5MW to provide an amended design power of 24MW in the plasma. This modification was completed early in 1990, and a power of 22MW has been successfully coupled to the plasma. A diagram of the systems main components are shown in Fig 9.

During long pulse operation, up to 30MJ per RF antenna have been launched into the plasma without indication of impurity release specific to ICRF edge effects. This performance was achieved in the plasma limiter configuration. The evolution of the JET experiment has entailed the use of different plasma configurations such as double and single-null X-points with the possibility of large distance between the last closed magnetic surface and the antennae. In addition, the antenna impedance does not present a constant load, resulting in stringent conditions for delivering the heating power.

Radio Frequency Heating

Ion Cyclotron Resonance Frequency (ICRF) heating has been chosen for JET and the wide operating frequency band (23-57MHz) allows the system to be operated with the various mixes of ion species required in the different phases of the scientific programme and to choose the location where the heating in the plasma occurs.

The ICRF heating system has been designed in eight identical modular units. Each unit is composed of a tandem amplifier chain, a network of coaxial transmission lines and matching elements and finally on antenna located in the vacuum vessel on the outer wall. Ultimately, the eight RF generators will produce a maximum output power of 32MW.

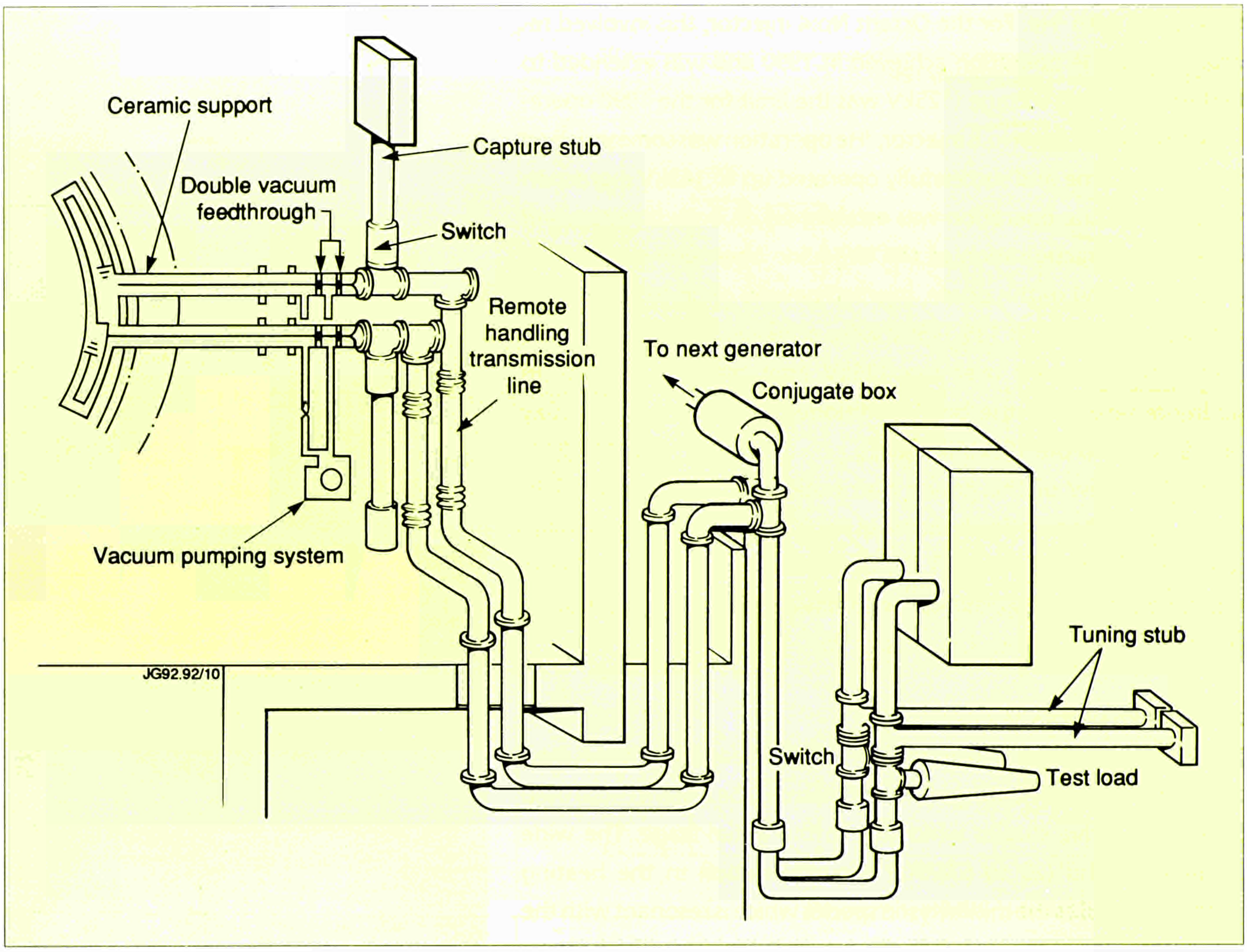


Fig.9: Diagram of the !CRF System;

Control and protection circuitry has been developed to handle naturally occurring transient antenna breakdowns, incoherent energy cross-coupling between antennas and rapidly varying plasma loads such as during events like L to H-mode transitions. Fig.10 shows a schematic diagram of the control and matching system for each ICRF amplifier, of which there are sixteen in the ICRF plant. If any of the transmission line or source parameters are exceeded, the power output is limited automatically to that value. Arcs in the antenna or generator cause a RF power trip for ~20ms before reapplication. If the number of trips exceeds a preset maximum value or if an arc in the tetrode occurs, the RF pulse is terminated prematurely.

This improved control system has permitted the ICRF plant to operate with some of the plasma configurations (in particular, tilted single-null plasmas), frequently used with high power Neutral Beam Injection, which were not suitable until recently for ICRF heating. The reason was the large separation between plasma and ICRF antennae, which, together with the poloidal asymmetry of the plasma-antenna distance, implied low loading resistance. During 1991, a significant amount of ICRF power was coupled in these

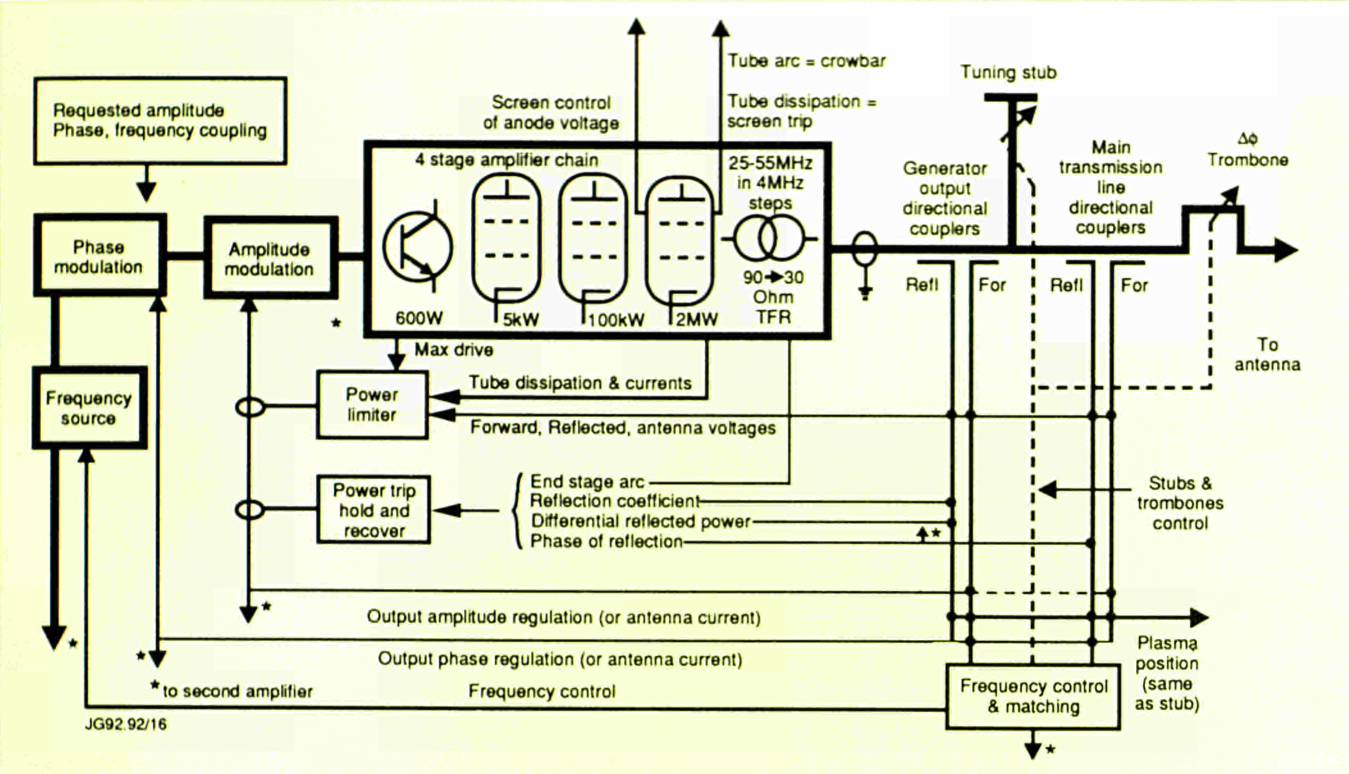


Fig.10: Schematic of the control and matching system for each ICRF amplifier;

configurations. In particular, 8MW of ICRF power was coupled to the single-null X-point configuration, with the last closed flux surface at least 15 cm from the antennae. At this power input, H-modes were readily obtained.

Lower Hybrid Current Drive

The Lower Hybrid Current Drive (LHCD) system (12MW at 3.7GHz) is intended to drive a significant fraction of the current flowing in the plasma, by direct acceleration of the plasma electrons through interaction with lower hybrid waves. This should stabilize sawtooth oscillations, thereby increasing the central electron temperature and improving overall JET performance. This will be the main tool in JET for controlling the plasma current profile. A prototype system consisting of two launching units (one built by CEA Cadarche, France, and the other built by JET) fed by a total klystron power of 4MW was installed during 1990. The prototype system has coupled to the plasma up to 2.2MW for 8s and 2.5MW for 1s in a large variety of plasma conditions. A view of the LHCD system in the Torus Hall is shown in Fig. 11.

Both radio-frequency (RF) systems have been instrumental in achieving long pulse plasma operation of up to one minute flat-top duration, by providing a high temperature, low resistivity plasma to limit resistive flux consumption from the Ohmic transformer and in part by direct current drive. A one minute long pulse was achieved at a plasma current of 2MA, with 50s LH pulse at 1MW (only six

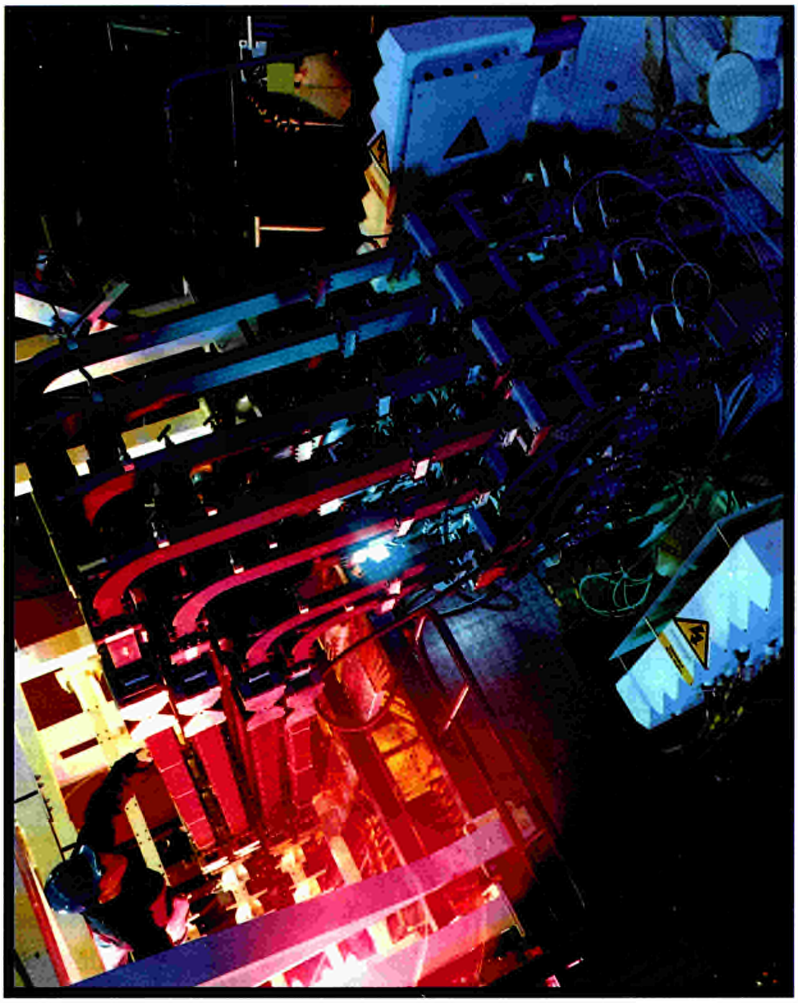


Fig.11: View of the LHCD end plate, of the vacuum windows together with the ancillary and protective systems and of the splitting waveguide network in the Torus Hall.

klystrons out of eight were operated due to power supply limitations) and three successive 20s 3MW pulses of ICRF power. The combined application of two RF systems provided 230MJ of energy out of a total of 280MJ in the plasma , which included the ohmic contribution, and reduced the flux consumption by about 40%. Both the ICRF coupling resistance and the LH reflected power were maintained approximately constant throughout the pulse, indicating steady coupling conditions with no impurity influxes associated with antenna and launcher operation.

Pellet Injection

An important requirement for the successful operation of future fusion plasmas is a fuelling and density control system. One method of raising the density, and replenishing it during operation, is to inject pellets of solid hydrogen into the plasma at high speed so that these penetrate the outer plasma layers and reach the centre before completely evaporating. The resulting clean plasmas are comparatively resistant to disruption.

The multi-pellet injector on JET was built and operated under a collaborative bilateral agreement between JET and the US Department of Energy (USDoE). The agreement relates to work on the three-barrel, repetitive, pneumatic, single-stage gun launcher with pellets of 2.7, 4 and 6 mm sizes and speeds around 1.3 kms^{-1}, constructed by Oak Ridge National Laboratory (ORNL). During 1991, the attendance at JET of US personnel was somewhat lower than the four man-years per year foreseen under the Agreement. Negotiations for a prolongation of the Pellet Agreement have so far not been successful. Experimental work under the Agreement will therefore cease at the end of the 1991/92 campaign, though joint evaluation of JET data will continue to the end of 1992. The ORNL Launcher will be decommissioned in 1992 and returned to the US, where its implementation on another tokamak experiment is being considered. During the Preliminary Tritium Experiment (PTE), the pellet injector was valved off. However, immediately following the PTE, the injector resumed operation. A careful assessment of the possible amount of exposure of the ORNL launcher to tritium was jointly carried out and showed that, with certain conditions maintained, the level will not impede the re-installation of the launcher on another experiment.

The pellet injector was extensively used during 1991 in a number of experiments for various purposes of density profile shaping and fuelling. Further details on new developments relating to pellet injectors for fuelling on JET will be reported in a later section of this report.

Diagnostics

The location of the JET measuring systems (or diagnostics) is shown in Fig.12 and their status at the end of 1991 is shown in Table.4. Operational experience has been good and most of these systems operate automatically with minimal supervision from scientific staff. The measurements obtained are accurate and reliable and provide important information on the behaviour of the plasma.

Temperature and Density Measurements

The three instruments in the electron cyclotron emissions (ECE) measurement system, the Michelson interferometer, the grating polychromator and the heterodyne radiometer, continue to provide electron temperature data on almost all JET pulses. Although the Michelson interferometer has only moderate spatial and temporal resolution (~15cm and ~15ms, respectively), it provides several hundred absolutely calibrated temperature profiles on all plasma pulses with toroidal field above 1.5T. These temperature profiles are used routinely to cross-calibrate the temperatures measured by the fast, multi-channel instruments (a 12-channel

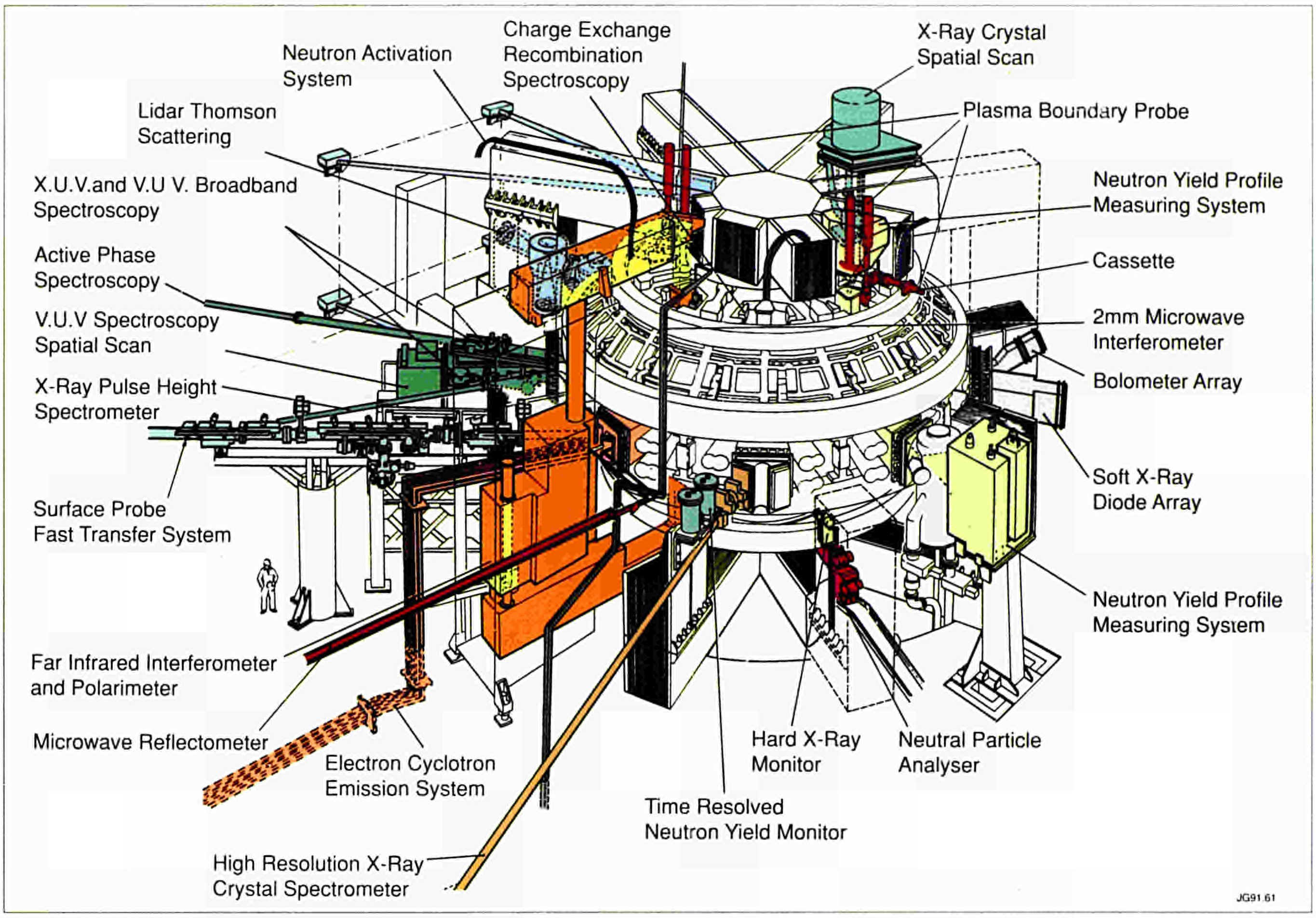


Fig.12: General layout of diagnostics in the JET machine

grating polychromator and a 44-channel heterodyne radiometer), which are difficult to calibrate with black-body sources. This permits a detailed study of the time evolution of the absolute electron temperature at many fixed radii in the plasma, with both good spatial resolution (~6cm for the polychromator and ~3cm for the radiometer) and temporal resolution down to ~10ms.

The absolute response of the complete Michelson/waveguide/antenna system was re-measured during the shutdown in early 1991, using the beryllium compatible high-temperature black-body source mounted in front of the antenna, in the vacuum vessel. Subsequent comparisons with the plasma electron temperatures measured by LIDAR Thomson scattering continue to show good agreement within the estimated absolute uncertainty of ±10%.

The multichannel reflectometer probes the JET plasma along a major radius in the mid-plane with electromagnetic radiation propagating in the ordinary mode through the plasma. The instrument has twelve channels covering the frequency range 18.6 - 80GHz and so potentially can probe electron densities in the range 0.4 - 8 x $10^{19}m^{-3}$. In practice, the attenuation in the transmission system at the highest frequency is too high to permit routine

Table.4: Status of JET Diagnostics at the end of 1991.

System	Diagnostic	Purpose	Association	Automation
KB1	Bolometer array	Time and space resolved total radiated power	IPP Garching	A
KB2X	X-point bolometer	Time and space resolved power from X-point region	JET and IPP Garching	A
KC1	Magnetic diagnostics	Plasma current, loop volts, plasma position, shape of flux surfaces, diamagnetic loop, fast MHD	JET	A
KE3	Lidar Thomson scattering	T_e and n_e profiles	JET and Stuttgart University	A
KE5	q-profile Thomson scattering	Measurement of q-profile	JET	SA
KF1	High energy neutral particle analyser	Ion energy distribution up to 3.5MeV	Purchased from Ioffe St Petersburg	SA
KG1	Multichannel far infrared interferometer	$\int n_e ds$ on six vertical chords and two horizontal chords	CEA Fontenay-aux-Roses	SA
KG3	Microwave reflectometer	n_e profiles and fluctuations	JET and FOM Rijnhuizen	A
KG4	Polarimeter	$\int n_e B_p ds$ on six vertical chords	JET and CEA Fontenay-aux-Roses	SA
KH1	Hard X-ray monitors	Runaway electrons and disruptions	JET	A
KH2	X-ray pulse height spectrometer	Monitor of T_e, impurities, LH fast electrons	JET	SA
KJ1*	Soft X-ray diode arrays	MHD instabilities and location of rational surfaces	IPP Garching	SA
KJ2*	Toroidal soft X-rays	Toroidal mode numbers	JET	SA
KK1	Electron cyclotron emission spatial scan	$T_e(r,t)$ with scan time of a few milliseconds	NPL, UKAEA Culham and JET	A
KK2	Electron cyclotron emission fast system	$T_e(r,t)$ on microsecond time scale	FOM Rijnhuizen	A
KK3	Electron cyclotron emission heterodyne	$T_e(r,t)$ with high spatial resolution	JET	SA
KL1*	Limiter viewing	Monitor hot spots on limiter, walls, RF antennae, divertor target tiles	JET	A
KL3	Surface temperature	Surface temperature of target tiles	JET	M
KM1	2.4MeV neutron spectrometer	Neutron spectra in D-D discharges, ion temperatures and energy distributions	UKAEA Harwell	SA
KM3	2.4MeV time-of-flight neutron spectrometer		NEBESD Studsvik	A
KM7	Time-resolved neutron yield monitor	Triton burning studies	JET and UKAEA Harwell	A
KN1	Time-resolved neutron yield monitor	Time resolved neutron flux	UKAEA Harwell	A
KN2	Neutron activation	Absolute fluxes of neutrons	UKAEA Harwell	SA
KN3*	Neutron yield profile measuring system	Space and time resolved profile of neutron flux	UKAEA Harwell	A
KN4	Delayed neutron activation	Absolute fluxes of neutrons	Mol	A
KR2	Active phase NPA	Ion distribution function, $T_i(r)$	ENEA Frascati	A
KS1	Active phase spectroscopy	Impurity behaviour in active conditions	IPP Garching	SA
KS2*	Spatial scan X-ray crystal spectroscopy	Space and time resolved impurity profiles	IPP Garching	SA
KS3	H-alpha and visible light monitors	Ionisation rate, Z_{eff}, impurity fluxes from wall and limiter	JET	SA
KS4	Charge exchange recombination spectroscopy (using heating beam)	Fully ionized light impurity concentration, $T_i(r)$, rotation velocities	JET	SA
KS5	Active Balmer α spectroscopy	T_D, n_D and $Z_{eff}(r)$	JET	SA
KS6*	Bragg rotor X-ray spectrometer	Monitor of low and medium Z impurity radiation	UKAEA Culham	SA
KS7*	Poloidal rotation	Multichannel spectroscopic measurement of poloidal rotation	UKAEA Culham	M
KT1*	VUV spectroscopy spatial scan	Time and space resolved impurity densities	CEA Fontenay-aux-Roses	A
KT2*	VUV broardband spectroscopy	Impurity survey	UKAEA Culham	A
KT3	Active phase CX spectroscopy	Fully ionized light impurity concentration, $T_i(r)$, rotation velocities	JET	SA
KT4*	Grazing incidence + visible spectroscopy	Impurity survey	UKAEA Culham	A
KX1	High resolution X-ray crystal spectroscopy	Central ion temperature, rotation and Ni concentration	ENEA Frascati	A
KY2	Surface probe fast transfer system	Plasma wall and limiter interactions including release of hydrogen isotope recycling	UKAEA Culham	Automated, but not usually operated unattended
KY3*	Plasma boundary probes	Vertical probe drives for reciprocating Langmuir and surface collector probes	JET, UKAEA Culham and IPP Garching	
KY4	Fixed Langmuir probes (X-point belt limiter)	Edge parameters	JET	SA
KZ3*	Laser injected trace elements	Particle transport, T_i, impurity behaviour	JET	M
Kγ1	Gamma-rays	Fast ion distribution	JET	M

* Not compatible with tritium A=Automatic, SA=Semi-automatic, M=Manual

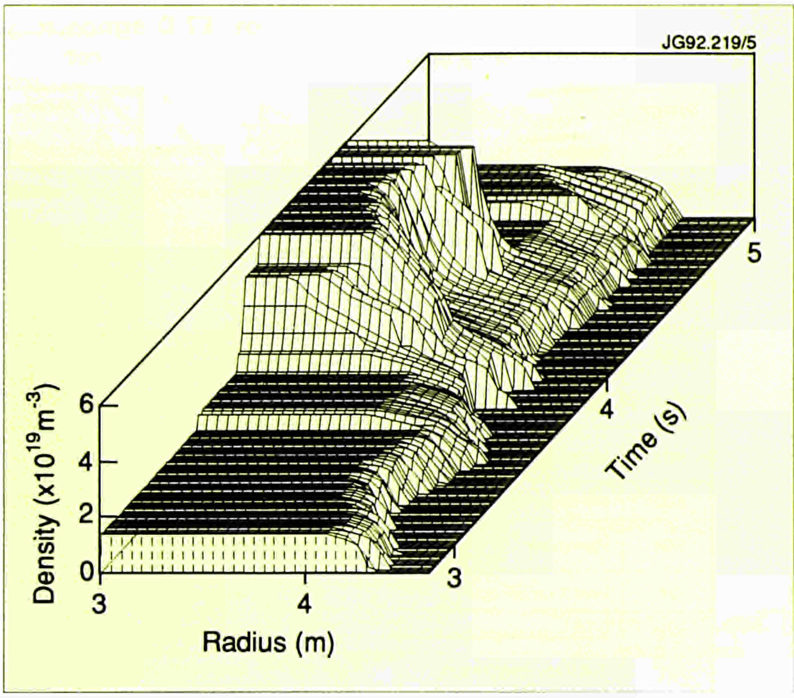


Fig.13: Edge density profiles measured with multichannel reflectometer on a pulse in which a pellet was injected at t = 3.8s

operation and so the highest density probed is ~7 x 10^{19} m^{-3}. The instrument can be operated with the source frequencies periodically swept, to measure the electron density profile, or with the probing frequencies fixed to measure relative fast movements of the different density layers.

Improvements to the system have meant that as many as 400 density profiles of the outer region of the plasma are now frequently obtained on individual JET pulses. Good agreement is generally found with density profiles measured with the LIDAR diagnostic. The reflectometer profiles complement those obtained with the transmission interferometer and with probes. They are used in a wide range of studies, particularly studies of edge related phenomena. Dramatic changes in the profiles are observed (Fig.13) as a result of pellet injection, although the time response of the system is not adequate to make measurements during the injection process itself.

The main LIDAR scattering system has operated throughout the year with high reliability, producing full profiles of electron temperature, density and pressure for most pulses. An improvement in the long term reliability of the density calibration has been achieved by carrying out in-site window cleaning of the inside of the vacuum windows. A gradual build up of a coating slowly reduces transmission during long periods (months) of plasma operation. With the present mode of operation, this produces a gradual change in density as measured by the LIDAR system. This can be clearly seen on the comparison of the LIDAR

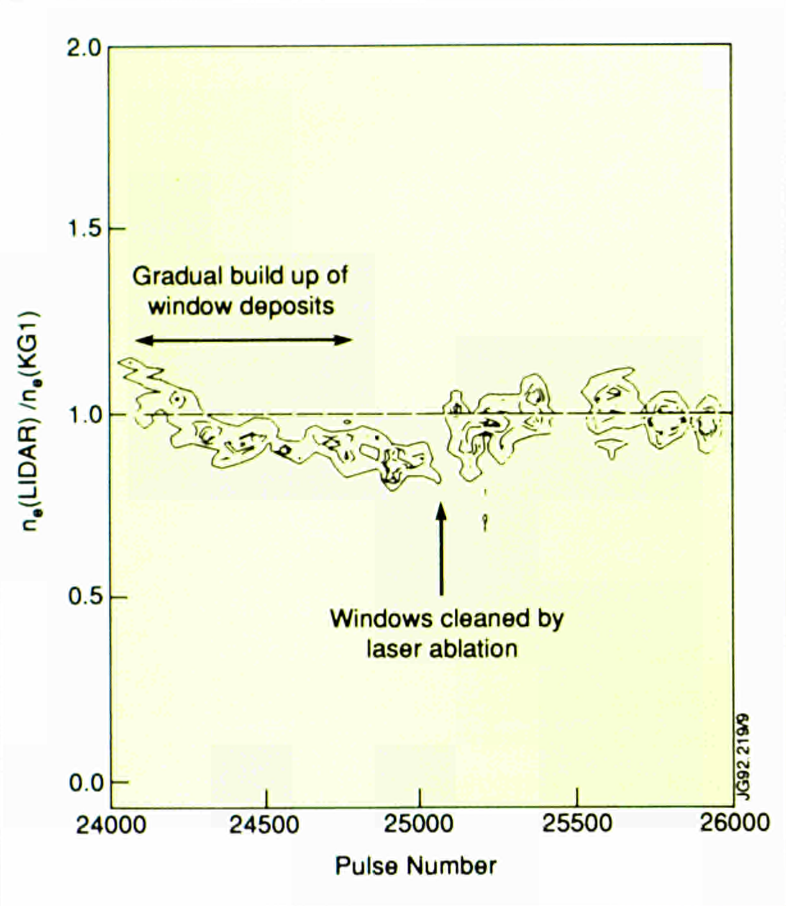


Fig.14: Comparison of line integral density data measured with the multi-channel interferometer and the LIDAR system showing the effect of the build-up of deposits from plasma operation and their removal by laser ablation. The grid cell size is 0.02 x 50 discharges and the contour lines are drawn at 7,10,13,16 and 19 observations per cell

and interferometer density measurements (see Fig.14). The LIDAR window monitoring system shows that the coating is chromatically neutral, a fact confirmed by a T_e comparison with ECE measurements, which shows no effect. The windows can now be periodically cleaned by laser ablation of the internal surface coating. The LIDAR laser is remotely steered to a point on one of the collection windows and a series of nine pulses, each of a few joules energy is fired, cleaning a 10cm^2 area. A computer program automatically points the laser to all areas of the collection windows. In this way, the whole window area can be cleaned in one or two commissioning shifts. The improvement is clearly seen in the density comparison data.

Despite recent set-backs, the goal of increasing the repetition rate of the LIDAR measurements has continued and a contract has been placed for the development of a 4Hz ruby laser. If this is fully successful it may also be possible to upgrade the existing laser to the same performance. This would allow LIDAR measurements at 8Hz when the laser beams are combined along the same path.

Boundary Measurements

The number of single element Langmuir probes has been increased from 23 to 46 during the 1991 campaign. Two poloidal arrays were built in the carbon upper (12 probes) and in the Beryllium lower (16 probes) divertor targets as well as in the belt limiters (10 probes), and in the ICRF (4 probes) and LH (4 probes)antennae protection tiles. Single element probes in the upper and lower targets were originally designed to offer a well defined collector area. This design has worked well but probes in high heat load areas were progressively destroyed. Useful information has been obtained, both on hardware and on software, from operations of the divertor probes in plasma conditions similar to those envisaged for the pumped divertor. This information is being used on the design for the probes in the pumped divertor targets.

Both the Fast Transfer System and Plasma Boundary Probe system on Octant No: 1 have been used to expose collector probes. Among the subjects studied are the efficiency of the beryllium evaporation and the nature of the escaping fast He ions from the core. Particularly important was the exposure of a stationary probe during the preliminary tritium experiment, from which the tritium coverage of the first wall could be deduced. The Surface Analysis Station has now been mothballed due to shortage of staff effort. However, analysis continues on a contract basis. The Plasma Boundary Probe System will be retained for the pumped divertor phase.

CCD cameras equipped with filter carousels (CI, CII, CIII, OII, BeI, BeII, and H_α/D_α) have been extensively used to observe general discharge behaviour and more importantly to assess the loading of the Carbon and Beryllium targets. Camera observations of the new X-point targets shows that both upper and lower targets had been well aligned and poloidal contours of impurity influxes and hydrogen recycling were obtained from the video images. These poloidal contours are thought to be representative of the overall interaction with the target.

Density profile measurements in the scrape-off plasma can be made using an energetic neutral lithium beam. As the beam traverses the boundary layer, it is attenuated, mainly due to ionization by electron impact, and the neutral lithium atoms are excited to higher energy states. The beam attenuation profile can be deduced from measurements of the spatial profile of intensity of spectral emission, from which the electron density profile over the beam penetration distance may be obtained. Application of this technique in JET during the pumped divertor phase should be feasible. An appropriate system has been defined, and detailed design and procurement are in progress. To achieve good spatial resolution in JET, it will be necessary to probe the plasma near the top of the torus at the

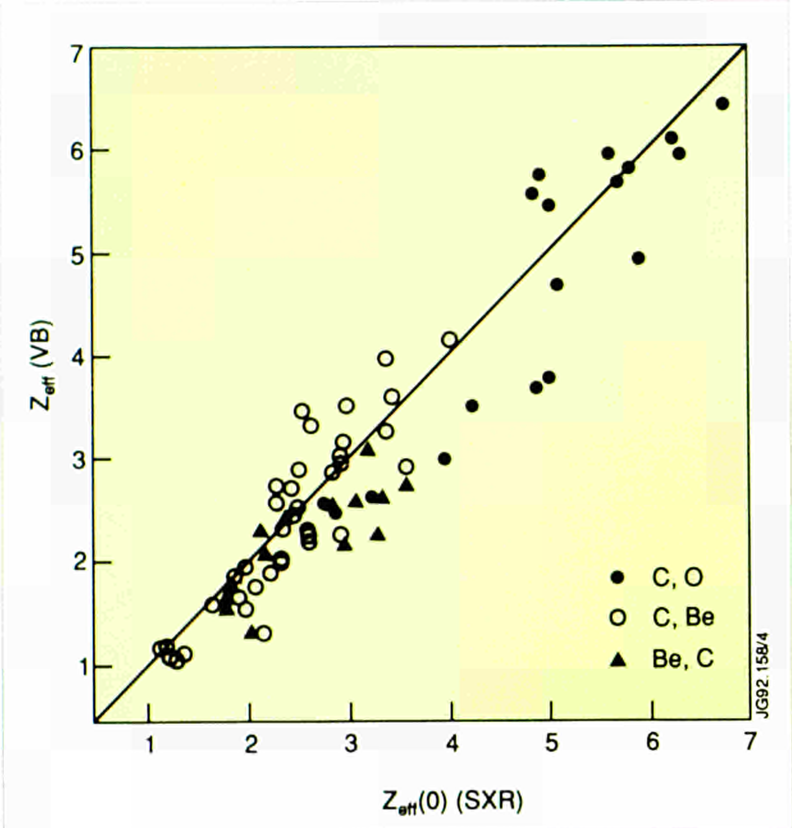


Fig.15: Comparison of Z_{eff} derived from analysis of the soft X-ray emissivity measurements and visible bremsstrahlung measurements

stagnation point in the scrape-off layer, where the density scrape-off length should be 3-5cm. It should be feasible to make measurements with ± 5mm spatial resolution over a path of 20cm, and time resolution of ~10ms.

Impurity Analysis

JET plasmas offer a variety of mixtures of light impurities in the hot plasma core for analysis by various spectroscopic means. Radial profiles of the effective ion charge, Z_{eff} are determined from Abel inverted profiles of the multi-chord visible Bremsstrahlung signals and simultaneous measurements of main light impurities by charge exchange spectroscopy. The dilution factor (the ratio of deuterium density to electron density, n_D/n_e) is derived from the density of electrons and light impurities.

Absolutely calibrated signals from the soft X-ray cameras can be used to determine the radial and time resolved values of Z_{eff}. In JET, the main impurities are the light elements Be, C and O which are fully ionised, with occasional small contributions from Ni and Cl, which generally only contribute a few per cent of the radiated power. To determine Z_{eff}, calculations are made of the radiated power observed through the Be filters which cover the detectors. An ionization equilibrium calculation is made both with and without transport effects. In addition, the relative contributions of the different low-Z species in a particular plasma are required and these were determined from VUV spectroscopy and active charge exchange spectroscopy. The results of these calculations are compared with the tomographically inverted soft X-ray data and Z_{eff} is then found. Good agreement is found with existing methods, and a comparison is shown in Fig.15 with the visible bremsstrahlung data for shots with flat Z_{eff} profiles. This method allows Z_{eff} measurement on a rapid time-scale.

A new instrument, a Bragg rotor X-ray spectrometer, was installed and operated during 1991. A wide range of crystal and multilayer diffractors are mounted on a hexagonal rotor to give almost complete coverage of the soft X-ray spectrum between 0.1nm and 10nm. This allows a single instrument to monitor K- L- or M-shell line radiation from highly ionised species of all plasma impurities. This instrument is particularly useful for measurement of light impurities such as beryllium and carbon which usually contribute most to the radiated power and effective charge. The instrument shares the shielded, tritium-compatible beam-line of the Active Phase Double-Crystal Monochromator, and both instruments were operated successfully to monitor all main impurities throughout the PTE.

Neutron Measurements

The time-of-flight 2.5MeV neutron spectrometer was rebuilt during the 1991 shutdown. The new spectrometer incorporates much of the equipment delivered as part of the 14MeV neutron spectrometer intended for use during the D-T phase of JET operation. It retains the energy resolution of the former spectrometer (120keV) but has about 6 time greater detection efficiency. This increase is critical as it permits high quality neutron energy spectra to be acquired on a time-scale ~ 200ms (for the highest intensity D-D discharges), compared with 1s previously. Since JET rarely sustains high neutron emission strengths for as long as one second, the improvement is obvious. The time evolution of the neutron energy spectra can now be studied, as illustrated in Fig.16. The first example shows the spectrum due to beam injection into a cold plasma (beam-plasma reaction dominated); the subsequent spectra are obtained as the plasma is heated to its peak temperature so that the thermal reaction contribution becomes important. Finally, an impurity influx takes place and the beam power is reduced, so that the separate thermal and beam-plasma contributions are seen. By taking advantage of such sequences of spectra, the maximum plasma temperature attained can be estimated, whereas analysis of the single high-temperature spectrum on its own can only provide an upper limit on the peak temperature.

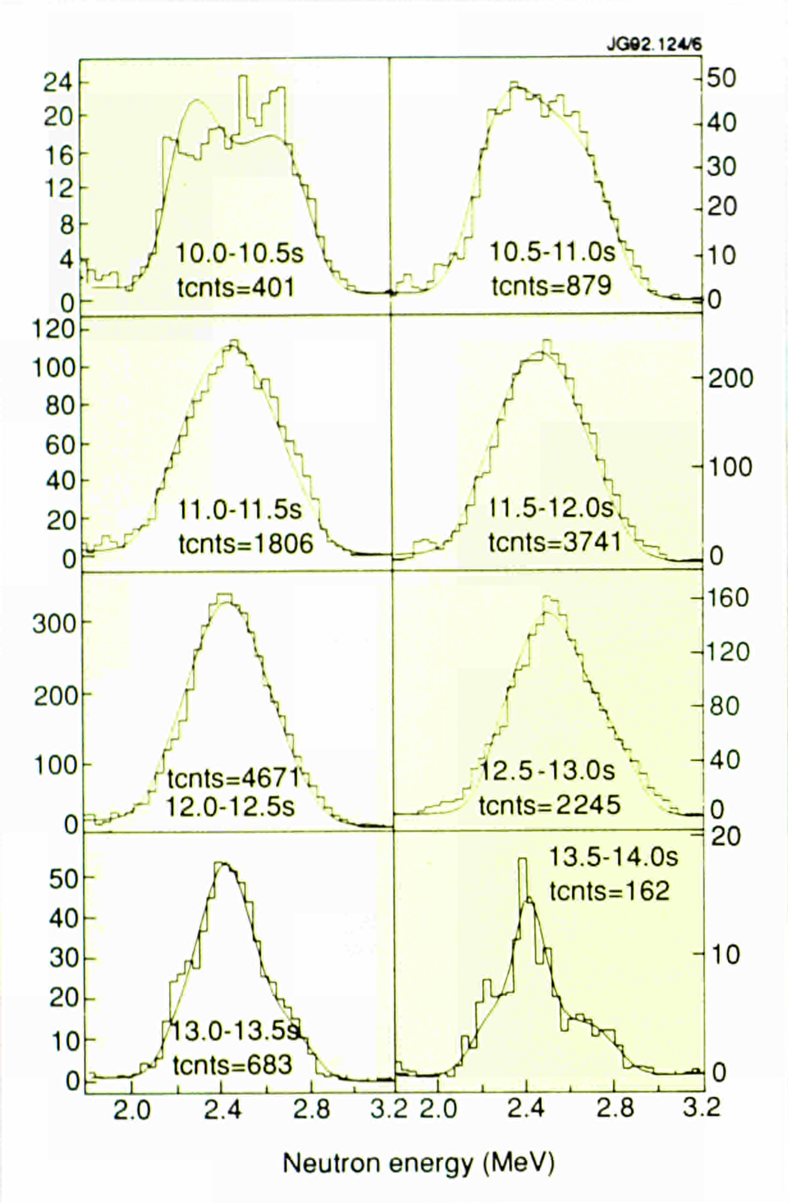


Fig.16: A series of neutron energy spectra obtained with the time-of-flight spectrometer showing how the spectrum shape for a deuterium beam-heated discharge varies with time after heating is applied. Analysis of such discharges allows determination of the time dependence of axial plasma temperature despite strong interference from beam-plasma neutron emission

The strength of the 2.5MeV neutron emission from JET plasmas formed with deuterium fuel is measured with pairs of fission chambers positioned near three of the main horizontal ports of the machine. Each pair comprises a ^{235}U chamber for low neutron yields and a ^{238}U chamber for high yields. The two discharges of the main preliminary tritium experiment, were initially formed in deuterium but were heated with 14 PINI's injecting deuterium and two PINI's injecting tritium. In these discharges, the 14 MeV neutron emission was about 30 times the strength of the 2.5 MeV neutron emission, with a peak intensity of $6 \times 10^{17} s^{-1}$. These discharges provided the first opportunity to test the fission chambers with 14 MeV neutrons. In particular, the ^{235}U fission chambers were operated close to the limit of their associated electronic equipment; this was not the case for the ^{238}U fission chambers as these are 6000 times less sensitive. The fission chambers were calibrated retrospectively against activation measurements.

Since the fission chambers cannot discriminate between 2.5 and 14MeV neutrons, the 14MeV neutron emission was recorded separately using several silicon diodes positioned strategically to cover the anticipated intensity range. These diodes provide an effective energy threshold of about 8 MeV; these were calibrated against activation measurements prior to the main discharges.

The Neutron Profile Monitor provides a measure of the spatial variation of neutron emissivity throughout the plasma volume. During the 1990/91 shutdown, the pulse-processing and data-acquisition system of the monitor was effectively doubled so that 2.5 and 14MeV neutron measurements could be made independently. The motivation for this upgrade was originally to permit the burn-up of the 1 MeV tritons emitted during high yield deuterium discharges to be studied, but proved most opportune for the study of the tritium discharges. The instrument performed well for the 1% tritium injection, for which separate 2.5 and 14MeV neutron emission profiles were obtained and the 2.5 and 14MeV counting rates were of similar magnitude. For the 100% tritium injection, the collimation channels were reduced in aperture with suitable collimator inserts to prevent the electronics from being overloaded. The results were satisfactory, even though the profile monitor operated close to the limit of its capability. The present type of detector (liquid scintillator) is unsuitable for power levels in excess of 2MW.

Fast Particle and Alpha-Particle Studies

A new high energy neutral particle analyzer (NPA) instrument, purchased from the AF Ioffe Physical-Technical Institute, St. Petersburg, Russia, was installed on JET during 1991. Consisting of eight energy channels with common mass selection, it measures fluxes of hydrogen, deuterium, tritium, helium-3 and helium-4 atoms and their energy spectra in the range 0.5 - 3.5MeV. MeV ions in the plasma arise from hydrogen and helium minority ICRF heating, or from fusion reactions. These ions then neutralize by charge exchange with low energy atoms in the plasma, or by recombination, forming MeV atoms which exit the plasma. The low energy plasma atoms are from injected neutral beams, or from recycling.

The performance of the instrument has exceeded initial expectations in respect of discrimination against neutron noise and magnitude of high energy atomic flux measured. The instrument has been applied in elucidating ICRF minority heating physics; in questions of

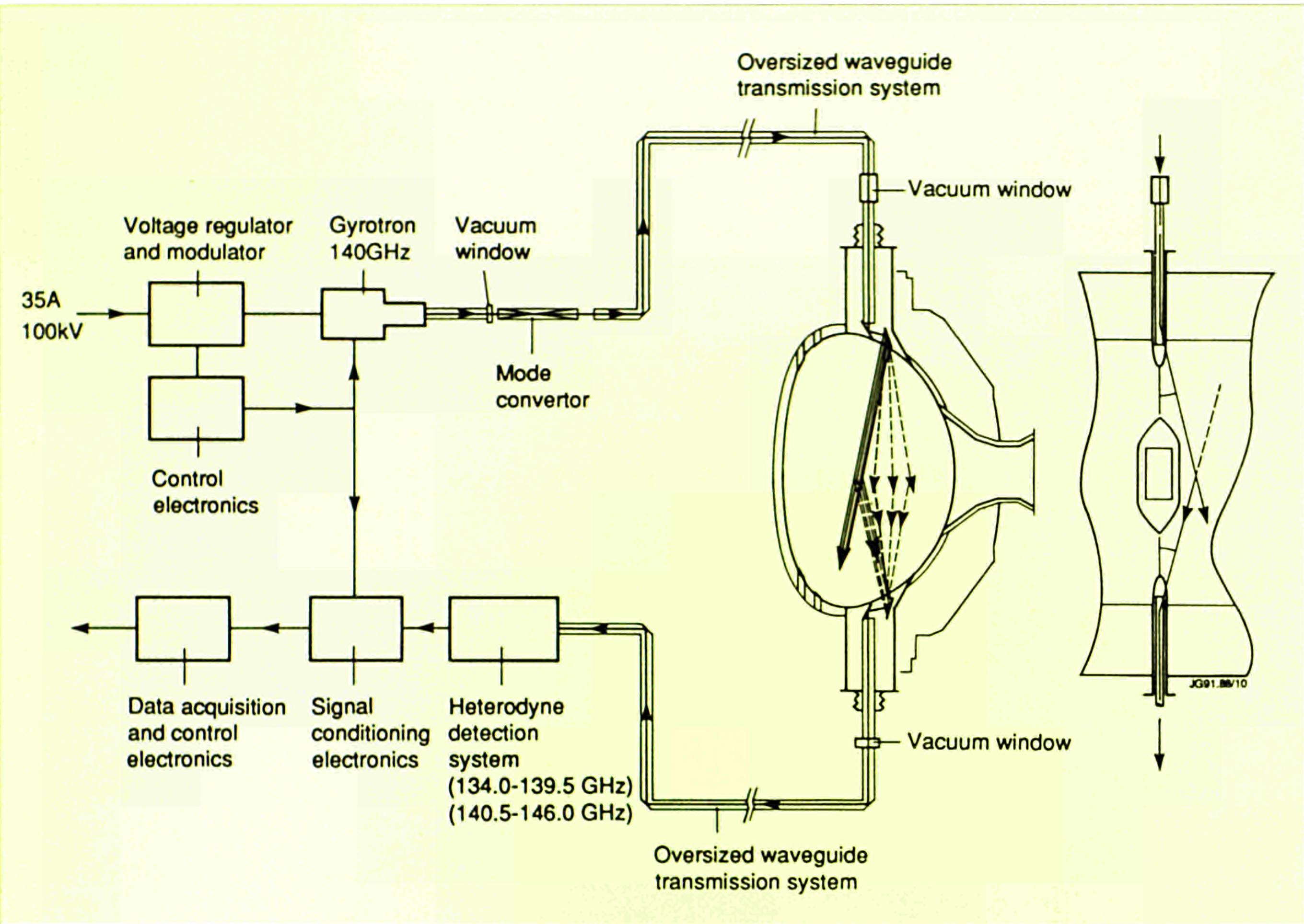


Fig.17: Schematic of collective scattering system for measuring spatial and velocity distributions of fast ions including alpha-particles

high energy ion loss due to MHD activity and to static field fluctuations; and in aspects of ion neutralization processes and atomic cross-sections. In He^3 minority ICRF heating experiments, detection of MeV energy He^3 by double charge-exchange on injected He^4 beams has been demonstrated. Time resolved measurements were undertaken of fusion α-particle energy spectra in the MeV range.

The time-of-flight NPA instrument, located at the bottom of Octant No:1, with a vertical line-of-sight crossing the torus mid-plane at R = 3.1m, has been deployed for measurements of energy spectra of flux of H, D, T, ^{3}He, and ^{4}He atoms emitted from the plasma, in the energy range 0.5 - 200keV, from which energy spectra of ions in the plasma can be deduced. This facility has been used routinely to make inferences about plasma behaviour, relative concentrations of different ions in the plasma, and physics of NBI and ICRF heating.

A diagnostic system to measure the velocity and spatial distributions of fast ions, including α-particles in the D-T phase, is in preparation. The system is based on collective scattering of radiation with a frequency of 140GHz generated by a powerful gyrotron source. The principal system is shown in Fig.17. During 1991, all the technical systems for the initial phase of operation of this system were installed and commissioning started. The in-vessel components, including steerable launch and receive mirrors, were installed during the shutdown. The installation of the launch and receive waveguides was completed with special attention being paid to the alignment. Care was taken to ensure that the alignment was precise and insensitive to thermal expansion. Quasi-optical universal polarizers for launch and receive channels were constructed and tested, and found to perform adequately. The protection and preliminary control systems, and detection and data acqusition systems were constructed.

During the year, a collaborative agreement with the USDoE was established. Under the terms of the agreement, the DoE is loaning JET a 140 GHz, 60 kW, CW gyrotron and high voltage power supplies. While lower in power than the gyrotron that will eventually be used on the system (400kW), this gyrotron should be sufficient for planned preliminary measurements. In addition, specialist USDoE staff have assisted with the commissioning of the system and preparation for plasma measurements.

Remote Handling

The main objective of the remote handling programme is to prepare for the introduction of tritium into JET, which will generate a large number of D-T fusion reactions, with a high flux of 14MeV neutrons. Some of these neutrons will be captured by the structure of the machine, making it too radioactive to approach. Therefore, all maintenance will need to be carried out by remote control from outside the Torus Hall. Special equipment and methods are also being developed for safe working with increased background radiation levels, slightly active dust and the use of beryllium in the torus.

To gain experience, remote handling equipment is being deployed during shutdowns, in hands-on or semi-automatic mode. Whilst the use of remote handling equipment during JET shutdowns provides an excellent testing ground for the effectiveness and reliability of equipment, it also means that as a consequence of the associated manpower requirements, only limited effort has been available for development of full remote handling procedures. Therefore, it is now envisaged that the remote handling for the D-T phase will concentrate on a few essential ex-vessel and in-vessel operations, including gaining remote entry into the vacuum vessel using the telescopic articulated remote mast (TARM) and the articulated boom with special end effectors or the MASCOT IV manipulators. The latter would allow, to a certain extent, carrying out non-planned activities inside and outside the vacuum vessel.

The large remote-controlled articulated boom has been de-

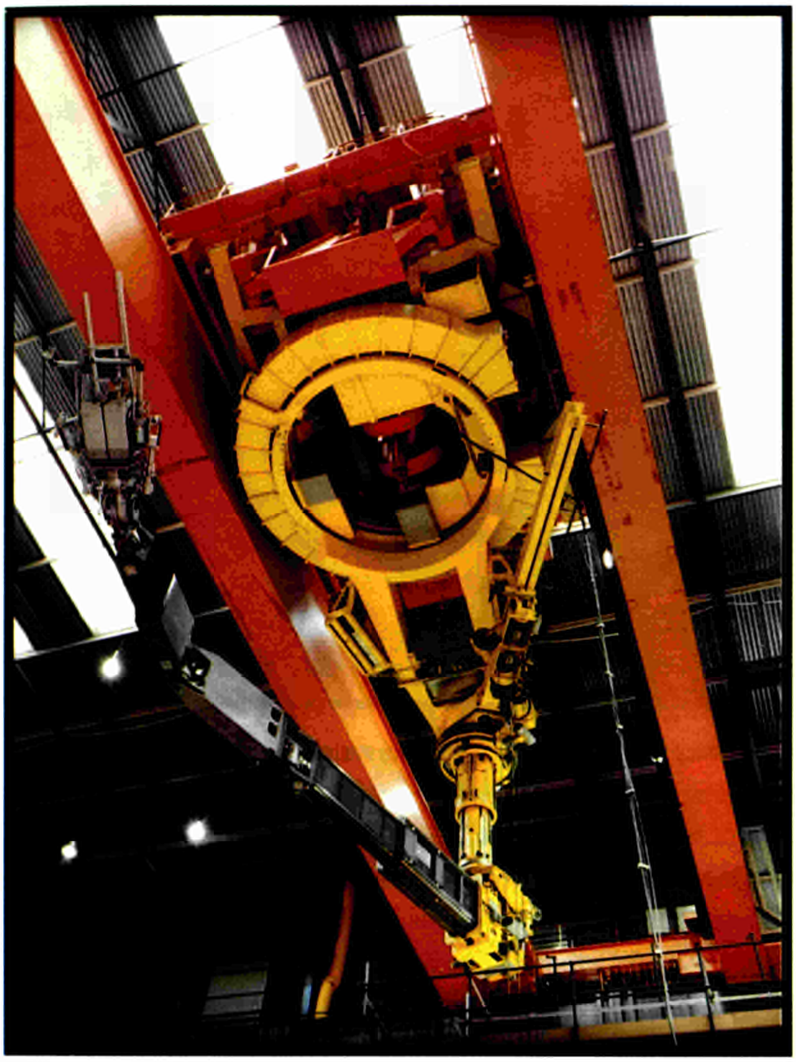


Fig.18: The TARM connected to the main crane with its horizontal telescoping section extended and MASCOT IV manipulator fitted

signed for reaching into the JET vacuum vessel for maintenance and repair when the machine has become radioactive. It can carry loads of up to 1 tonne at the end of its 9m horizontal reach. Subsequent to its use in the 1991 shutdown, the articulated boom underwent extensive stripdown and maintenance. No problems or unexpected wear in any components was found. Detailed improvements have been made to boom end-stops, camera arm drives and joint gaiters, and some modifications to improve maintainability of the drives have been incorporated. As part of the continued preparation for use in the remote phase, the boom video system has been commissioned. In order to prepare the boom for operation during the Pumped Divertor phase a new boom addition has been ordered. This will comprise three links and two joints, compared to one joint in the existing boom addition. With the new addition, the boom will be about one metre longer and will have an additional articulation to provide greater manoeuvrability within the more constrained in-vessel environment.

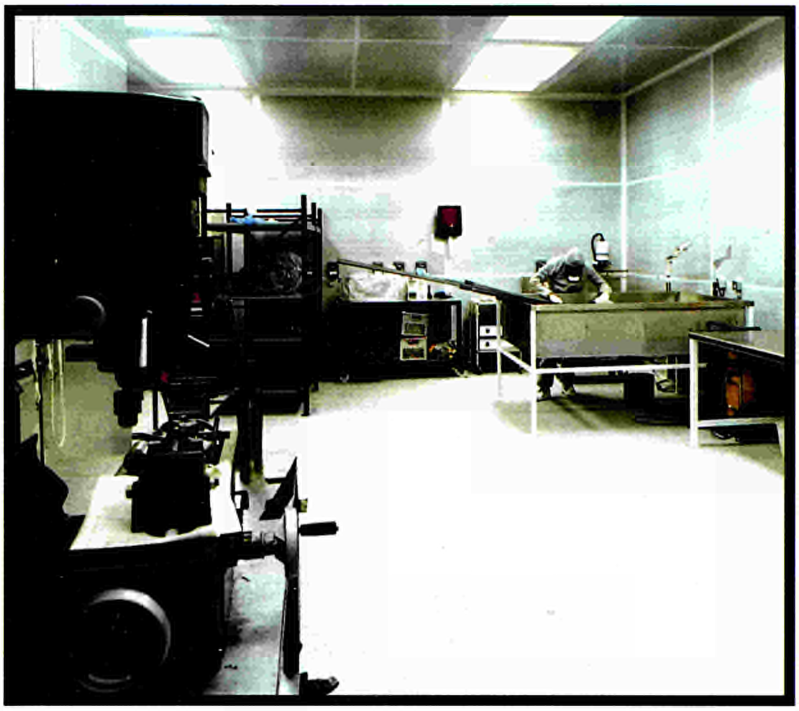


Fig.19: View inside the Beryllium Handling Facility in the Assembly Hall

During 1991, the Telescopic Articulated Remote Mast (TARM) was installed on the main 150 tonne crane on three occasions for various tests. The main crane has been modified and strengthened to accommodate the test findings and the work has culminated in a fully load tested TARM/crane combination certified for use in the 1992/93 shutdown for A2 antennae installation. Fig.18 shows the TARM connected to the main crane with its horizontal telescoping section extended and MASCOT IV manipulator fitted. Work to fully commission the electrical and control system of the TARM on the crane will be undertaken in 1992. The TARM gas, hydraulic, welding and cutting tool services have been progressively commissioned.

The In-Vessel Inspection System (IVIS) has been of great value in JET operation. Development work to improve the quality of image and reliability is continuing. Good progress has been made on the viewing tube, in which the glass cylinder has been replaced by a flat sapphire window. Testing at operating temperature and in vacuum has started. IVIS has been shown to be so important to JET operations, that is has been decided to enhance the system for use during the D-T phase.

The responsibilities of the Waste Management Group include the provision of the infrastructure in JET for waste handling (radwastes and beryllium-contaminated wastes) and the provision and maintenance of respiratory protection systems and equipment. Following the PTE, the 1992 shutdown will be the first time that JET has had to deal with considerable quantities of tritium-contaminated materials, when these are removed from the vacuum vessel.

During the first part of the year, considerable effort was involved in the operation of facilities for shutdown related work. These included the Torus Access Cabin (TAC), Beryllium Handling Facility (see Fig.19), Suit Cleaning Facility and PVC Workshop. At the same time, packaging and disposal of beryllium contaminated and active waste arisings was carried out. In the case of low level solid radwastes, the finalisation of both a quality assurance programme and waste management procedures to meet the requirements for disposal at the UK Repository was an ongoing activity for much of the year.

The Group was subsequently involved in planning for the preliminary tritium experiment and preparation of facilities and procedures for the 1992 Shutdown, where many of the operations will be affected by tritium as well as beryllium and induced activity. The main emphasis on the management of tritiated wastes concerned the Divertor Shutdown, but procedures and equipment were prepared to handle water leaks in case of an incident.

During the initial period in the 1992 shutdown, a considerable volume of active components and associated contaminated secondary wastes will be produced. The materials will be contaminated with tritium as well as beryllium and activation products. A new Waste Handling Facility has been designed (see Fig.20) and is being

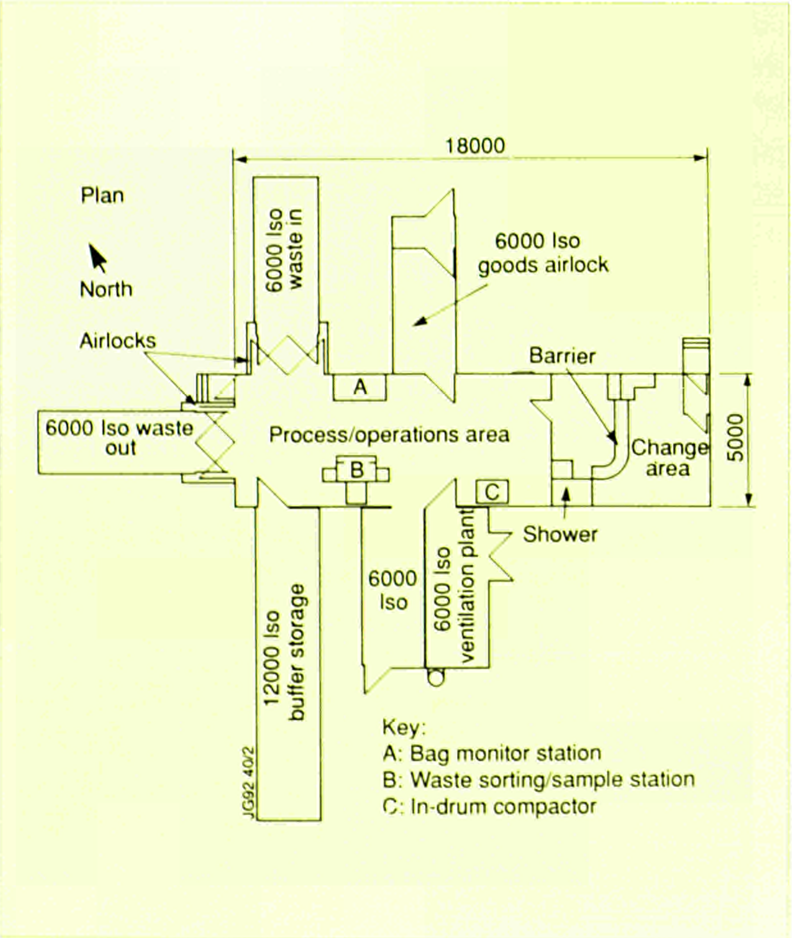


Fig.20: Layout of new Waste Handling Facility

constructed for commissioning before the start of the shutdown. The facility will be used to carry out the sorting, packaging, analysis and preparation for disposal of the materials and wastes.

Control and Data Management

The JET Control and Data Acquisition System, CODAS, is based on a network of minicomputers. It is the only way to operate JET and it allows centralised control, monitoring and data acquisition. The various components of JET have been logically grouped into subsystems, such as Vacuum, Toroidal Field, Lower Hybrid additional heating, etc. Each subsystem is controlled and monitored by one dedicated computer interfaced to the machine and its diagnostics, through CAMAC instrumentation and EUROCARD-based signal conditioning. Embedded front-end intelligence is implemented through CAMAC-based microprocessors for real-time applications. The actions of the various computers are coordinated by a supervisory software running in the Machine Console computer.

As mentioned in 1990 Progress Report, it had been decided to move CODAS to a UNIX environment in order to maintain its services in the best possible way until the end of the Project in 1996. The structure of the system will be similar. The CAMAC interface will be retained and the Man Machine Interface (MMI) will be upgraded to use Windows. The changeover will be implemented during the 1992 Shutdown.

The Computing Service is based on an IBM 3090 three-way processor mainframe with two vector facilities. There are 90GBytes of disc storage and a further 240GBytes of IBM mass storage. The JET Mainframe Data Processing Centre is housed in a specially designed building at AEA Technology, Harwell Laboratory, and operated for JET under contract by a team from that Laboratory.

The JET Computing Centre has been operating since June 1987 and the computing load has grown significantly since that date. In order to maintain good interactive response and also accommodate an increasing load of background batch work, the central computer was upgraded in February 1990 from an IBM 3090/200E to an IBM 3090/300J with three processors, two vector facilities and 196 MBytes of memory (64MB central and 128MB expanded). This has almost double the processing capacity. The upgrade has permitted a significant growth in all areas of the mainframe computing workload, most critically in the intershot processing, the CAD work from the JET Drawing Office and interactive (TSO) work. These improvements have significantly enhanced the Project's Design and Data Processing capabilities.

The Data Management Group provides the contact between the

Control and Data Acquisition

Due to the high number of components and their distribution throughout a large site, the operation and commissioning of JET is supported by a centralised Control and Data Acquisition System (CODAS). This system is based on a network of Norsk Data minicomputers interfaced to the experiment through CAMAC instrumentation (including front end micro-processors) and signal conditioning modules. The various components have been logically grouped into subsystems with each one controlled and monitored by a computer. After a pulse, all the information from the subsystem is merged together into a single file on the storage and analysis computer. This file is then transmitted to the IBM mainframe computer located at AEA Technology, Harwell Laboratory, for detailed analysis. A summary of information from the JET pulses is held in the JET Survey Data Bank.

users, operators and system programmers, through the Help Desk Service, backed up by specialists in the Group. This ensures the smooth running of the system. The data communications between the JET site system and the Computer Centre are mainly the responsibility of CODAS Division and these have operated reliably.

Technical Preparations for the Preliminary Tritium Experiment

To gain timely information on the introduction of tritium into JET, including retention of tritium in wall materials, operation of diagnostics, radiation monitoring and waste handling, it was decided in early 1991 to prepare for a limited tritium experiment which would involve only a few plasma pulses. This would restrict the total amount of tritium used in the experiment so that the resulting activation of the vacuum vessel would not increase significantly above the level resulting from D-D operation during 1991 and early 1992. Then, the impact of the first tritium experiment on the major shutdown planned for early 1992 would be minimised.

Preparations for the experiment included a physics programme to define the optimum plasma parameters, method of injection and diagnostics requirements, as well as a technical programme to design and install special equipment to inject tritium and recover tritiated exhaust gases. In addition, the tokamak, its diagnostic equipment and auxiliary systems had to be brought to a state of readiness for the experiment. Generally, only systems essential for the experiment were used. The pellet injectors and radio frequency handling systems and many diagnostics were not used and these systems communicating with the torus vacuum were isolated.

Objectives and Planning

The main motive for carrying out the first tritium experiment was to gain important technical and physics information for the full D-T phase of JET. The principal objectives are summarised, as follows:

i) to produce more than 1MW of fusion power in a controlled way;
ii) to validate transport codes and provide a basis for predicting accurately the performance of deuterium-tritium plasmas from measurements made in deuterium plasmas; to establish for these plasmas the consistency of different experimental measurements; and to calibrate diagnostics;
iii) to determine tritium retention in the torus walls and the neutral beam (NB) injection system;
iv) to demonstrate the technology related to tritium usage (tritium NB injection, cryopumping and tritium handling);
v) to establish safe procedures for handling tritium in compliance with the regulatory requirements.

Prediction, Interpretation and Analysis

The prediction of performance by computer simulation, the interpretation of data, and the application of analytic plasma theory are of major importance in gaining an understanding of plasma behaviour in JET.

- *Prediction work continuously checks the measured behaviour against the different computational models, and provides a basis for long term programme planning;*
- *Interpretation plays a key role in the assessment of plasma performance, and hence in optimisation studies and programme planning;*
- *A major role of analytic theory is to compare the observed behaviour against that expected from existing analysis, and to modify the latter when there is divergence.*

A central task is to provide a quantitative model of tokamak plasmas with the ultimate objective of including all the important effects observed in JET and other tokamaks. It is preferable to understand each effect theoretically, but in some cases it may be necessary to rely on an empirical description.

For carrying out these tasks it is important that JET data is held in a readily accessible and understandable form.

After an initial study on the feasibility to carry out a limited D-T experiment, two Task Forces were formed to prepare for the experiment. A Physics Task Force (P) was charged with studying physics issues, including definition of optimum plasma parameters, prediction of expected performance, as well as selection of the actual tritium injection scenario. A Technical Task Force was charged with procurement of tritium, its safe and secure storage on the JET site, design, testing and installation of tritium injection systems, and an exhaust gas collection system. Further activities included preparation of a safety analysis report, obtaining statutory and other approvals, installation of radiological protection instrumentation and preparation of waste management facilities and procedures.

Extrapolation of the best results obtained in D-D discharges during earlier campaigns in JET to D-T pulses predicted that fuelling with 14% gas mixtures (1 part tritium to 6 parts deuterium) would yield a total fusion power of 0.9 -2.4MW depending on which prediction model was used (provided that the wall recycled material was of similar isotopic composition). This tritium concentration in the plasma was used as the basis for the first tritium experiment.

The proposal to use PINIs for tritium injection showed that this approach offered certain advantages over gas puffing:

i) the tritium beam would have a known deposition profile which, dependent only on plasma density, could be peaked in the plasma centre;
ii) the total tritium introduced into the torus would be minimised;
iii) development, testing, installation and commissioning would have minimal impact upon tokamak operation.

The experiment was split into two phases. The initial calibration phase used a very weak tritium-deuterium mixture (1% tritium in deuterium) for the 'tritium' PINIs and would include checking of diagnostics, especially neutron diagnostics. This phase involved 10 to 15 tokamak discharges. The second phase, the experiment proper, used pure tritium for the two 'tritium' PINIs and involved two useful tokamak discharges.

One of the main constraints on the experiment was to keep the activation resulting from D-T neutrons at a low enough level to allow a prolonged period of work inside the vacuum vessel during the shutdown planned to start in February 1992. The total production of D-T neutrons was therefore restricted to 1-2 x 10^{18} resulting in an in-vessel dose rate of 25-50 μSv/hour some twelve weeks after the experiment. This dose rate is lower or comparable to that expected from D-D neutron activation (~50 μSv/hr) from the 1991-1992 experimental campaign.

For this reason, the total amount of tritium to be used was limited to 74TBq (2,000 Ci) which was sufficient for approximately six

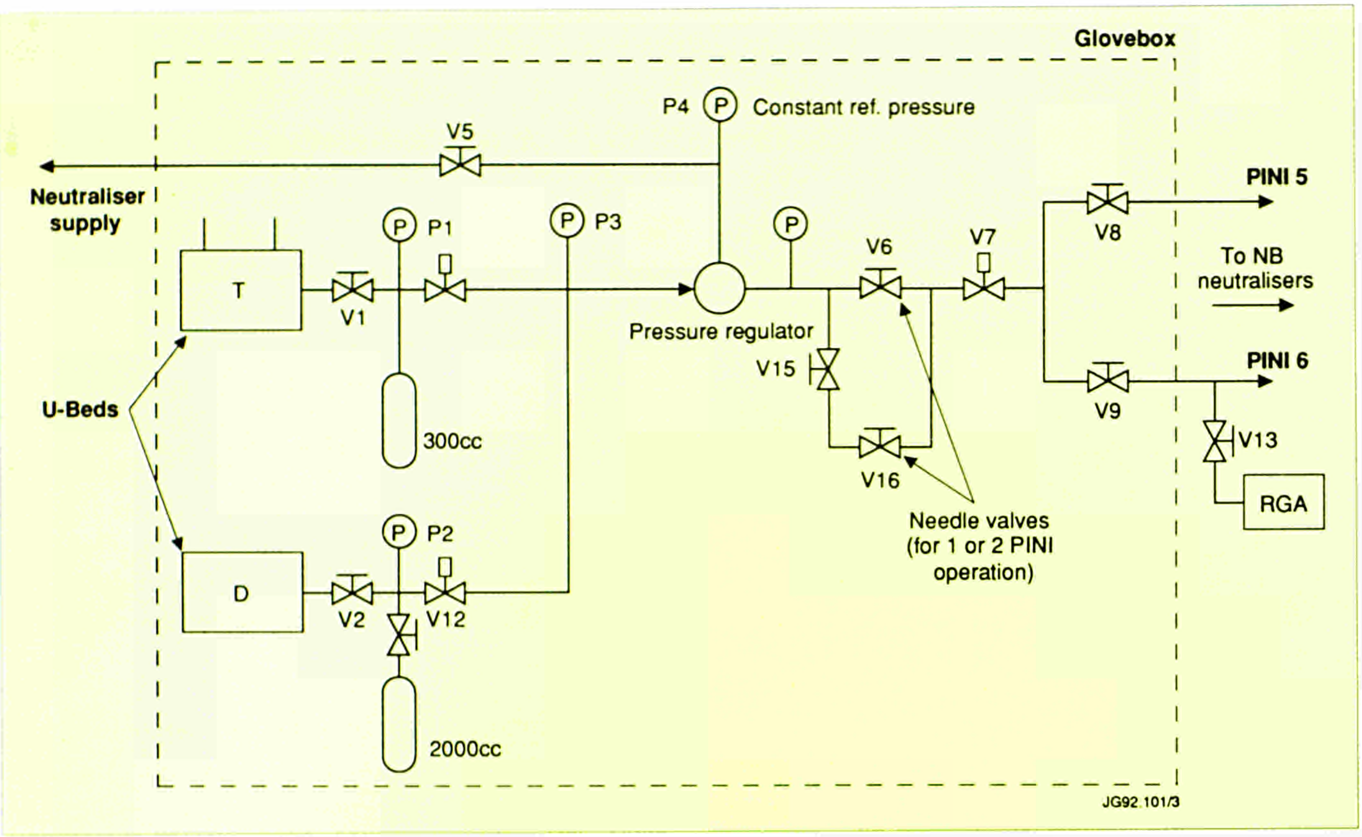


Fig.21: Layout of tritium gas supply system

injection pulses of 2s duration with two beams. Two or three of these pulses would be used for injection into the plasma, the others would be used as commissioning pulses of the tritium beams which would not involve injection of tritium into the torus.

For the full D-T phase, an exhaust gas processing system, the JET Active Gas Handling System (AGHS), will be available to remove impurities from the hydrogen isotopes, to separate purified hydrogen into isotopic fractions and to re-supply the isotopic fractions to the tokamak subsystems. The installation of the AGHS is not yet complete and, therefore, for the initial experiment, special equipment had to be installed. This not only involved a gas collection system, but also modifications to the neutral injectors for tritium introduction.

Tritium Gas Introduction

Two PINIs of one neutral injector were modified for tritium usage. The gas was supplied from two uranium beds (U-beds), one loaded with deuterium (for commissioning) and the other with tritium (for fuelling) mounted close to the PINIs outside the beamline magnetic shielding. The remaining PINIs of the Octant No: 8 beamline were used for deuterium injection, both with and without concurrent tritium injection. The gas for the six deuterium PINIs was supplied under existing arrangements. The gas supply arrangements for the two tritium PINIs are shown schematically in Fig.21. All tritium

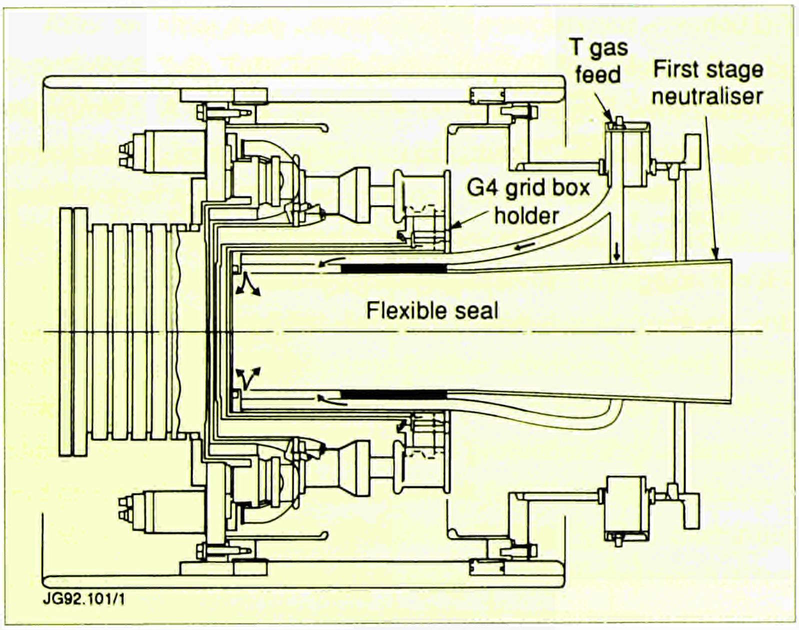


Fig.22: Modified JET PINI for use with tritium

primary containment and associated instrumentation were installed within a proprietary glass re-inforced plastic glove box. The glove box was connected by temporary ducting to an authorised discharge stack equipped with ionisation chamber and sampling system to monitor tritium releases.

The standard PINI configuration utilized two gas feeds; one at high potential for the plasma generator (ion source) and one at ground for the neutralizer gas. For technical reasons, it was decided to dispense with the gas feed at high potential and to introduce the source and neutralizer gas from a common feed at ground potential. The final optimised configuration is shown in Fig.22.

Optimisation of arc and gas stabilisation times immediately prior to beam extraction enabled these to be reduced by a factor two or three to 0.5s, which reduced the overall tritium gas consumption. Two PINIs were modified for use as tritium injectors, both of which were conditioned and fully characterised at 80kV prior to installation on the tokamak.

The major preparatory work related to the design and manufacture of the control and instrumentation for the Gas Introduction System and its integration into the operational software system which controlled the injectors. Computational studies of the power and particle deposition profiles of tritium and deuterium in the envisaged target plasmas, were also carried out with a view to maximising deposition in the plasma centre.

A basic feature of the control system was the use of a single set of controls plus cables and pneumatics for operation of *either* the deuterium U-bed *or* the tritium U-bed. This ensured that there was no possibility of cross-contamination of the deuterium by tritium, or vice-versa. In addition, it enabled considerable operational experience and familiarity to be gained with the total system using deuterium prior to the PTE. The main disadvantages were the need to enter the Torus Hall to effect the change-over and the time taken to cool one U-bed and heat the other. However, these disadvantages were outweighed by the simplicity. All operations relating to installation and commissioning, change of D_2 and T_2 U-beds, operation of hand valves were governed by extensive strict written procedures.

The Gas Collection System

The operational functions of the Gas Collection System (Fig.23) were:

i) To act as a temporary tritium-compatible primary pumping system, for the duration of the experiment;

ii) to constitute a measuring unit to account for tritium exhausted from the torus and the two neutral beam injectors;

iii) to separate hydrogen isotopes from the residual exhaust gases and safely store those isotopes;

iv) to store tritiated residual gases for future reprocessing;

v) to detect, safely handle and facilitate recovery from air inleakages into the torus, neutral injector, vacuum transfer lines or the collection system itself;

vi) to assist recovery from a water leak by providing means for pumping and collecting water vapour from the torus or neutral injectors.

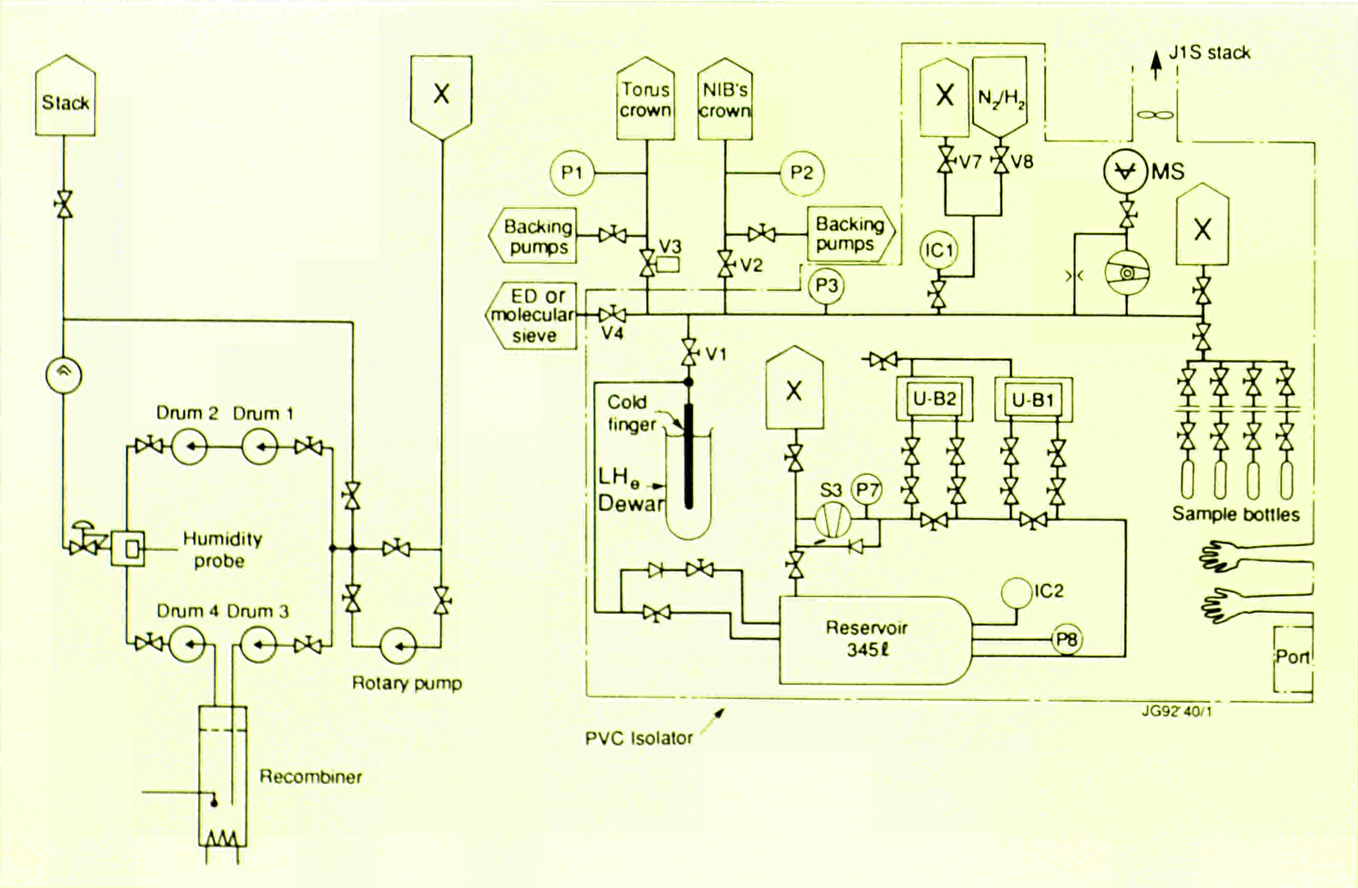


Fig.23: Layout of gas collection system

Safety Analysis Report

A detailed probabalistic safety analysis report was prepared to appraise the radiological risks, principally those resulting from the introduction of tritium into the torus and subsequent collection in the first tritium experiment. Due to the fact that failure rate data on components used in a plasma environment are very limited, failure rates based on JET operating experience since 1983 were widely used, taking account of factors intrinsic to JET design and operation. The safety analysis demonstrated that for all foreseeable worst cases of each category of accident, the estimated risks (based on conservative assumptions of occurrence probability and release consequences) adequately satisfied standards for public and worker risk.

Safety Matters and Instrumentation

Radiological Protection Instrumentation was prepared and implemented for glove-box monitoring, working area monitoring, stack monitoring and environmental monitoring. Further tasks involved the training of staff, relating to operation of special equipment, and concerning the general handling of tritium, which involved a large number of JET staff. Other activities included the preparation of a measurement programme to assess tritium retention in the wall materials, during and after the experiment and the preparation of a site emergency plan as well as installation of an emergency control desk in the Control Room.

Implementation of the Experiment and Clean-Up

Installation, commissioning and operation of the additional equipment was undertaken according to procedures established and approved within the Project. During the experiment, daily meetings were held between the two Task Forces, where all aspects of the experiment were discussed and the programme for the next day decided. Any changes of procedures that became necessary were fully discussed and decisions recorded.

At the start of the clean-up phase, ~1.6TBq of tritium was estimated left in the torus, whereas all tritium injected into the neutral injector had been accounted for, within the accuracy of measurement. Subsequent tritium neutral injector regenerations delivered 1.85TBq and 0.26TBq, whereas the tritium recovery from the torus rapidly dropped from 16.6GBq per discharge during the first operational day after injection to 0.93GBq per discharge during the eighth operational day after the injection experiment. Several techniques, ie glow discharge cleaning, gas purging, etc, were tested. However, the rate of evolution continued to fall and allowed the torus to be reconnected to its normal backing pumps, exhausting directly into the monitored discharge stack three weeks after the start of the clean-up phase.

Up to the end of 1991, ie. 10 weeks after the start of the clean-up phase, only ~ 0.2 TBq of tritium were discharged. This amounts to 0.25% of the total amount of tritium handled and represents a very small fraction of the radioactive discharge authorisation, indicating that the injection system and Gas Collection System operated well.

Special equipment based on components used in the JET Active Gas Handling System worked very well and according to specifications. The experience gained with some of the components may however lead to some modifications in the JET Active Gas Handling System, in particular with respect to U-beds for intermediate storage and stack sampling systems.

Valuable information has already been gathered on the torus decontamination and tritium retention of vacuum vessel walls and in-vessel components. However, due to uncertanties associated with measurement accuracy, the residual amount of tritium inside the tokamak can only be finally quantified when samples of first wall materials are analysed during the 1992 shutdown.

Decontamination of the tritium neutral beam injector took longer than originally estimated. As a consequence, additional U-beds had to be installed to recover tritiated exhaust gases, which required the preparation of a special safety assessment. Preparations for the full D-T phase will take this situation into account.

The tritium experiment has initiated a study of the waste arising from tritium operation. The study revealed that detritiation of in-vessel components would be required. Decontamination techniques which might include baking and surface treatment are now under investigation. A waste handling facility to be used during the 1992 shutdown and later during the full D-T phase is under construction.

Summary of Machine Operation

During 1991, JET operations were essentially made up of three main periods :

a) *The first period (Week 11 to Week 19)*

This period included the following activities :

- CODAS and power supplies commissioning in parallel with shutdown activities. Initially, this was carried out in extended-day operation (Weeks 11-18), but as plasma operation approached, double-shift-day operation started in Week 19.

b) *The second period (Week 23 to Week 35)*

This period included the following activities :

- Power Supplies and plasma commissioning (Week 23 to Week 25). Plasma commissioning took some time as the vessel required reconditioning following the vessel entry;

- Execution of the experimental programme by the different task forces in 6-day working weeks, broken only by Public Holidays, JET Open Days (end Week 26) and maintenance and remedial days (Monday and Tuesday of every third week);
- Plasma operation also involved the study of new in-vessel elements installed during the shutdown (eg. lower and upper dump plates, modified gas introduction system, new diagnostic systems and the NBI system modified to operate at 140kV);
- Alternating current (AC) operation of the plasma for 4 days in Week 31. A plasma current pulse of 2 MA in one direction with a flat-top of 6s was reversed to achieve a current of 2MA in the reverse direction for a flat-top of 6s. Additional heating in both phases of the current pulse was carried out.

c) *The third period (Week 39 to Week 51)*

This period followed a shutdown in which preparations for the Preliminary Tritium Experiment (PTE) were made. The time for preparations was extended due to rapid venting of the vessel caused

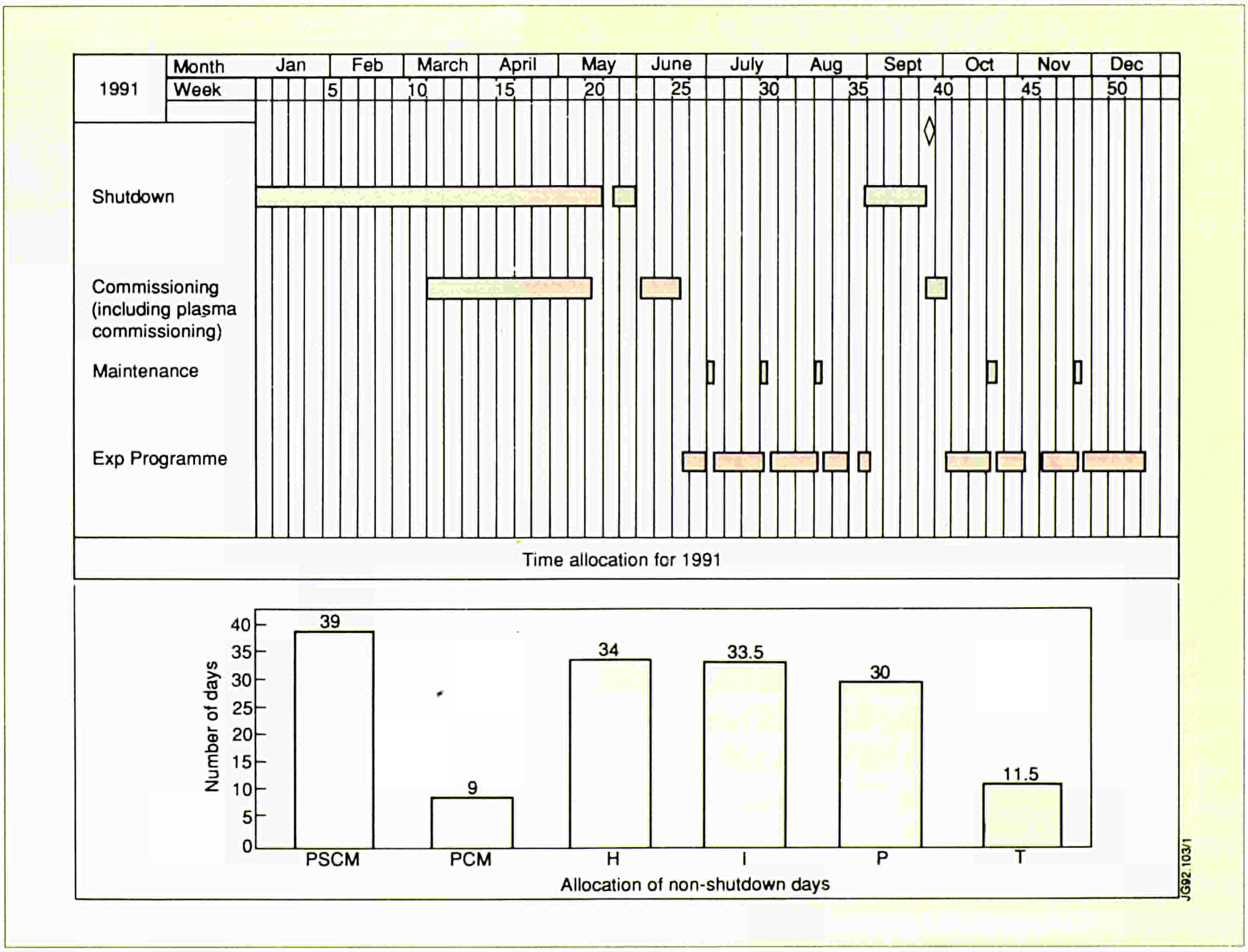


Fig.24: Allocation of days on which machine pulses were performed during 1991. (PSCM = Power Supplies and CODAS Commissioning, PCM = Plasma Commissioning, H = Task Force H, I = Task Force I, P = Task Force P, T = Tritium Task Force)

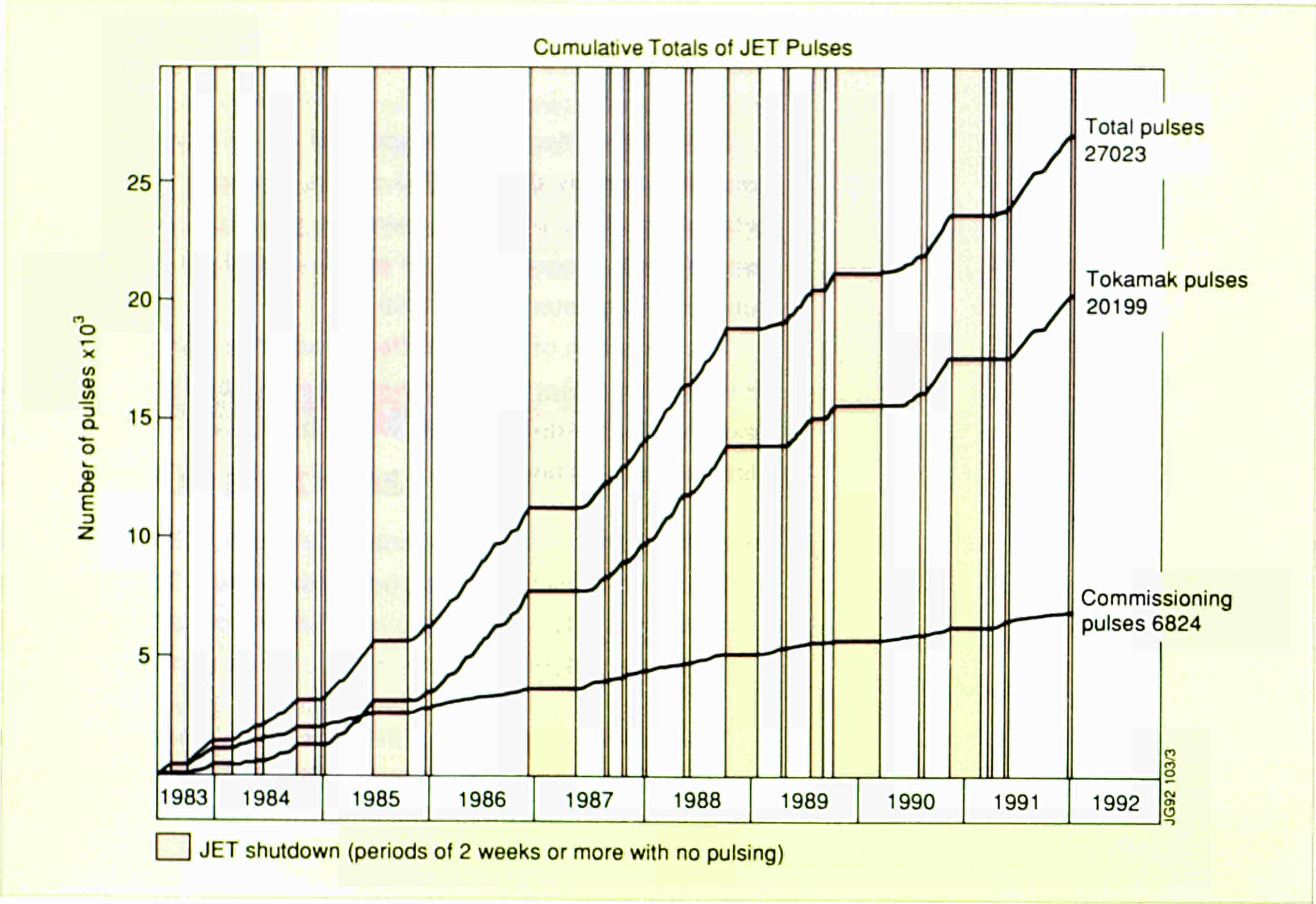


Fig.25: Cumulative totals of JET pulses - 1983 to 1991

by a vacuum leak of the in-vessel inspection system. Repairs were effected on a vacuum bellows and the opportunity was taken to replace tiles on the upper X-point dump plates. These new carbon tiles were carefully machined and accurately installed to spread the heat load more evenly.

This period included the following activities :

- Installation and commissioning of a uranium bed supplying deuterium to two sources of one of the neutral beam injectors. This was carried out to gain experience in operation of the NBI gas introduction system used in the PTE;
- Execution of the experimental programme in 6-day working weeks broken only by maintenance and remedial days and a two-day period involving final preparations for the PTE;
- The Preliminary Tritium Experiment: this experiment (which is described in detail later in the report) involved tritium neutral beam injection and the operation of a gas collection system with tritium recovery. Special operational precautions were taken to ensure the success of this operation. During and following the actual experiment, machine operations were carried out in two-shift days and the gas collection system was also operated during the third shift;

- Neutron production was restricted which meant that different gases for the plasma and neutral beam heating were used;

In 1991 the machine was operated for 157 days, which was considerably more than in 1990. In fact, this approached the sum total of the 1989 and 1990 operation periods. The time for the experimental programme (109 double-shift days) was therefore substantially greater than in 1990.

The allocation of time to different activities is shown in Fig.24. The experimental programme was carried out by three Task Forces and the double-shift days in which these were involved were distributed as follows :

Task Force H: High Performance	31.2%
Task Force I: Impurity Transport & Exhaust	30.7%
Task Force P: Physics Issues	27.5%
Tritium Task Force: PTE	10.6%

The number of pulses in 1991 was 3493, bringing the total number of JET pulses to 27023 (Fig.25). The percentage of commissioning pulses in 1991 (about 20%) was roughly the same as in 1990.

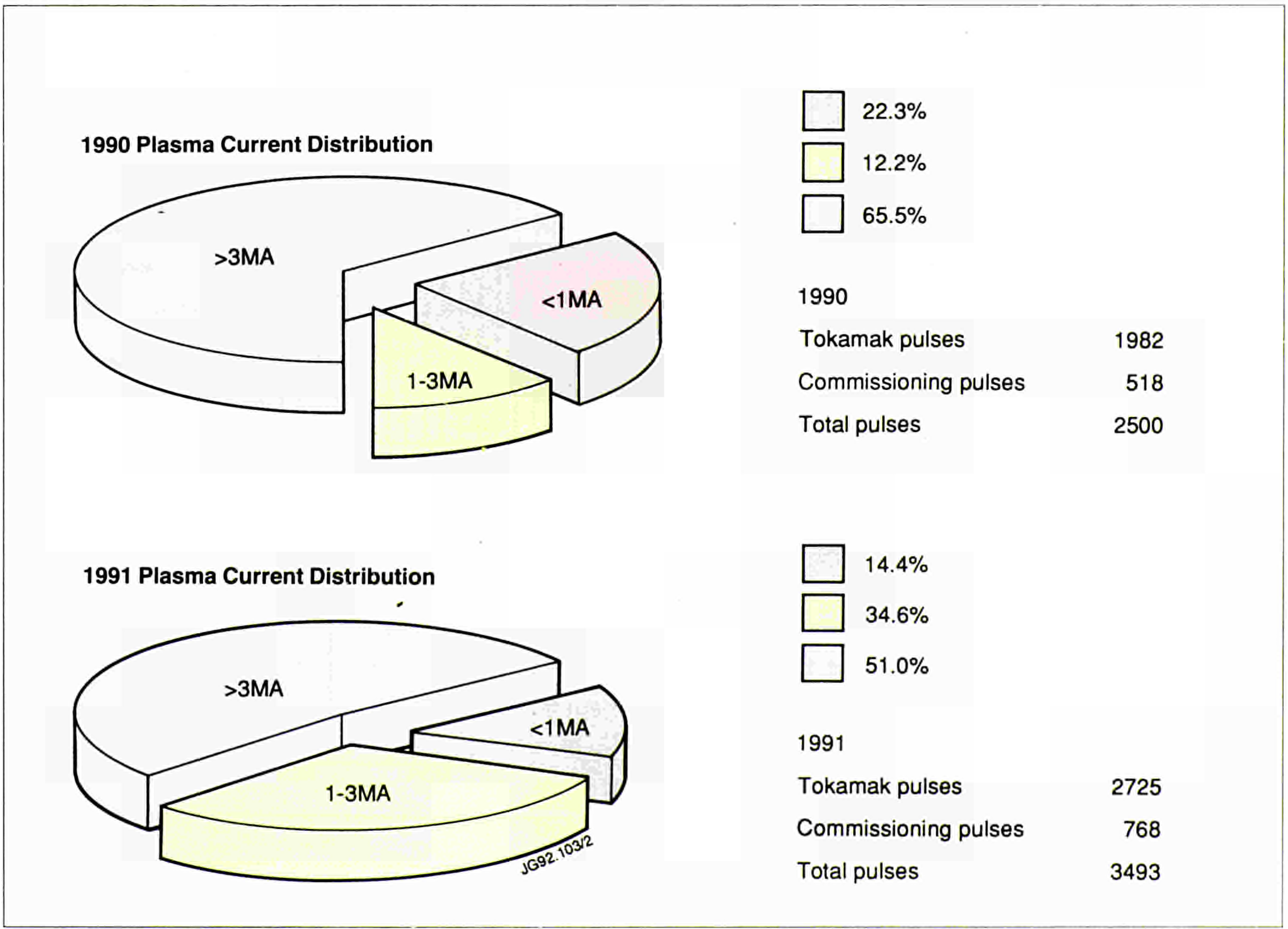


Fig.26: Comparison of numbers of pulses and distributions of plasma currents for 1990 and 1991. The distributions are for all pulses.

To avoid the possibility of damage to in-vessel components before the PTE, plasma currents were limited in 1991 to the smallest values compatible with programme requirements. This is shown in the plasma current distribution (Fig.26), where the higher plasma currents (>3 MA) make up only 51.0% of the total (cf. 65.5% in 1990). Nevertheless, the percentage of low plasma current pulses (which include plasma commissioning and "failed" discharges) is also much reduced.

Technical Developments for Future Operations

Considerable effort was devoted during the year to the design and procurement of equipment for installation on the machine during future major shutdowns. Reference should be made to the section on the Future Programme of JET to relate these technical developments to the overall JET Programme.

Lower Hybrid Current Drive

The Lower Hybrid Current Drive (LHCD) technique will be the main method of decoupling the plasma current and temperature profiles in JET. The main objectives of current drive and profile control are:

- to suppress sawteeth activity and to benefit from higher core reactivity by sustaining peaked profiles of both density and temperature;
- to modify local values of the current gradient and improve energy confinement in the plasma centre;
- to assess the current (and efficiency) required for non-inductive operation of large tokamaks.

The JET LHCD system will be powered by 24 klystrons operating at 3.7GHz produce a total power of 12MW. The launcher will produce a narrow wave spectrum with a parallel wave index which can be varied from 1.4 to 2.4. The horizontal row producing this spectrum is composed of 32 waveguides; with 12 rows, the total number of waveguides is 384. The system is summarized in Table.5.

The system will be installed in JET in two stages. The first prototype stage was installed and tested during 1990 and 1991. The first stage was composed of two prototype launchers, one built using the same technique as foreseen for the final system and one built by CEA Cadarache, France, using the technique developed for Tore Supra. The prototype launcher comprising one third of the number of waveguides of the full system was tested and installed on the torus. The corresponding eight klystrons and associated drive, phase control and power transmission system were also installed and commissioned, and the system used in operation during the experi-

Table.5: LHCD System Parameters

Generator	**Prototype system**
Frequency	3.7 GHz
No of klystrons	24
Power (launched)	10MW
Launcher	Multijunction type
Fixed phasing in the multijunction	90 degrees
Central $N_{\parallel}$	1.8
Range of $N_{\parallel}$	1.4 - 1.8
No of waveguides in horizontal row	32
Phase accuracy	10 degrees
Width of the $N_{\parallel}$ spectrum	0.2
Directivity	80 %
Density limit	8×10^{20} m^{-3}
Power handling	4-5kW cm^{-2}
Estimated drive current	
at $n_e = 2 \times 10^{19}$ m^{-3}	3.6MA
at $n_e = 5 \times 10^{19}$ m^{-3}	1.2MA

mental campaign. Up to 1MW has been coupled to the plasma for 50s during long pulse operation, and up to 2.2MW for 8s and 2.5MW for 1s have been coupled to the plasma in a large variety of plasma conditions.

With the prototype launcher, full current drive in almost steady-state conditions has been achieved with LHCD alone at plasma current of 0.4MA and in combination with ICRF at plasma currents up to 1.5MA. Current drive efficiencies (measured as the ratio [nRI/P]) have been shown to increase with the electron temperature, as indicated in Fig.27. Efficiencies up to nRI/P=0.4x10^{20}m^{-2}MA/MW have been obtained. Although these efficiencies are the highest achieved, they are still lower than those required to fully drive the plasma current in a reactor, which is estimated to be between 0.4 and 0.8x10^{20}m^{-2}MA/MW at densities near 10^{20}m^{-3}.

The launcher for the next phase (L1) has been modified to take into account the pumped divertor geometry and is now in the assembly stage. A view of the end plate where the forty-eight double vacuum windows will be welded is shown in Fig.28. This launcher will be able to match the so-called "fat plasma" configuration, but it will not be possible to use the launcher with the alternative 'thin plasma' configuration. To overcome this prob-

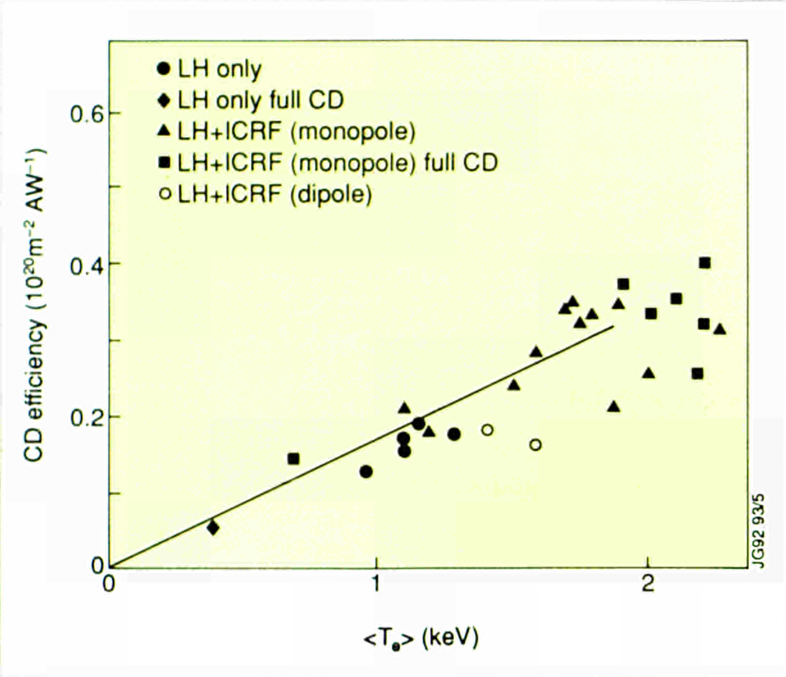


Fig.27: Current drive efficiency versus temperature for the prototype launcher.

lem, a conceptual solution has been studied which is based on a new concept: the 'hyperguide'. The hyperguide is a larger overmoded waveguide of ~5m long and may transmit large RF power with good vacuum conditions allowing avoidance of the electron resonance zones where breakdowns are likely to occur. Such a system can make use of the present vacuum vessel and of

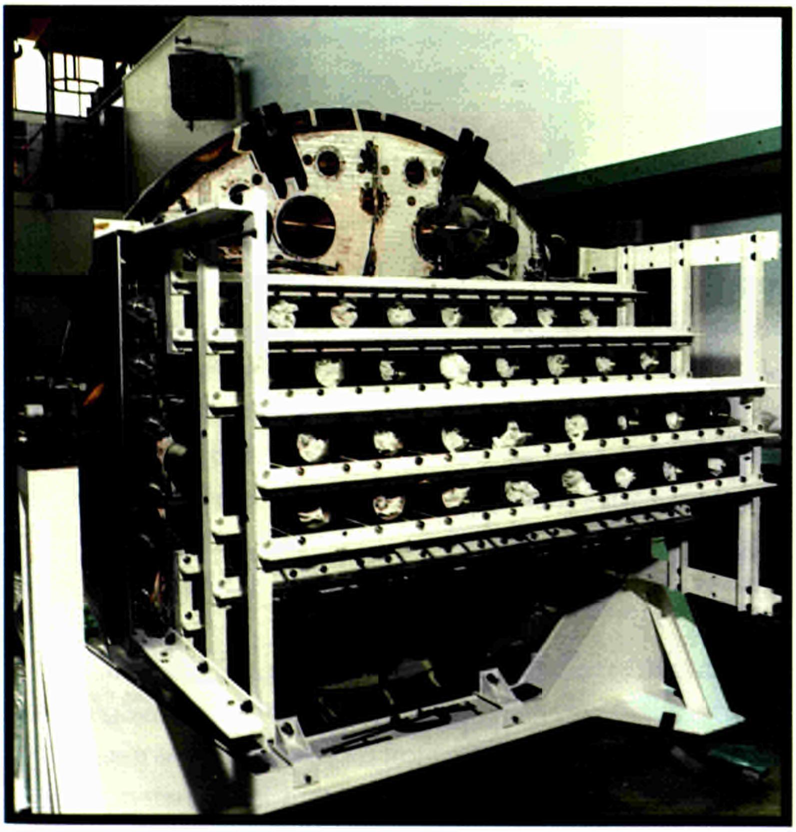


Fig.28: View of the L1 Launcher end plate in the JET Assembly Hall

the L1 end plate. It will be less costly to build than the L1 launcher and might accommodate either a short conventional waveguide mouth or possibly a quasi-optical coupler. Studies are still continuing on this new development.

Pellet Injection

The injection of solid hydrogen pellets is one method of providing a particle source inside the recycling boundary layer of a future fusion reactor without simultaneously depositing excessive power. The ablation of the pellet by hot plasma electrons requires very high speed pellets, in order to penetrate beyond the q=1 surface to the plasma centre. So far in JET, measurements have only been carried out in a limited velocity range up to 1500ms^{-1}.

Until the latter part of the year, JET had planned to implement for the divertor and later phases of JET a repetitive high-speed pellet launcher system - the Advanced Pellet Launcher (APL). This would have been alongside an intermediate-speed repetitive

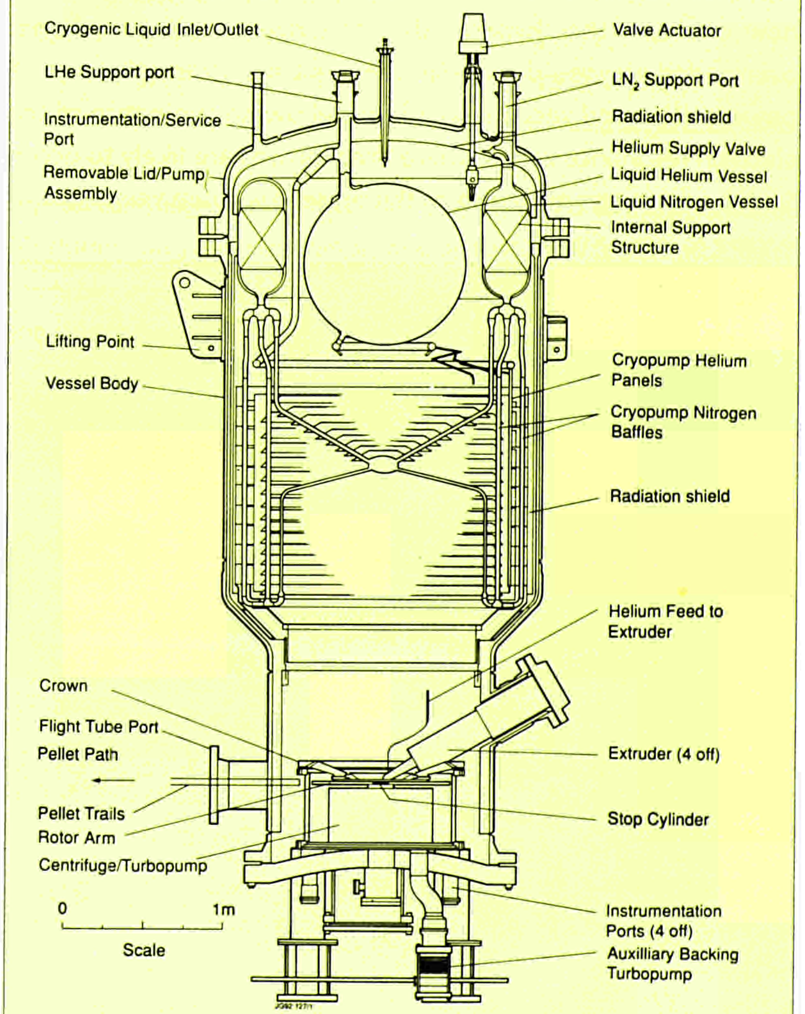


Fig.29: The Gas Centrifuge

pneumatic launcher, JPL II, as a replacement for the present launcher (JPL I), built by Oak Ridge National Laboratory (ORNL), USA. In addition, there would be a low-speed pellet centrifuge. Due to the budgetary situation at JET and in the USA, the APL will now not be pursued for financial reasons, and the negotiations on the supply of an upgraded ORNL Launcher, JPL II, were not fruitful. Although JET owns the rights to proceed with the conceptual design of the JPL II, jointly carried out by ORNL and JET, insufficient funds and manpower made it impossible to proceed. However, JET decided to follow up the work on the centrifuge and on the high speed prototype launcher, which had not been successfully completed. The prototype employ a 6mm single-shot, two-stage gun launcher with a pellet velocity of ~4kms^{-1}.

The pellet centrifuge is to provide a source of deuterium particles at varying depths beyond the recycling layer and with it a minimum recycling flow into the divertor, which is sufficiently strong to sweep impurities into the divertor and hinder impurities to flow back from it. The injection parameters chosen are pellet sizes of 1.5 - 3mm at repetition frequencies up to 40s^{-1} at speeds of 50-600ms^{-1} for long pulses approaching 1 minute. At the upper end, this provides up to 1000mbarℓs^{-1} of gas flow regardless of pellet speed and the penetration depth variation is over a factor 12. The plan is to position the centrifuge at Octant No: 2 midplane, alongside the pneumatic pellet injector. The detailed design (see Fig.29) of the vacuum vessel, the centrifuge and the cryopump, needed to absorb the gas load evaporated from the pellets during acceleration and guidance, has been finalised. Advice has been given, by IPP Garching, Germany under an Article 14 contract, on implementation of the pellet cetrifuge in JET

Tritium Handling

Following the extension of the Project to the end of 1996, the Active Gas Handling System (AGHS) will be required to be fully operational before the start of the full D-T phase of JET scheduled for 1996. As a result of the planning for and implementation of the preliminary tritium experiment during 1991, substantial effort had to be devoted to these tasks. As a consequence, work on the preparation for the full D-T phase, including preparation of the Final Safety Analysis Reports and the installation of the AGHS, slowed down.

At the end of 1991, mechanical plant installation of the Active Gas Handling System (AGHS) was nearing completion with the exception of the gas transfer connections from the AGHS to the tokamak and auxiliary systems, which can only be installed during the 1992/93 shutdown. Electrical connections were well advanced, commissioning

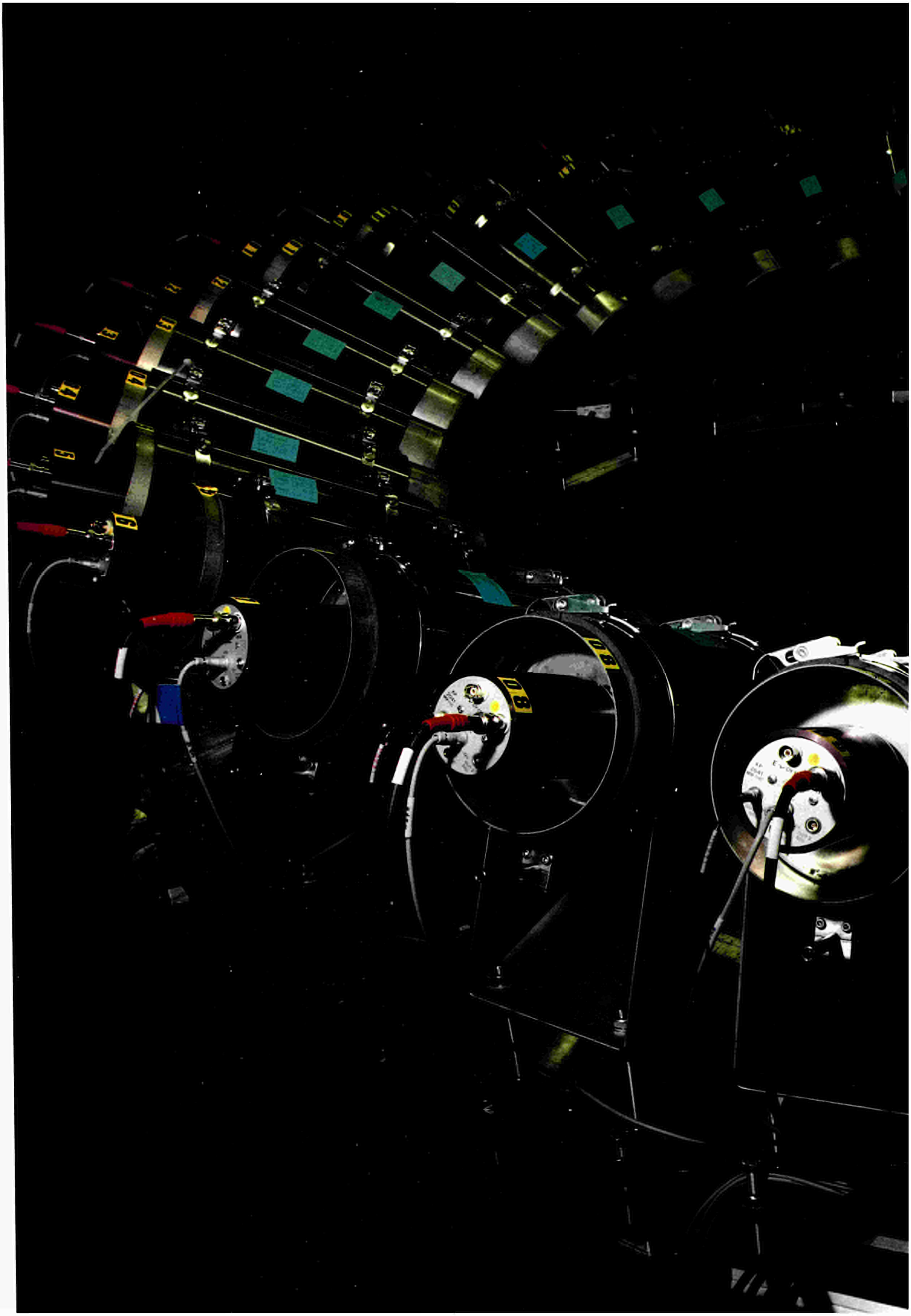
Fig.30: The Active Gas Handling System showing the five cryogenic forevacuum modules with buffer tanks and crodistribution system in the background.

of subsystems, including cryodistillation system and exhaust detritiation system and active drainage system were underway. Major installation activities during 1991 included the five modules of the cryogenic forevacuum system, the impurity processing system, the Uranium-beds for intermediate storage, gas chromatography and product storage systems. The control system is operational and software tests are in progress. Fig.30 shows the five modules of the cryogenic forevacuum system with buffer tanks and the liquid helium and liquid nitrogen distribution system in the background.

The safety assessments for the main process systems were completed in 1990. Effort in 1991 has concentrated on assessment of the common systems, such as building ventilation, services, valve box over/under pressure protection systems forming part of the Final Safety Analysis Report. The final design of the hard-wired protection system has been established and will be reviewed to demonstrate compliance with the reliability assumptions made in the plant Design Safety Reviews. A fire hazard assessment has been completed for the AGHS building which has concluded that the overall hazard is low.

Radiation Dose Assessments for releases of tritium and activation products to the atmosphere and River Thames have been carried out in connection with the application to Her Majesty's Inspectorate of Pollution (HMIP) for discharge authorisations. In addition, a new study has been initiated with the Canadian Fusion Fuel Technology Project (CFFTP, Canada) to use the Ontario Hydro (Canada) code ETMOD to assess doses from tritium both in the form of HT and HTO. The original studies for JET assumed tritium was in the form of HTO, as little information was available at that time on the conversion in the environment of HT to the more hazardous form of HTO.

Aspects of the ETMOD code have been validated against field studies on tritium releases carried out in Canada and France. The results of the analysis have confirmed the calculations carried out for the earlier submissions and have shown that releases in the form of HT (or T_2) rather than HTO result in significantly lower doses.

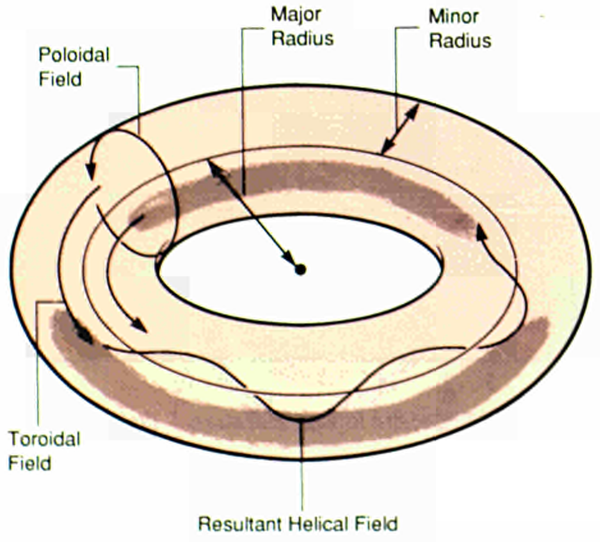


Results of JET Operation in 1991

Introduction

The overall objective of the JET Project is to study plasma in conditions and with dimensions close to those that would be needed in a fusion reactor. The central values of temperature, density and energy confinement time needed for a reactor operating with deuterium and tritium must be such that the product ($n_i\tau_E T_i$), exceeds the value $5\times10^{21}m^{-3}$skeV. Typical values for these parameters which must be attained simultaneously in a reactor, are given in Table.6.

With ohmic heating alone in JET, temperatures of 3keV and 4keV for the ions and electrons, respectively, densities of $4\times10^{19}m^{-3}$ and energy confinement times of 1s are the limits that have been achieved. These parameters were obtained simultaneously during one discharge and result in a fusion product of $1.2\times10^{20}m^{-3}$skeV. However, higher peak values of electron and ion temperature have been reached using additional radio frequency heating and neutral beam heating and combinations of these two methods. Even so, these substantial increases in temperature were associated with a reduction in the energy confinement time as the heating power was increased. Thus, gains in plasma temperature have been partly offset



Magnetic Field Configuration

The toroidal and poloidal magnetic fields combine to form helical magnetic field lines, which define a set of magnetic surfaces. As the strengths of the magnetic fields vary across the minor cross-section of the machine, the pitch of the field lines vary and usually decrease with increasing minor radius. The number of turns a field line must traverse around the major direction of the torus, before closing on itself, is denoted by the safety factor, q. Of special importance are the positions where q is numerically equal to the ratio of small integers, as these regions are specially sensitive to perturbations. Instabilities arising from these perturbations can result in enhanced energy losses.

In addition, the maximum plasma pressure, which can be maintained by a given magnetic field is dependent on the plasma current value. The effectiveness with which the magnetic field confines the plasma is given by β, which is defined as the ratio of plasma pressure to the magnetic field pressure.

JET can be operated with elongated plasma cross-section rather than circular. This enables larger plasma currents to be carried for given values of magnetic field, major radius and minor radius, as well as producing larger values of β.

Breakeven

This condition is reached when the power produced from fusion reactions is equal to that necessary for maintaining the required temperature and density in the plasma volume.

by degradation in energy confinement time. The fusion products obtained have not shown the full gains anticipated over conditions with ohmic heating only. However, a substantial increase in the values of the fusion product has been achieved, by operating in the magnetic limiter (X-point) configuration. During 1991, values of 9-10x10^{20}m^{-3}skeV were obtained using up to 16MW of additional heating.

Table.6: Reactor Parameters

Central Ion Density, n_i	2.5x10^{20}m^{-3}
Global Energy Confinement Time, τ_E	1-2s
Central Ion Temperature, T_i	10-20keV
Triple Fusion product, $(n_i\tau_E T_i)$	5x10^{21}m^{-3}skeV

Higher values of temperature, density and energy confinement have been obtained individually in separate experiments, but not simultaneously during one discharge. These include peak ion temperatures up to 30keV, energy confinement times up to 1.8s and central densities up to 4x10^{20}m^{-3}.

Ignition

Ignition of a mixture of deuterium and tritium would be reached if the power produced by the alpha-particles (20% of the total thermo-nuclear power) released from the fusion reactions is sufficient to maintain the temperature of the plasma.

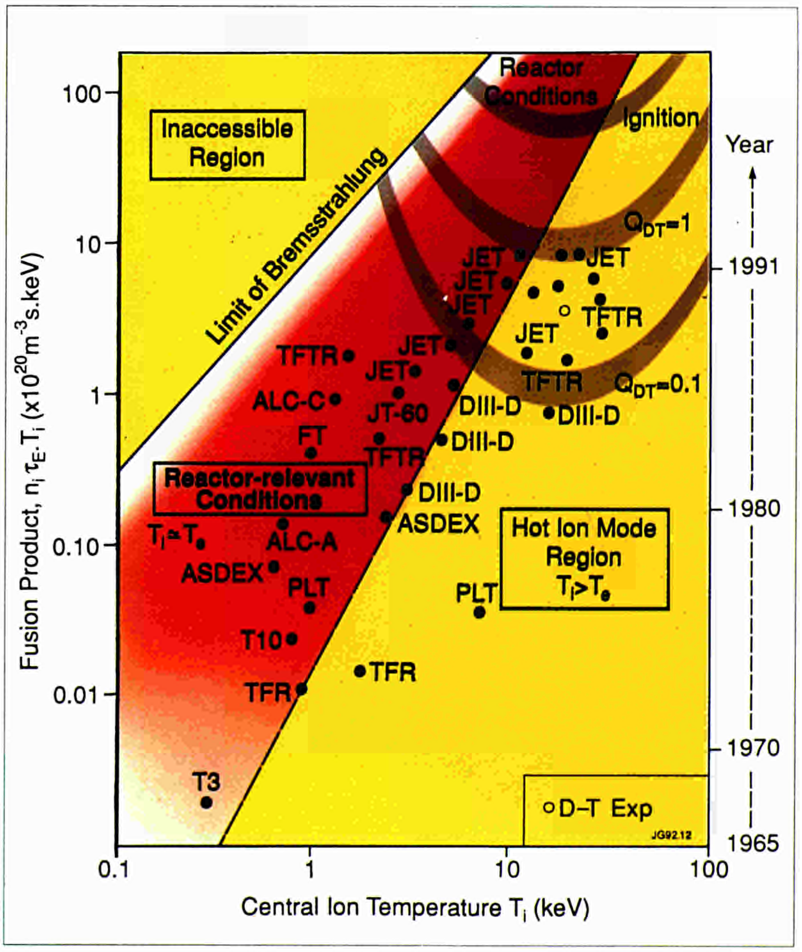


Fig.31: Fusion product ($n_D\tau_E T_i$) versus central ion temperature, $T_i(0)$, for a number of machines worldwide in the period 1965-1991.

The highest value of the fusion product attained so far in JET is equivalent to a value exceeding breakeven conditions. A factor of 5-8 increase would bring the conditions in JET to those required in a reactor. The increases in performance that have been achieved on JET and other tokamaks since 1965 are shown in Fig.31.

As the global energy confinement time scales favourably with plasma current in discharges with both magnetic and material limiters, the modifications carried out in JET, to increase the plasma current in both of these modes of operation, give confidence that significant alpha-particle heating will be observed when JET is operated in deuterium and tritium together.

Experimental Programme

The strategy of JET is to optimise the fusion product by building up a high density and high temperature plasma in the centre of the discharge, while still maintaining an acceptably high confinement time. These conditions should ensure that sufficient alpha-particles are produced with deuterium-tritium operation so that their confinement and subsequent heating of the plasma can be studied.

The original scientific programme of JET was divided into four phases as shown in Fig.32. The Ohmic Heating, Phase I, was completed in September 1984 and Phase II - Additional Heating Studies - started early in 1985. By December 1986, the first part, Phase IIA, had been completed. The machine then entered a planned shutdown for extensive modifications and enhancements before the second part of the Additional Heating Studies, Phase IIB, which started in June 1987. The objective of this phase, from mid-1987 until late-1988, was to explore the most promising regimes for energy

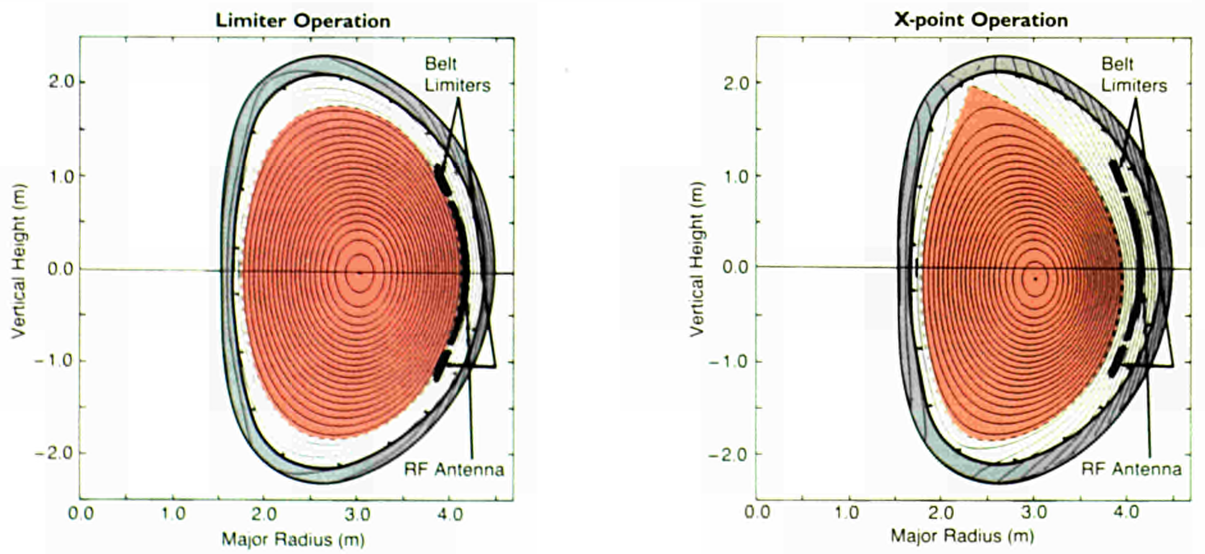
Operating Modes

Under normal operating conditions the magnetic surfaces are nested inside each other. The edge of the plasma is defined by the magnetic surface which intersects the limiter. The only magnetic field lines intersecting the walls of the chamber are those beyond the region bounded by the limiters as shown in the diagram on the left. This is termed material limiter operation.

The magnetic field configuration on JET can be modified so that one of the closed surfaces near the limiter is opened up so that it intersects with the vacuum vessel wall. In this configuration, the magnetic separatrix is moved to within the vacuum chamber.

This so called X-point configuration (or magnetic limiter) can be operated with the two nulls of the separatrix within the vacuum chamber (double null) or with only one inside (single null) as shown in the diagram on the right.

During X-point operation with additional heating, the plasma can behave, with respect to confinement, as though its edge were bounded by limiters. This is called the Low (L)-mode. Under certain circumstances, the plasma can be induced to behave in a different manner which produces better plasma confinement. This is termed the High (H)-mode of operation.



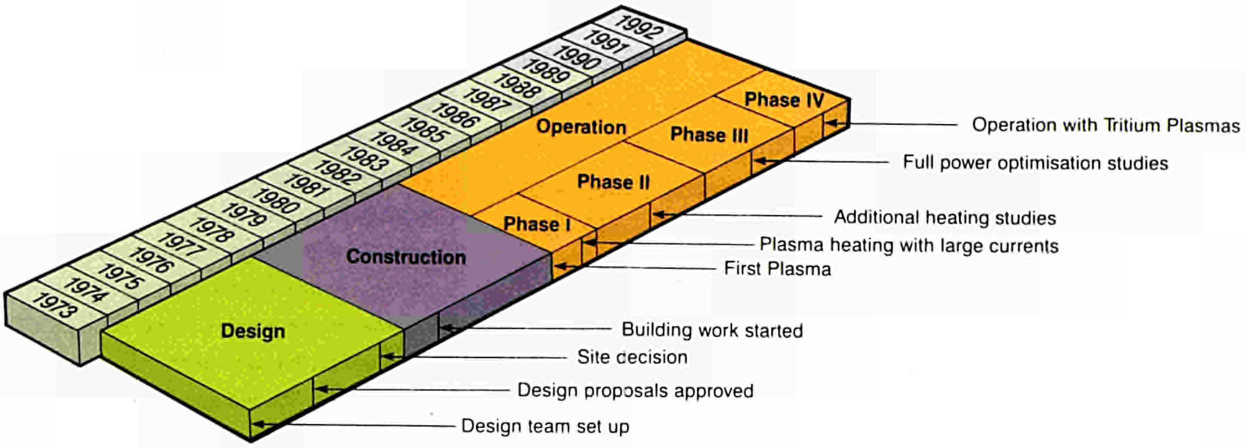


Fig.32: The original overall JET programme

confinement and high fusion yield and to optimise conditions with full additional heating in the plasma. Experiments were carried out with plasma currents up to 7MA in the material limiter mode and up to 5MA in the magnetic limiter (X-point) mode and with increased radio frequency heating power up to 18MW and neutral beam heating power exceeding 20MW at 80kV. The ultimate objective was to achieve full performance with all systems operating simultaneously. Phase III of the programme on Full Power Optimisation Studies started in 1989 and is due for completion in early 1992.

The 1991 experimental programme was executed by three Task Forces, with the programme objectives divided, as follows:

Task Force H: High Performance

(involving progression to full performance in material limiter and magnetic limiter configurations with currents up to 7MA, with high energy content and including progression to the highest fusion product, long pulse operation, steady state conditions and exploration of operation limits, etc.)

Task Force I: Impurity Transport and Exhaust

(Involving optimization of plasma purity and studying exhaust phenomena, and including studies of divertor physics and edge effects; impurity retention; erosion and redeposition; control of boundary instabilities, etc.)

Task Force P: Physics Issues

(involving studies of control and optimization of plasma profile effects (using LHCD, RF, NB and pellet injection, etc.) and optimization of heating effects, especially electron heating. This includes particle and energy transport studies in transient conditions; disruption and sawtooth stabilisation; high beta regimes; α-particle and fusion simulation studies; and issues related to Next Step devices.

Main Scientific Results

During 1991, the experimental programme concentrated on four main areas:

- introduction and exploitation of new facilities;
- optimization of plasma performance;
- advancing understanding in key certain areas of tokamak physics, such as: physics of the H-mode; energy transport and confinement; and transport of particles and impurities;
- establishing the basis for Pumped Divertor and Next Step physics (including a Preliminary Tritium Experiment).

The main new facilities introduced in JET were new X-Point Target Plates (Carbon - top; Beryllium - bottom) to improve overall power loading capability of X-point configurations; both NB systems brought up to 140keV deuterium operation; modifications to power supply systems (including poloidal shaping amplifier upgrade to 50kA; and introduction of reactive power compensation system); X-point gas introduction system installed; and preparation of new systems for the introduction and recovery of tritium for the preliminary tritium experiment (PTE).

Experiments aimed at improving understanding of tokamak physics have covered a wide range of topics including studies of thermal, particle and impurity transport, magnetic topology, and various aspects of H-mode physics. Experiments to improve performance have concentrated on high current limiter discharges, on optimizing fusion performance of X-point discharges and on the preliminary tritium experiment. Impurity control in JET, as for other long-pulse high power tokamaks and Next Step devices, is of fundamental importance and significant effort has been devoted to this area of study.

The scientific results in these areas are described in the following sections, under the headings: Impurities; Density Effects; Temperature Enhancement; and Energy Confinement Studies.

Impurities

Impurities present a major problem in tokamaks as they cause:

- large power losses from the plasma by radiation;
- dilution of the number of effective ions in the plasma available for productive fusion reactions, with a corresponding fall in thermonuclear yield;
- reductions in the density limit.

Impurities

Impurities released from interactions between the plasma and material surfaces can have major effects on plasma behaviour by causing:

(a) increased radiation losses;

(b) dilution of the number of ions available in the plasma between which fusion reactions can occur.

A measure of the overall impurity level is given by Z_{eff} which is defined as the ***average*** *charge carried by the nuclei in the plasma. A pure hydrogen plasma would have Z_{eff}=1 and any impurities in the plasma would cause this value to be increased. In JET, Z_{eff} is generally in the range from 1.2-3.*

Major energy losses can result from two radiation processes:

- *Bremsstrahlung Radiation - radiation is emitted when electrons are decelerated in the electric field of an ion. The amount of radiation emitted increases with Z_{eff}. Bremsstrahlung radiation imposes a fundamental limit to the minimum plasma temperature that must be attained in a fusion reactor;*
- *Line Radiation - heavy impurities will not be fully ionised even in the centre of the plasma and energy can therefore be lost through line radiation.*

Considerable effort is made to keep the level of impurities in the JET plasma to a minimum. The vacuum vessel is baked at 300°C to remove gas particles trapped on the vessel walls which might be released by plasma bombardment.

Interactions between the plasma and vacuum vessel walls would result in the release of heavy metal impurities. To reduce this possibility, the edge of the plasma is defined by upper and lower belt limiters. These are cooled structures circling the outboard torus wall with carbon or beryllium tiles attached. Carbon and beryllium have a relatively low electric charge on the nucleus.

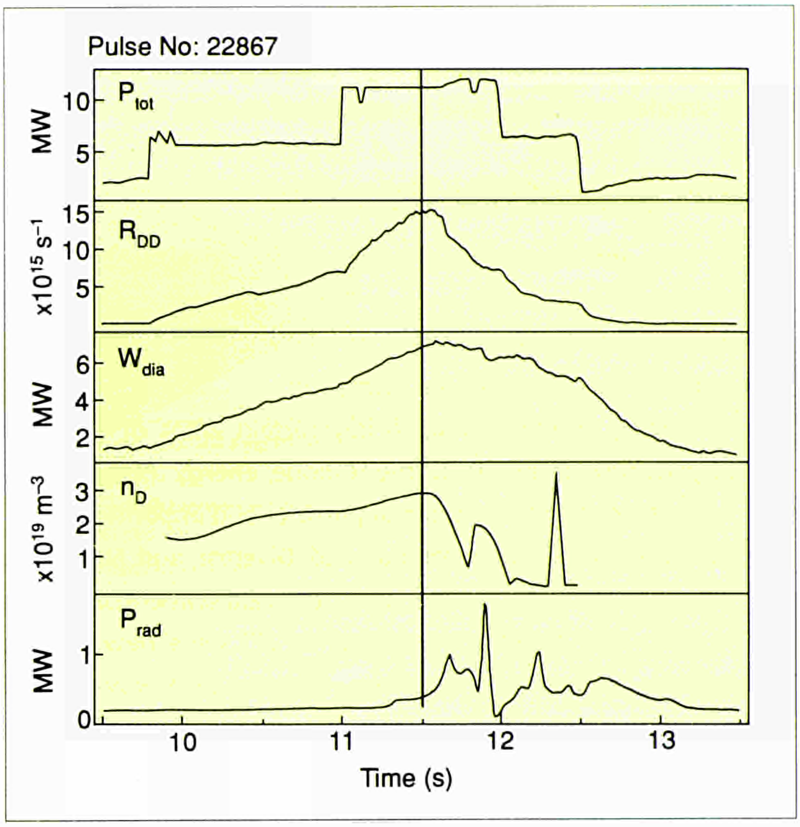


Fig.33: Total input power (P_t), fusion rate (R_{DD}), plasma energy (W), deuteron density (n_D) and radiated power (P_{rad}) for a plasma displaying characteristics of the impurity bloom.

The parameter which provides a measure of the impurity content of a plasma is the effective ion charge, Z_{eff}, which is the average charge carried by an ion in the plasma.

A 'catastrophic' event which limits the performance of high temperature plasmas is the process known as an 'impurity bloom'. This can be characterised as the result of regions of the target tiles overheating and the concomitant impurity influx overwhelming the plasma. The event is signalled by an enhancement in impurity light emission and radiated power and an increase in the effective charge Z_{eff}. An example of this process is shown in Fig.33. A delay is discernable between the onset of the 'bloom', as seen in the radiated power, and the fall-off in deuterium concentration, neutron rate and plasma energy content. Also, the roll over of all these quantities is quite gentle. Behaviour of this kind might be anticipated, since the impurities are produced at the boundary and these must be transported to the plasma core to effect performance.

Investigation of the performance of divertor plasma utilizing the newly installed toroidal X-point target tiles was a central activity throughout 1991. The choice of beryllium tiles for the lower target and carbon fibre composite (CFC) tiles for the upper target permitted a careful study of the relative merits of the two materials in this role. In addition, in the course of these experiments, two designs of

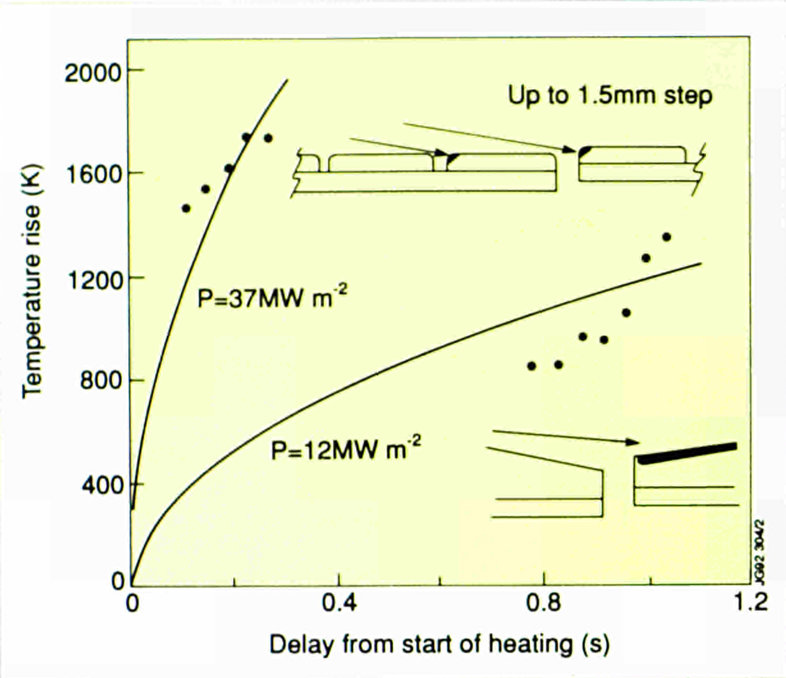


Fig.34: Temperature rise as a function of delay from start of heating for the two types of target plate tiles.

CFC tile were examined and this allowed the limitations arising from heating of tile edges to be studied. The principal aims of the target assessment were: to establish the power handling capabilities of different target materials and of different target designs; to investigate the causes of limitations in the power handling capabilities; and to study the influence of different target materials on bulk plasma performance. Two designs for the carbon plates were compared. The first set of carbon plates had relatively sharp exposed edges (radius about 8mm). Assembly tolerances, although tightly controlled, resulted in steps up to 1.5mm between adjacent tiles. The temporal evolution and spatial distribution of the surface temperature was measured. Hot spots with temperatures in the range 1500 to 1880 K were seen on exposed plate edges within 200ms of the start of the heating (Fig.34). These edges of the second set of carbon plates were sloped at a shallow angle (about 4°) which was calculated to protect a step of 1.5 mm at a field angle of 1°. Measurements confirmed that edge heating had been eliminated and that the heating was distributed over a larger area on the sloping part of the tile. The surface temperature rise reached the range 800 to 1200 K in about 800ms. Temperature rises on both types of plate were consistent with calculations. The effective heated area of the sloping plates was estimated to be about 5 times larger than that of the sharp-edged plates.

High performance, hot-ion H-mode discharges with both sets of plates were terminated by carbon blooms, but there were significant differences between the threshold energies and the nature of the

Disruptions

There is a maximum value of density which can be contained with a given plasma current. If this value is exceeded a disruption occurs when the plasma confinement is suddenly destroyed and the plasma current falls to zero in a short period of time. Under these conditions high mechanical and thermal stresses are produced on the machine structure. Disruptions are thought to be caused by instabilities mostly developing on the magnetic surface where q=2.

blooms. There was very little difference between the energy at which carbon blooms occurred for the plates with exposed edges compared to the poloidal bands of plates previously used at the divertor in JET. In both cases the neutron production rate remained high (indicating that the core plasma parameters had not degraded) for a few tenths of a second after the influx of carbon started to increase at the plasma edge. In contrast, the influx of carbon from the plates with sloping edges caused an almost immediate degradation of the core plasma. The threshold energy for a bloom was higher on the plates with sloping edges, but only by a factor of about two, compared to the sharp-edged plates although the effective heated area was estimated to have been improved by a much larger factor (about 5).

Beryllium has been used in JET since 1989 as a limiter and divertor plate material, for the screens on ICRF antennae and as an evaporated layer on the carbon surfaces. The introduction of beryllium has effectively eliminated oxygen (by gettering) and nickel (previously used for the antennae screens) from JET plasma. There has been a marked improvement in deuteron dilution. Beryllium reduces radiated power losses and allows H-modes with ICRF heating. Beryllium pumps hydrogen and helium providing better density control. High density discharges with beryllium usually terminate in a MARFE rather than a disruption.

Beryllium plates with sloping edges were used at the lower divertor. At modest heating powers (< 10MW), there appeared to be little difference in the confinement properties of medium density H-modes compared to discharges on the carbon plates, but the ion temperature was higher with beryllium due to reduced radiation cooling. At the higher heating powers (>10MW) and lower densities required for the hot-ion H-modes which give the highest fusion performance in JET, the beryllium plates were vulnerable to melting which resulted in the rapid deterioration of the discharge due to a beryllium bloom. The melting occurred progressively earlier on plates that had been melted by previous discharges. The highest fusion reaction rate obtained with beryllium was about a half of that obtained with carbon. Beryllium plates could withstand higher heating powers when combined with strong gas puffing to increase the radiated power and cool the plasma edge. Puffing was effective at the X-point and at the mid-plane.

An important observation reported during 1990 was that test impurities introduced into the plasma showed evidence of better confinement during H-modes. The temporal evolution of nickel impurities in the L- and H- phases of two similar discharges showed that nickel impurities have considerably longer confinement times in the H-phase that in the L-mode of confinement. This suggested that the H-mode might be a disadvantage in a reactor, where impurities and helium ash could be retained leading to increased fuel dilution in the plasma centre. Further experiments with a greater range of impurities has shown that there was evidence of accumulation of metallic impurities (nickel) injected by laser ablation, but there was no indication of accumulation of low Z-impurities (such as carbon or beryllium) which are the dominant intrinsic impurities in JET.

Density Effects

Disruptions and Density Limits

In most tokamaks, disruptions can occur under certain circumstances. Disruptions are dramatic events in which plasma confinement is suddenly lost, followed by a complete loss of current in the plasma. Disruptions pose a major problem for tokamak operation as they can impose a limit on the density range in which stable plasmas can be achieved and their occurrence leads to large mechanical stresses and to intense heat loads on the vacuum vessel.

Experiments in JET have improved understanding of the conditions under which disruptions occur and of the evolution of the plasma towards a major disruption. Recently, emphasis has been given to the development of techniques for avoidance of disruptions, or for the minimisation of their consequences. A substantial advance was made in this respect through improvements to the Plasma Fault Protection System (PFPS). Early detection of a helical non-rotating instability (or 'locked' mode), which is invariably a precursor to major disruptions, has enabled measures to be taken (reduction of plasma current, elongation and of heating power) to minimize the impact of the resultant disruption on the torus.

This has been of particular value in overcoming the problems resulting from vertical instabilities which follow major disruptions. Analysis has shown that these arise from loss of vertical stability in elongated plasmas at the energy quench and that their consequences are aggravated by the slow post-disruptive current decay observed since the introduction of beryllium into JET. Such instabilities lead to substantial forces on the vacuum vessel (several hundred tonnes) and have resulted in damage to internal components. By using detection of the occurrence of a 'locked' mode to trigger a rapid ramp down in driving current (the dominant source of the destabilizing force) to zero, it has been possible to postpone the loss of vertical stability until much later in the current decay with a consequent reduction in forces by an

Density Control

Increasing the density can be achieved by introducing additional gas into the vacuum vessel, by the injection of energetic neutral atoms (neutral beam heating) and by solid pellet injection.

Increasing the input power to the plasma through additional heating raises the electron density limit. However, problems can occur when this heating power is switched off, if the electron density is too high. To overcome this problem, the plasma is moved, prior to the switch-off point, so that it bears on the carbon tiles covering the inner wall. The tiles have been found, to provide a pumping mechanism for removing particles so that the density can be reduced below the critical limit.

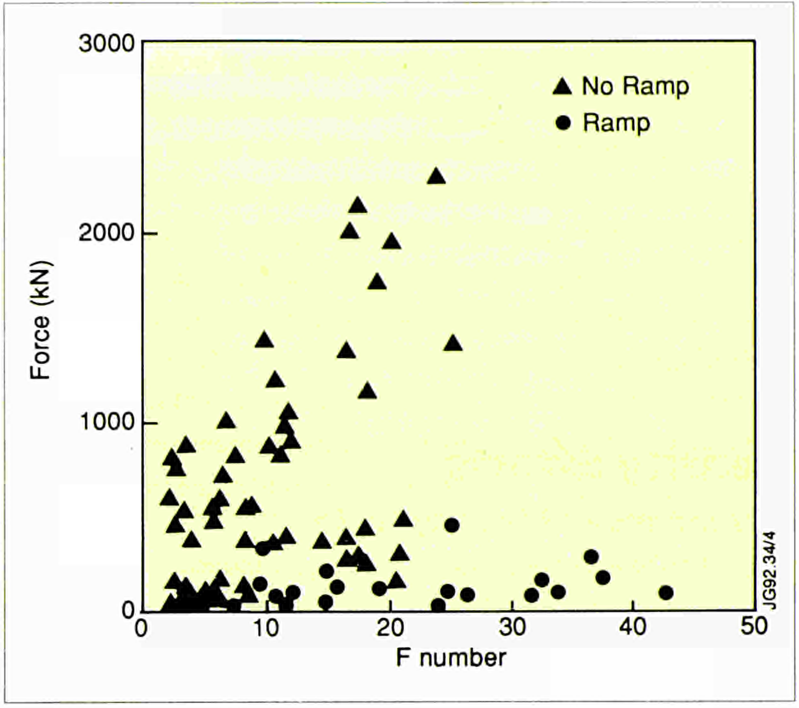


Fig.35: Vertical forces on the vacuum vessel during disruptions versus a measure of the vertical destabilisation force on the plasma (F number). The triangles show the normal case whereas the circles show the effect of ramp down action of the current following detection of the disruption precursor.

order of magnitude. Figure 35 shows the reduction in vertical forces on the vessel during disruptions when the current is ramped down rapidly.

A further disruption instability investigated is that caused by the growth of large amplitude n=1 'locked' modes as the internal separatrix (X-point) is formed. This mode becomes more persistent at low safety factor, q, and has a low density threshold. Detailed experiments have confirmed these observations and have identified the most likely cause of the mode as external error fields arising from details of construction of the poloidal field coils. The principal source of these error fields appears to be the vertical field coils and the threshold density depends on the coil configuration, the helicity of the internal magnetic field (the detrimental arrangement occurs when the helicity of the tokamak magnetic field matches that of the calculated error field) and the edge safety factor. The occurrence of the mode does not depend on the existence of an X-point, nor even of a plasma more elongated than the natural elongation in JET (~1.4). In the worst cases, it is not possible to establish plasmas with the safety factor q < 3, since the low density limit due to these 'error field' modes overlaps with the high density limit.

Pellet Fuelling

For a reactor, it is important to obtain an optimum central density, n(0), while minimizing the edge density in contact with cool material

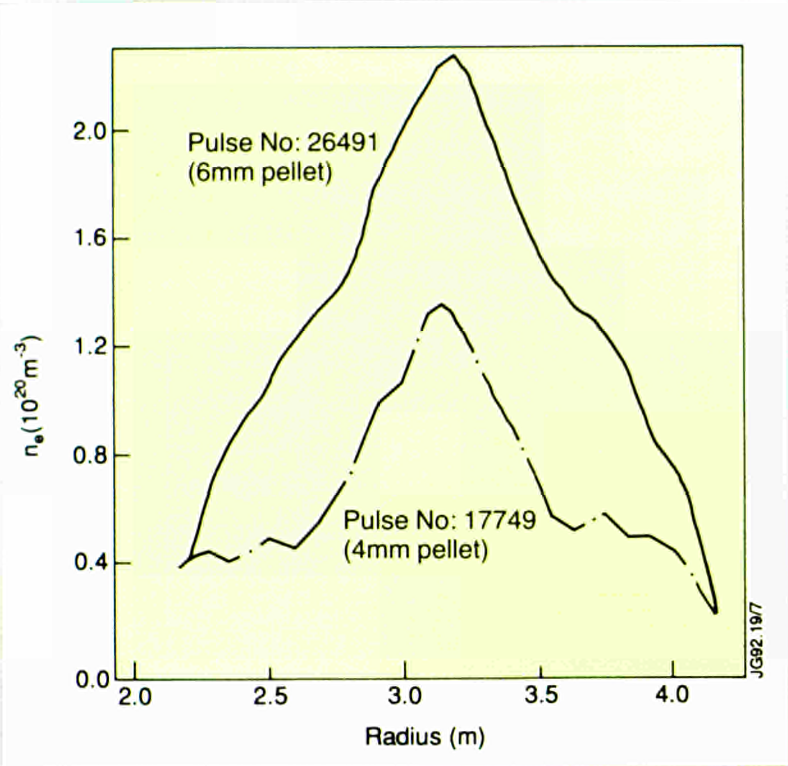


Fig.36: Density peaked over the full radial plasma profile with a 6mm pellet, rather than just the central core, as was found in the 4mm pellet case.

surfaces. This optimizes the number of useful nuclear fusion reactions whilst providing good insulation of the hot plasma from it surroundings. This means that a large density peaking factor, n(0)/<n>, (where <n> is the average density) is desirable for optimum performance. An effective method of achieving this aim it to inject solid pellets of hydrogen or deuterium into the plasma centre.

The injection of multiple 2.7, 4mm and 6mm diameter solid deuterium pellets has been undertaken into JET plasmas under various conditions including material limiter and magnetic limiter (X-point) discharges, with plasma currents up to 6MA in ohmic, neutral beam and RF heating situations. This has been undertaken since 1988 as a collaborative effort between JET and a US team, under the umbrella of the EURATOM-USDoE (US Department of Energy) Fusion Agreement on Pellet Injection. A jointly built three-barrel, repetitive multi-injector has been used to inject pellets at a maximum frequency of several per second with nominal speeds up to 1500ms^{-1}. A record central density of $4\times10^{20}m^{-3}$ was achieved by strongly peaking the density profile using a sequence of 4mm solid deuterium pellets injected at intervals throughout the current rise phase of a X-point discharge.

To improve the overall plasma performance in JET, two regimes of enhanced performance have been combined. This takes advantage of the good global confinement properties of the H-mode together with peaked profiles produced by pellet fuelling and central heating of the pellet-enhanced performance (PEP) mode. PEP modes generated by the injection of 6mm pellets into 4MA discharges have been studied. Contrary to results in 1990, the density peaked over the full radial plasma profile, rather than just in the central core as in the 4mm pellet case. This is displayed in Fig.36. Since the 6mm pellet particle content was three times higher than that of the 4mm pellet, these discharges were dense, and, therefore, relatively cold. The density and the density gradients decayed on a characteristic timescale of about 1 second, as was the case with the earlier 4mm shots. Therefore, such discharges require higher power than the available 10-15MW to heat to high thermonuclear performance in a time scale of 1s, (typically T_i does not exceed 5keV). These discharges are being further investigated.

Temperature Enhancements

Sawtooth Oscillations

In most tokamak discharges including JET, the central temperature and density is generally modulated by sawtooth-like oscillations. This is due to the periodic occurrence of magnetohydrodynamic (MHD) perturbations associated with the plasma surface whose safety factor, q, has a value of unity. Heating in the plasma centre leads to a gradual rise in the central temperature, which is terminated suddenly by the rapid growth of the MHD instability. The principal effect is to flatten the temperature and density profiles across the region inside the so-called mixing radius, which can range from about one to two-thirds of the plasma radius. As a consequence, high energy particles, produced by auxiliary heating or by fusion reactions, can be expelled from the plasma centre to larger radii where they may be lost rapidly to the plasma periphery.

The consequences of the sawtooth instability are detrimental in two respects. This instability flattens the central plasma temperature and density, and it can expel energetic particles (injected by neutral beams, accelerated by ICRF fields, or produced in fusion reactions) from the plasma core. Therefore, the repetitive occurrence of this instability can significantly limit the fusion power produced by the plasma. However, a number of techniques have been developed for the suppression of these instabilities and the resultant improvements in performance explored in some detail. In particular, it is planned to control the current profile directly, and hence to eliminate the q=1 surface. This can be achieved by the injection of lower hybrid (LH), radio-frequency (RF) waves or neutral beams in the plasma. In addition, a spontaneous stabilization process was discovered during additional heating experiments in JET, which has resulted in the production of long sawtooth-free periods (called 'monsters').

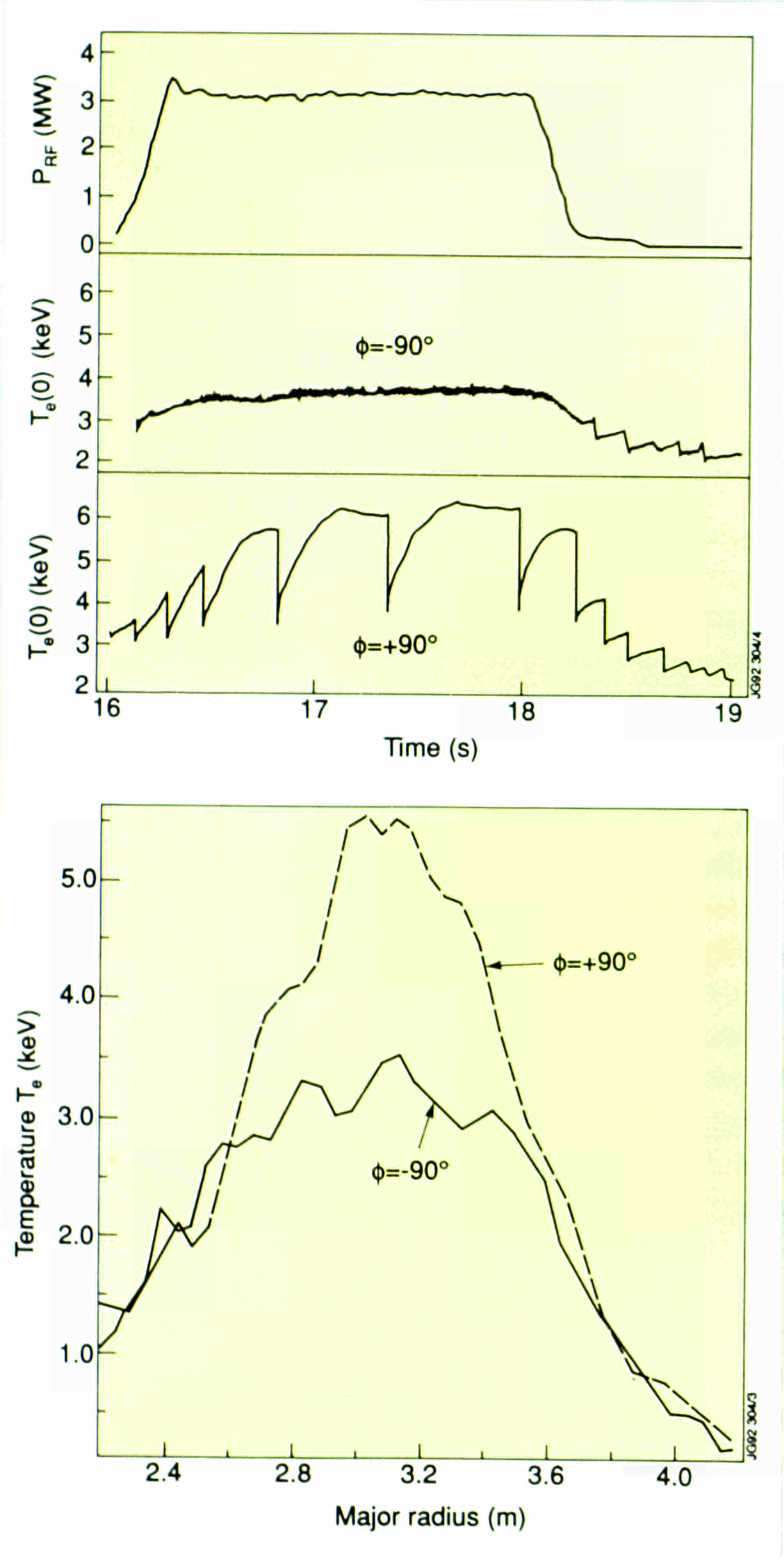


Fig.37: (a) Effect of power phasing ϕ =± 90° on sawtooth oscillations; (b) Corresponding effect on electron temperature profile.

Recent technical improvements to the ICRF heating system have enabled the two striplines in each antenna to carry currents with equal amplitude but arbitrary phasing. Travelling fast waves can be launched to perform current drive experiments. Such waves are absorbed at a minority ion fundamental resonance placed tangen-

Sawteeth

Perturbations on the q=1 magnetic surface can result in the formation of large fluctuations in the central temperature and density. These fluctuations have been termed 'sawteeth'. They are also associated with the expulsion of energetic ions from the central region of the plasma. Understanding this process is important as the alpha-particles produced from deuterium-tritium fusion reactions might be lost before they can produce any effective heating of the plasma.

Current Profile Control

The highest current density exists at the centre of the plasma, as this is the hottest region and the electrical resistivity decreases as the temperature increases. Without sawteeth, which occur on the q=1 surface, this high current density region would be squeezed or pinched inwards. Selective heating outside the central region would remove the q=1 surface from the plasma and so avoid the onset of the sawteeth. Another way is to decouple the plasma current and temperature profiles. On JET, it is intended that an electric current, additional to that generated by transformer action, should be produced by neutral beams and by radio-frequency power at 3.7GHz (called Lower Hybrid Current Drive (LHCD)).

tial to the q=1 surface and affect sawtooth stability through minority ion current drive modifying the current density and hence the magnetic shear. Such a scheme drives current in opposite directions on either side of the resonance layer to form a highly localised current density perturbation which can increase or decrease the shear locally at the q=1 surface, depending on the direction of propagation of the travelling waves. Figure 37(a) shows the effect of phasing the ICRF power with the resonance on the high field side of the magnetic axis. Applying RF power with +90° phase produced monster sawteeth in conditions where dipole phasing creates only long sawteeth. The electron temperature (T_e) profile showed central peaking characteristic of monster sawteeth despite the off-axis power deposition (Fig. 37(b)). In contrast, reversing the toroidal direction of the wave by using -90° phasing produced a dramatic destabilisation and created sawteeth of very short period (40ms). In this case, the T_e profile was flattened inside the inversion radius. By controlling the sawtooth period, and thus the rate of ejection of fast particles from the plasma centre, this method offers potential for burn control in a reactor.

Radio Frequency Heating

The ion cyclotron resonance frequency (ICRF) heating system is used for highly localized heating of the JET plasma. The wide frequency band (23-57MHz) allows variation in the radial heating position at which the ion cyclotron resonance frequency matches the local magnetic field position (which varies across the radius of the machine). A minority ion species (H or ^{3}He(1-10%), at present, and D in the future D-T phase) is injected into the plasma for this purpose, and absorbs the RF power at the local resonance position. These high energy localized ions collide with the main plasma electrons and ions, transferring energy to them and causing a rise in the local electron temperature (T_e) and ion temperature (T_i) of the main plasma. A maximum ICRF power of about 22MW for 1.75 seconds has been coupled to the plasma.

Significantly improved performance has been obtained in the Pellet Enhanced Phase (PEP) of operation, in which enhanced central confinement is obtained with deep pellet injection to produce high densities. ICRF heating has been used in the PEP regime to re-heat a densified core ($n(0) \approx 1.2 \times 10^{20} m^{-3}$) with a power deposition profile peaked on axis. It has been possible to increase the central ion temperature by using two frequencies (40 z and 36 MHz) accelerating two different minorities ions (H and ^{3}He). This method reduces the minority tail temperature driven by ICRF power so that the proportion of power heating the ions directly is increased to ~60%. Peak ion temperatures ranging from 16 to 18 keV and peak electron

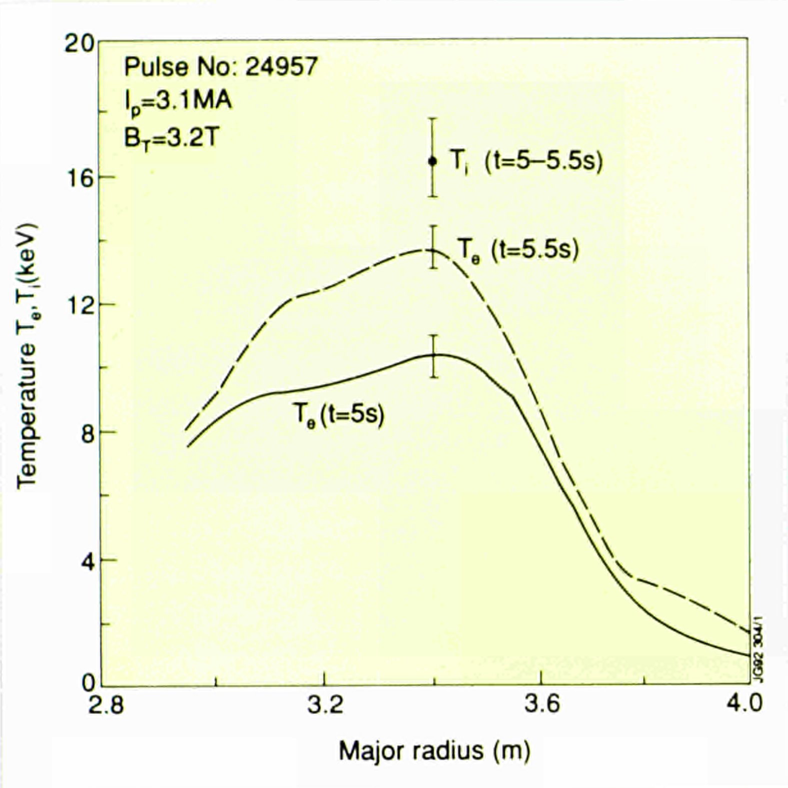


Fig.38: Electron temperature profile and central ion temperature during a PEP discharge heated with ICRF power suing two minority frequencies for H and ^{3}He.

temperature from 12 to 14keV have been obtained with this method. The electron temperatures profiles are shown in Fig.38.

The Lower Hybrid Current Drive system (LHCD) can drive a significant fraction of the plasma current by direct acceleration of the plasma electrons. This is the main tool for controlling the plasma current profile. It can be used to stabilize sawtooth oscillations, thereby increasing the central electron temperature. A prototype system consisting of two launching units fed by a total klystron power of 4MW has been tested during 1991 and up to 2.5MW has been coupled to the plasma. Full sustainment of the plasma current up to 1.5MA was achieved with a combination of LHCD (up to 2.3MW), and ICRF power (3 to 5MW). Full current drive was observed when the LH induced fast electrons have a rather peaked profile which was achieved at high toroidal field, B_T>3T, and low density, $n<3\times10^{19}m^{-3}$. An example of a discharge where the current of 1.5MA is sustained is shown in Fig.39. The current drive efficiency reached $nRI/P=0.4\times10^{20}m^{-2}A/W$, which is the highest demonstrated in a tokamak device.

Neutral Beam Injection

The two neutral beam injectors on the JET machine inject high energy neutral beams into the plasma tangential to the torus from diametrically opposite positions at Octant No: 4 and Octant No: 8.

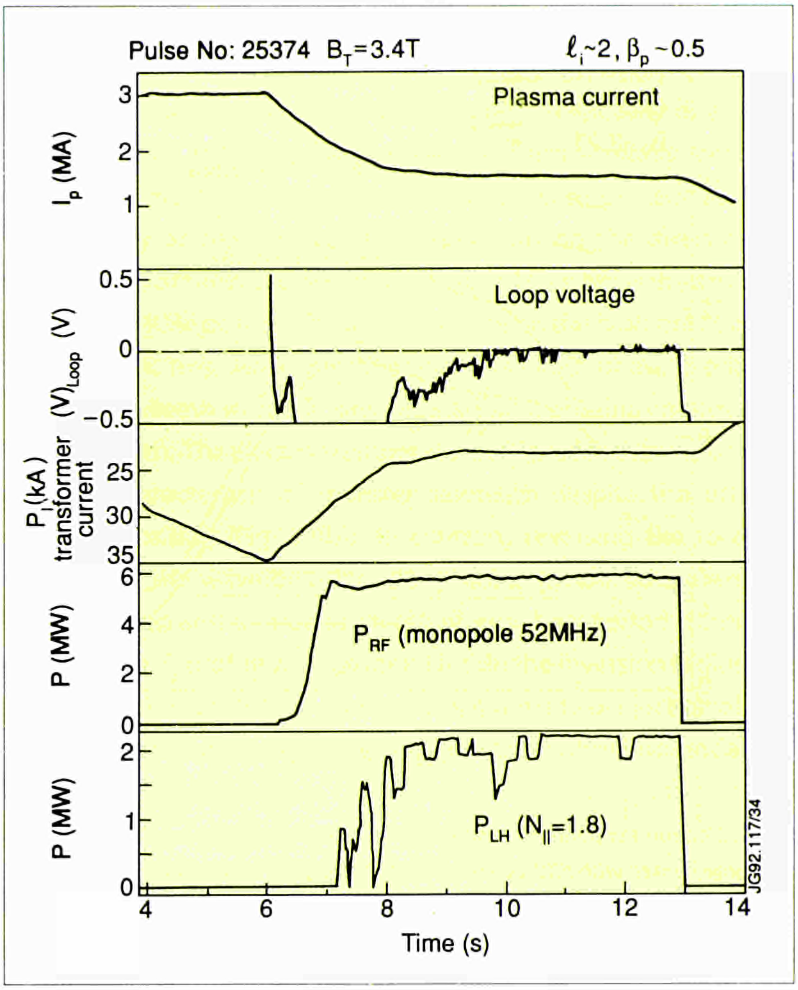


Fig.39: Full current drive at I_p=1.5MA obtained with combined LHCD and ICRF power. The current is fully sustained when the loop voltage falls to zero.

Originally, both injectors operated at 80kV energy and in 1988 deposited up to 21.6MW power in the plasma. During 1989/90, the Octant No: 4 system was converted to 140kV operation with deuterium beams and during 1991, the Octant No: 8 system was similarly upgraded. This provides greater penetration of the neutral beams into denser plasmas and deposits more power near the plasma centre. However, when operating both systems at 140 keV, the total power was reduced to ~16MW of beam power.

If densities are not too high, the neutral beams penetrate to the centre of the plasma and deposit their energy there. In these conditions, the beam energy is transferred mainly to the plasma ions, causing large increases in ion temperature. This is the so-called 'hot-ion mode' of operation. In this mode, record ion temperatures have been achieved up to 22keV with material limiter plasmas and up to 30keV in magnetic limiter plasmas, using neutral beam heating. In addition, during 1991, both injectors were adapted to produce He^3 beams at up to 155kV. The advantages with helium beams are that they avoid the production of beam-plasma neutrons which complicate the measurement of thermonuclear neutron production and result in activation of the vessel. A total of 13MW of ^{3}He beam power was produced.

The major focus of the work carried out with neutral beam heating system during 1991 was related to the Preliminary Tritium Experiment (PTE). Neutral beam injection was used, not only to heat the plasma using fourteen beams of deuterium, but also to inject the tritium fuel using two tritium beams. Neutral beam injection is an effective way of introducing tritium into the type of discharge selected for the deuterium-tritium experiment. It ensures that tritium reaches the hot, dense centre of the discharge where the reactivity is highest and minimises the amount of tritium injected into the torus. This is the first time that a neutral beam system had been used to inject energetic tritium neutrals at high power and long pulse duration into a fusion plasma and represents an important advance in this technology. Further, it demonstrated the importance and flexibility of the JET NB system, which had already been used to inject multi-magawatt beams of H, ^{2}D, ^{3}He and ^{4}He. Details of this successful experiment are described in the section on the Preliminary Tritium Experiment.

Energy Confinement Studies

The global energy confinement time in all JET plasma configurations, is defined by the relationship:

$$\tau_E = W_k/(P_t - dW_k/dt),$$

where W_k is the kinetic energy and P_t is total input power to the plasma without subtracting radiation losses. The values of τ_E reported are such that $(dW_k/dt) << P_t$ and so are quasi-stationary.

Material Limiter Configuration

Energy confinement times up to 1.8 seconds have been attained in JET ohmically heated discharges. However, the temperatures achieved are too low to achieve fusion reactor conditions, so that it is important to provide additional heating.

Energy confinement times on JET fall with increasing heating power independent of the type of heating, whether neutral beam injection, radio frequency heating or a combination of the two methods, with both carbon and beryllium limiters. However, with a beryllium first-wall improved density control is achieved due to higher wall pumping and the impurity content (and plasma radiation) is reduced. These factors allow higher plasma densities and result in improved fuel concentrations (n_D/n_e), which permit improved fusion performance.

The decrease of energy confinement time with increasing heating power in material limiter cases is shown in Fig.40. The energy

Energy Confinement

Energy confinement in tokamaks when the plasma is bounded by a material limiter generally degrades as the input power to the plasma increases. The result is that the energy confinement time, τ_E, falls approximately as the square root of the input power. This regime is said to exhibit L(low)-mode confinement. In plasmas with a magnetic limiter (that is with an internal magnetic separatrix or X-point), a transition can occur above a certain threshold input power to a regime in which the energy confinement time is increased by a factor of two or more greater than in the L-mode situation. This has been called H(high)-mode confinement. However, a similar degradation with input power is observed.

In addition to the improved energy confinement time, enhanced particle confinement is observed and the temperature and density close to the separatrix can increase substantially, resulting in the formation of plasma profiles with an edge 'pedestal'. The precise conditions for the transition into the H-mode vary with plasma parameters. For example, the threshold power for the transition increases at least linearly with the toroidal magnetic field. In recent years, the H-mode transition has also been observed in plasmas with a material limiter, although the power threshold is usually significantly higher than in magnetic limiter (X-point) plasmas.

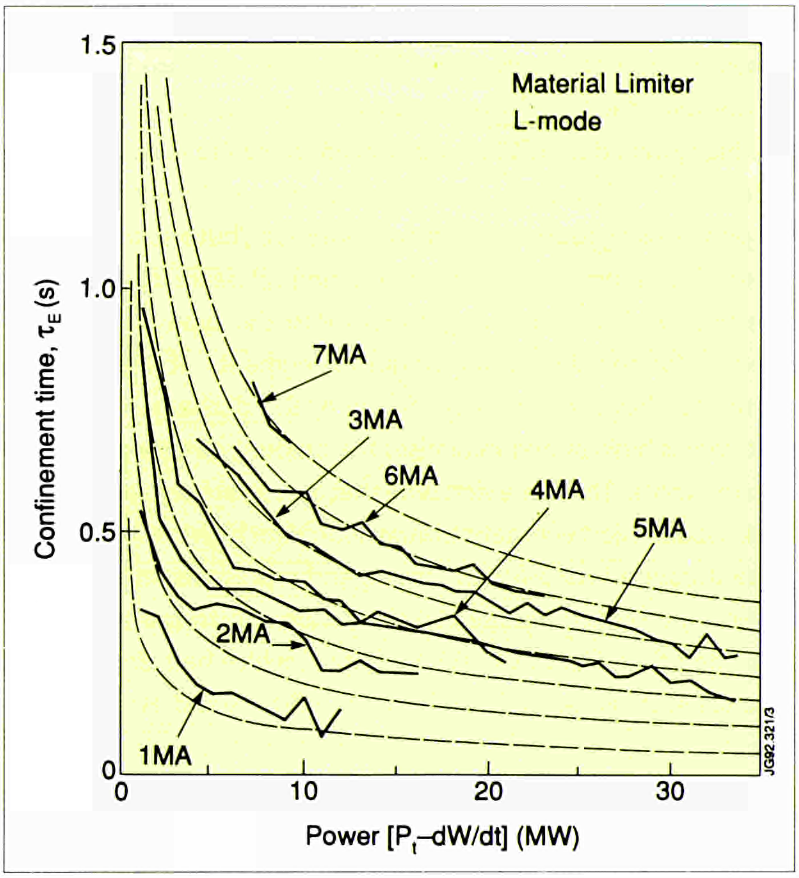


Fig.40: Confinement time as a function of input power for material limiter conditions, for different plasma currents.

confinement time shows only a weak dependence on density, but improves favourably with increasing plasma current. The global energy confinement time results have been fitted to results from thirteen tokamaks worldwide (including JET) by a simple power law relationship. The ITER and L-mode scaling law still remains valid and is given by

$$\tau_E(\text{ITER89-P})=0.048\ A^{0.5}I_p^{0.85}T^{1.2}a^{0.3}\kappa^{0.5}(n/10)^{0.1}B^{0.2}P^{-0.5}$$

where A is the atomic number, I_p(MA) is the plasma current, R(m) is the major radius a(m) is the minor radius, κ is the elongation, B(T) is the magnetic field, n($10^{19}m^{-3}$) is the density, and P(MW) is the total input power.

The dependence of global confinement properties upon global plasma parameters has been well established empirically. However, the dependence of local confinement, upon local values of plasma parameters is not well known. A series of experiments were conducted in which the plasma current was ramped up or down from one flat-top value to another. The current ramp took place at constant power and the density, magnetic field, plasma shape were also held constant. During the ramp and for some period after, the internal inductance, the safety factor and the shear profile were transiently decorrelated until a steady state was reached by resistive diffusion. The data from such experi-

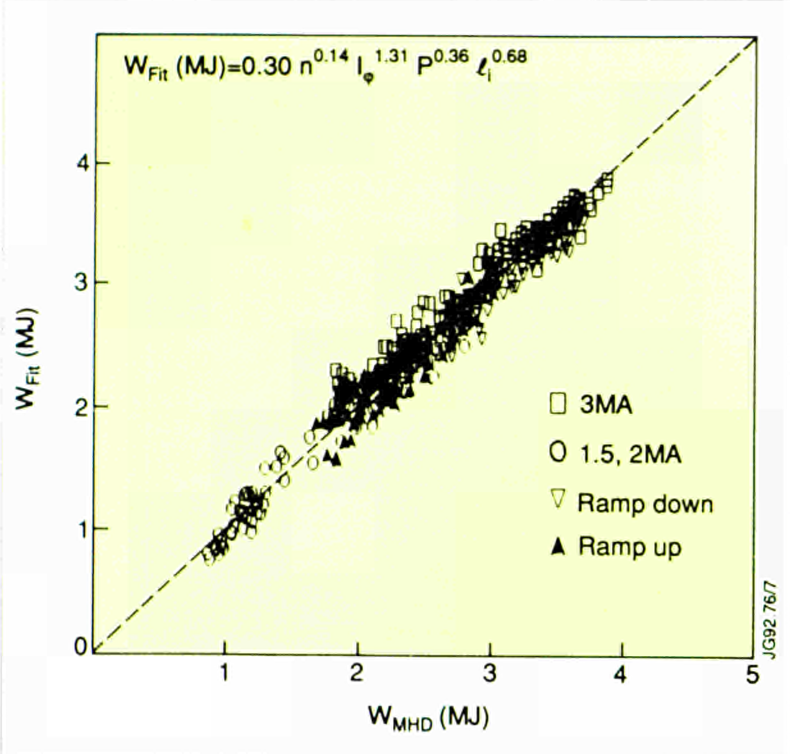


Fig.41: Fit to stored energy $W_{fit} = 0.3\ n^{0.14} I_p^{1.31} P^{0.36} \ell_i^{0.68}$ versus the measured stored energy (W) for the current ramp pulses. I_p is the toroidal current, n the average density, P the total power and ℓ_i the inductance

ments shows that global confinement depended on both the total plasma current I_p and its distribution, the internal plasma inductance ℓ_i. The scaling was found to fit the data, as shown in Fig.41, which is of the form $W = 0.30\ n^{0.14} I^{1.31} P^{0.36} \ell_i^{0.68}$ where W is the plasma energy, n($10^{19}m^{-3}$) is the density, P(MW) is the power and ℓ_i is the plasma inductance.

Most scaling laws of energy confinement predict that the energy confinement time, τ_E, scales as $A^{0.5}$ where A is the atomic mass of the ion species. However, such a dependence is difficult to justify on theoretical grounds. A series of well diagnosed discharges has been performed in JET with hydrogen (H), deuterium (D) and the helium isotope, He^3, to study this problem. Care has been taken to avoid mixtures of isotopes and to obtain discharges with the same plasma configuration, current, magnetic field, and similar density, power deposition profiles and impurity content. These aims have been achieved in ohmic and neutral beam heated discharges, and a complete set of reference discharges has been obtained for L-mode discharges in limiter configuration. Results of the analysis indicate that, in JET L-mode discharges, a dependence of the energy confinement time τ_E on the atomic mass A as strong as $A^{0.5}$ must be ruled out. No difference in energy confinement has been observed between D and He^3, while for H and D cases, any difference is less than 20%.

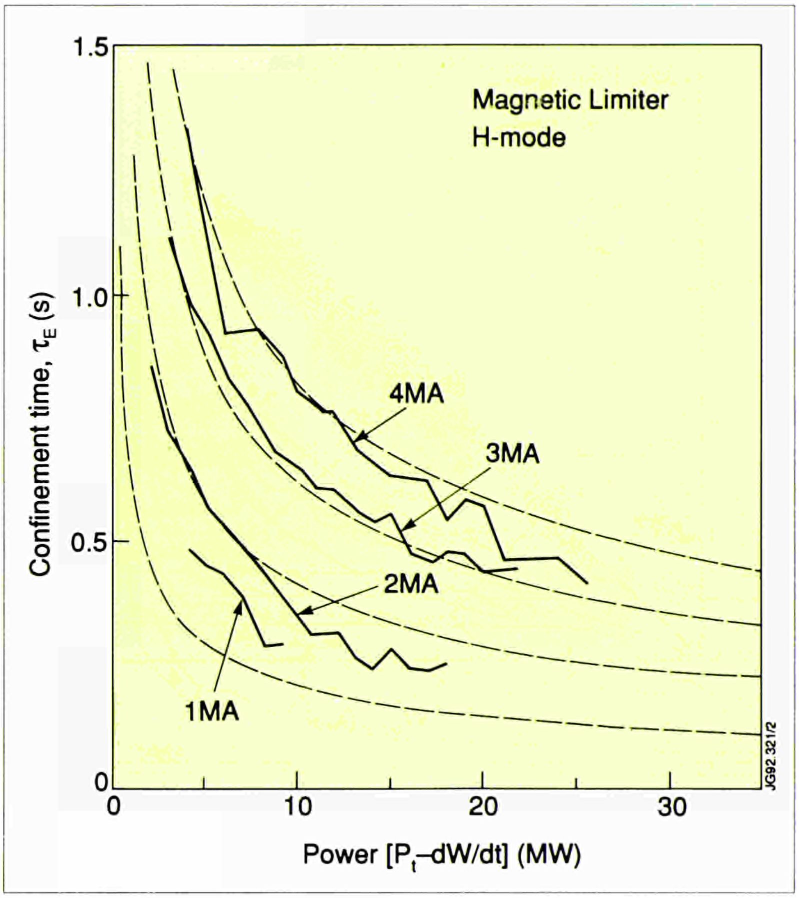


Fig.42: Confinement time as a function of input power for magnetic limiter conditions, for different plasma currents.

Magnetic Limiter Configuration

In the magnetic separatrix (X-point) configuration, the plasma is detached from both the limiter and inner wall and recycling occurs in an open divertor region near the X-point. With heating applied above a certain threshold power value, a transition occurs to an improved plasma confinement (H-mode) regime, which depends upon the toroidal magnetic field. Fundamental characteristics include a rise in energy content and plasma density. An increase in electron temperature also occurs near the separatrix, which produces a pedestal in the temperature profile, and a flatter density profile with a steep gradient near the separatrix. The energy confinement time in the H-mode exceeds that with limiter discharges (L-mode) by more than a factor of two, as shown in Fig.42.

Exploitation and investigation of plasma behaviour in the H-mode is a major component of the JET experimental programme and several new areas of H-mode behaviour have been addressed for the first time in JET. NB counter-injection experiments established that the power threshold for the H-mode was perhaps 10-20% lower than for co-injection. The most significant difference in behaviour was observed in central MHD behaviour. With counter-injection, frequent, small-amplitude sawteeth were observed which

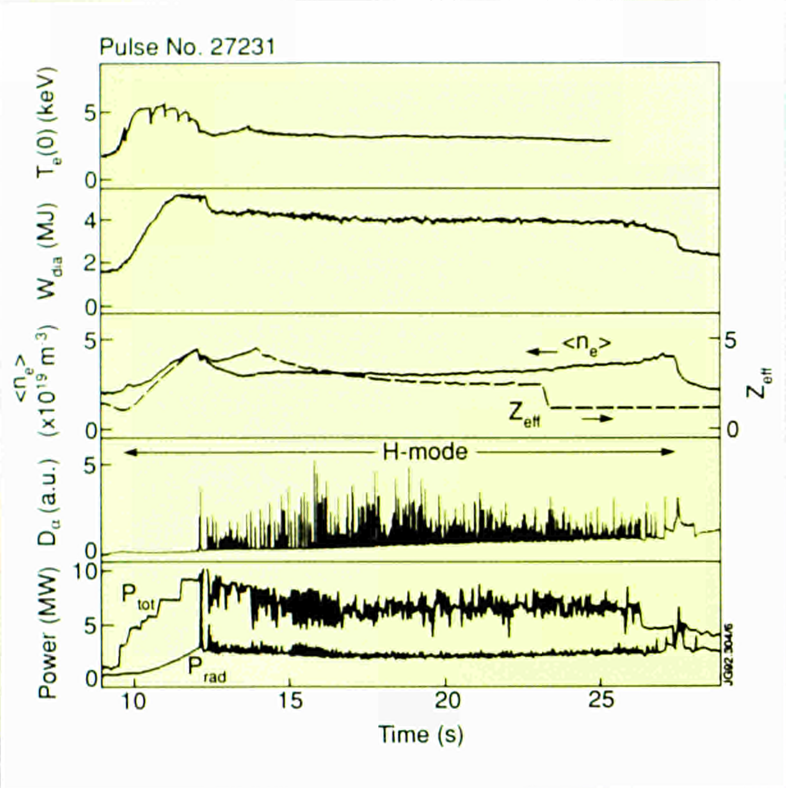


Fig.43: Behaviour of main plasma parameters during an 18 second H-mode at 2MA, held in a steady-state by ELM activity (visible on the D_α signal).

maintained flattened central plasma profiles. As a result, the energy confinement of counter-injection H-modes was somewhat reduced relative to that obtained with co-injection, with only the best counter-injection cases exhibiting energy confinement times equal to twice L-mode scaling values.

During some H-modes, Edge Localized Modes (ELM's) of instability appear at the plasma edge, leading to some loss of confinement. The majority of JET H-modes are free of ELM's. In this situation, the density rises throughout the H-mode and it is usually terminated by a high level of radiation (or, at the highest powers, by an impurity 'bloom'). However, impurity influxes can be reduced, and the H-mode duration extended, by heavy gas puffing. Even so, in these cases, the density continues to rise and the regime never achieves stationary conditions. Recent experiments have shown that, under certain conditions, a steady-state H-mode can be achieved in which the particle and impurity influxes are controlled by regular ELM's. These have been produced reliably by gas puffing from the midplane. After an initial phase, ELM's appear and in spite of continuous gas puffing, the plasma density, radiated power, impurity contact, stored energy and neutron production remained consistent for the duration of the H-mode.

Considerable efforts have been made to understand ELM behaviour and to investigate techniques for reliable ELM production to permit steady-state H-modes to be established. Steady-state

MARFE

A MARFE (Multifaceted Asymmetric Radiation From the Edge) is a toroidally symmetric band of cold, highly radiating plasma which normally forms at the plasma inner wall. It can occur when the plasma edge density is high and results from of an imbalance between the power flowing along magnetic field lines in the edge and the power lost locally due to radiation. A MARFE grows rapidly, on a timescale of ≈10-100 milliseconds, but it can persist for several seconds. In some cases, the MARFE leads to a disruption, but in others the main consequence is a reduction in the edge density.

ELM

An ELM (Edge Localized Mode) is an edge instability which occurs in the high confinement (H-mode) regime. It affects a narrow region in the plasma edge and leads to a loss of particles and energy from the edge on a timescale ≤1 millisecond and therefore is a rapid, but transient, instability. However, ELM's can occur as repetitive instabilities which cause a reduction in the time-averaged energy and particle confinement time.

H-modes in which plasma parameters are held constant by ELM activity have been produced at 2MA/2.3T, with a maximum duration of 18 seconds being achieved (Fig.43). At 2MA, the duration limit of the H-mode was set by technical, rather than plasma-related, limitations. This regime was established by combining strong gas puffing and using NB and off-axis ICRF heating. Energy confinement times of these H-modes were high, typically twice L-mode values.

Recent theories of the H-mode postulate that the L-to-H transition is due to shear in the edge poloidal velocity driven by a radial electric field gradient. A new active charge exchange diagnostic for the measurement of edge ion temperatures and poloidal rotation has permitted this question to be investigated in JET. Initial measurements have shown that the poloidal rotation velocity in the plasma edge does increase gradually during the period of the H-mode transition, reaching values ~$2\times10^4 ms^{-1}$. However, there was no rapid jump as has been reported from other tokamaks.

Energy confinement predictions for H-mode operation in Next Step tokamaks require a scaling law based on tokamaks of different dimensions. Throughout 1990/91, JET has continued to compile an H-mode database for global confinement scaling at the request of the ITER Project. The work was performed as a combined effort from JET and from other tokamaks (DIIID (General Atomics, USA), ASDEX (IPP Garching, FRG), JFT2M (JAERI, Japan), PBXM and PDX (PPPL, USA)). The database consists of measurements from neutral beam heated H-modes and the fit to the ELM-free data set gives:

$$\tau_E=0.082\ I_p^{1.02}B_T^{0.15}P^{-0.47}A^{0.5}R^{1.6}\kappa^{-1.9}$$

where τ_E(s) is the energy confinement time, I_p(MA) is the plasma current, P(MW) is the loss power, A is the atomic number, R(m) is the major radius and κ the plasma elongation.

Progress Towards a Reactor

Next Step Related Issues

A number of experiments have been carried out on certain issues of particular relevance to Next Step devices. These are described below:

AC Operation of JET

The toroidal plasma current required in a tokamak can be driven either inductively, or by various means of non-inductive current drive. Inductively current-driven tokamaks reactors are necessarily pulsed devices, in which electricity production during the down-time must be maintained by means of an external energy storage. This down-time, and thus the demands for external

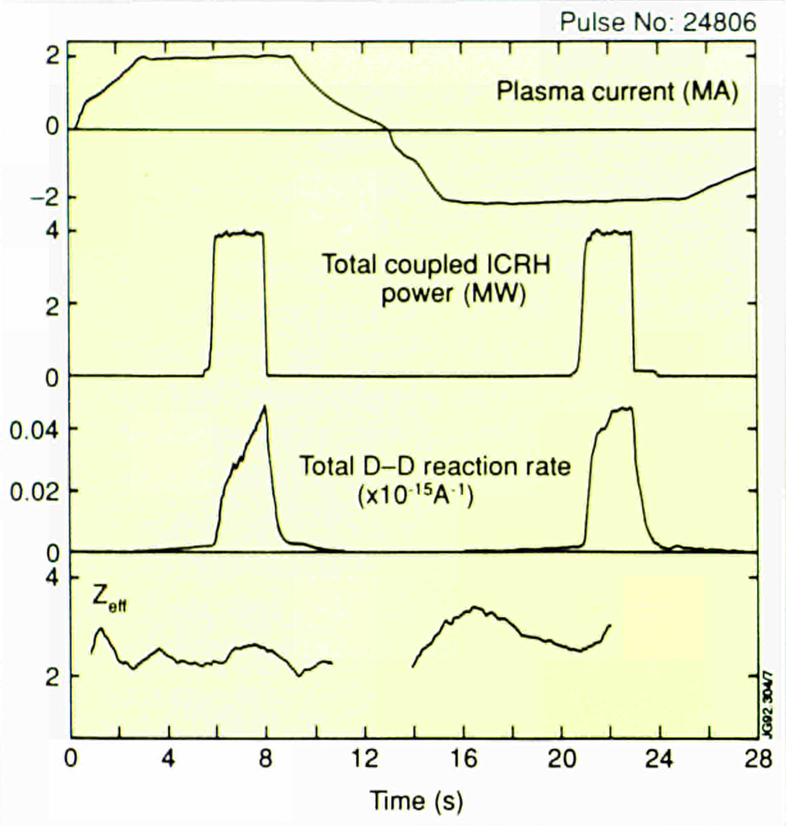


Fig.44: Parameters of a full cycle AC discharge. The plasma current, total ICRF power, total D-D reactive rate and effective ion charge, Z_{eff} are shown as a function of time.

storage, can be minimised by using AC operation, in which the current in the plasma current alternates in direction between subsequent burn phases. In AC operation, no re-charging of the central solenoid between burn-phases is necessary. A full cycle of AC operation has been demonstrated in JET. The first plasma is generated using low voltage breakdown with no bias current in the central solenoid; the loop voltage is applied directly by the solenoid power supply. The second plasma is generated by interrupting (at 13s) the current through the solenoid and redirecting this through a resistor. Both breakdown scenarios are equivalent to those used in normal JET operation.

The primary issue to be resolved is the one of the relative plasma performance and plasma purity of the successive cycles. Fig.44 shows that there is no significant difference in these respects between the first and second phase of an AC discharge. The effective ion charge, Z_{eff}, is approximately the same in both half-cycles. In addition, the plasma density in both phases is the same, maintained by the density feedback system. The electron temperature and the total radiated power are also equal for both cycles. The use of AC inductive current drive for a tokamak fusion reactor allows a reactor to operate with a minimum plant re-circulating power. It further allows more flexibility in the optimisation of the fusion power per unit capital

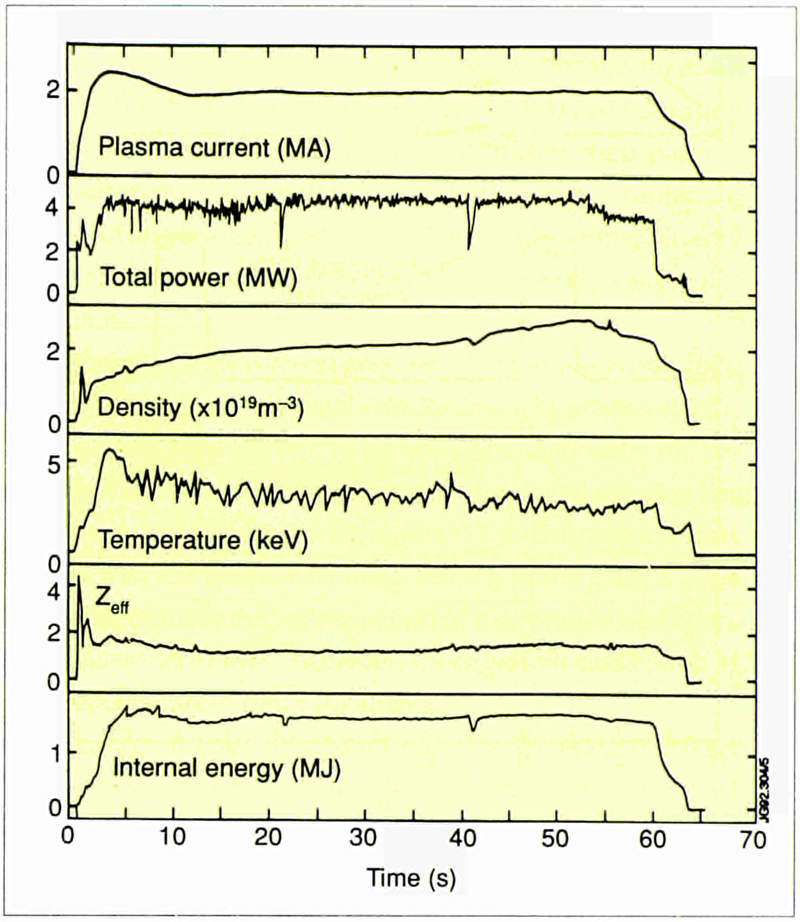


Fig.45: Various time traces for a one minute pulse with LHCD and ICRF heating.

investment. The machine parameters and the operating conditions are not restricted by the requirements imposed by non-inductive current drive methods.

Long Pulse Operation

In order to approach steady-state conditions and to demonstrate long pulse operation relevant to a Next Step device, discharges have been developed up to 1 minute in duration at 2MA and 1.9T, with additional heating powers up to 5MW, as shown in Fig.45. This example employed 2.5MW of ICRF heating together with 1MW of Lower Hybrid Current Drive (LHCD). ICRF power was applied to raise the electron temperature and reduce the plasma resistivity, allowing a more efficient exploitation of the primary inductive flux. Further inductive saving was obtained by applying LHCD at 1MW power level for 50 seconds. The combined application of the two RF systems allowed an extension in duration of the basic discharge by ~25 seconds.

However, the particle behaviour did not approach steady state. The gas flow required to maintain a constant density fell gradually over the first 30s. This fall was consistent with the model of deuterium pumping by dynamic retention in material surfaces, the increasing deuterium inventory in the surfaces gave rise to a larger efflux back into the plasma. However, instead of falling asymptotically to a low value, the gas flow reached zero at a finite time indicating an additional particle source, or a change in the dynamic retention. This particle source then contributed to the subsequent density rise, which was predominantly deuterium. A new density steady-state was not reached before the end of the heating pulse, but density pump-out occurred either if the heating power was stepped down or the plasma current was ramped down and so did not lead to a disruption in the termination.

The time into the pulse at which the gas flow became zero fell from 60s for about 2MW additional heating to 25s for 5MW, as would be expected from an effect of limiter temperature. Multiple 2.7mm pellets (contributing <10% of the externally supplied particles) have been injected into such plasmas to probe the edge region. The pump-out time of the pellet supplied density was initially short but increased until after the gas flow had reached zero, the density simply stepped up at each pellet and remained, suggesting complete recycling of the injected deuterium. Further analysis is underway, in order to understand these results which are important for density control in future devices.

Fusion Performance

The figure of merit which determines the fusion performance is the triple fusion product, $n_D T_i \tau_E$, and how close this approaches the value required for a fusion reactor. In a reactor operating with deuterium and tritium, this product must exceed the value $5x10^{21}m^{-3}$keVs for ignition conditions. The general performance obtained in JET and other tokamaks is shown in Fig.31. A range of experiments were carried out on 3-4 MA double-null X-point configurations, with input powers up to 30MW. ICRF heating, NB injection and combined heating were used with both D and He^3 beams. Low and high density regimes were explored. Good confinement and high stored energy in ELM-free H-mode regimes was achieved in a 4MA double-null configuration. A record 12.7MJ of stored plasma energy was obtained with a total input power of 20MW. The electron temperature reached ~10keV in the sawteeth free regime.

Considerable efforts were made to increase the fusion yield of plasmas in preparation for the PTE. The single-null X-point configuration with the ∇B ion drift away from the X-point, showed that the more optimal power distribution between the inner and outer X-point strike zones yielded a longer delay before the impurity influx terminated the high performance phase in this regime. Optimization of 3MA/2.8T discharges in this configuration yielded the highest

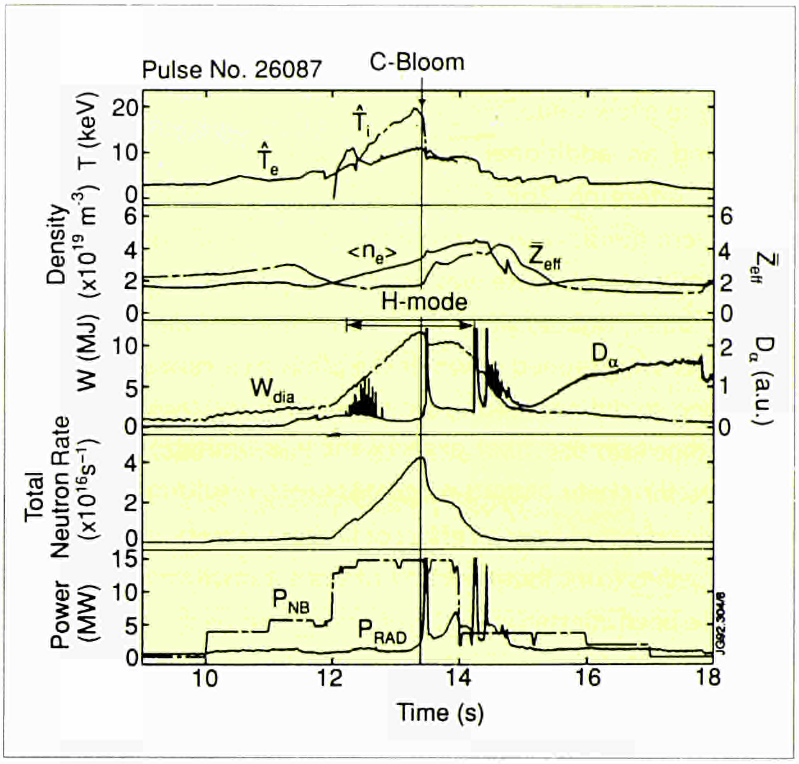


Fig.46: Time development of central electron and ion temperatures, volume-averaged electron density, line-averaged Z_{eff}, plasma diamagnetic energy, D_α emission, total neutron rate, and NB and radiated powers as a function of time for the highest performance deuterium pulse.

neutron yield obtained to date of 4.3 x $10^{16}s^{-1}$ (see Fig.46). The parameters obtained were central density of $4.1 x 10^{19}m^{-3}$, central ion temperature of 18.6keV and a confinement time of 1.2s. This corresponded to a fusion product $n_D(0)\tau_E T_i(0)=9.2 x 10^{20}m^{-3}$keVs. With an optimized tritium concentration, this pulse would have produced a fusion power of 11MW and an equivalent fusion amplification factor Q_{DT}=1.14, which exceeds the breakeven value.

Preliminary Tritium Experiment

To perform a deuterium-tritium experiment at this stage in the JET programme, it was necessary to limit the total neutron production to less than about $1.5 x 10^{18}$ neutrons so that the resulting vessel activation would be compatible with the pumped divertor modification work scheduled for 1992/1993. In addition, the total amount of tritium available was restricted to ≈0.2g (≈2000Ci), as the JET tritium processing plant is not scheduled to come into operation until 1993. Taken together, these limitations restricted, to a few, the total number of high performance discharges in this series of experiments.

The target plasma for this experiment was a hot ion H-mode in the single-null configuration at I_p = 3MA and B_T = 2.8T, described in the previous section. Neutral beam injection is an effective way of introducing tritium into the type of discharge. It ensures that tritium reaches the hot dense centre of the discharge where the reactivity is highest and minimises the amount of tritium injected into the torus. Two of the sixteen central beam injectors were adapted to operate with tritium (1.5MW) and the remaining fourteen injectors operated with deuterium (~13MW). The injectors were deliberately operated below maximum performance to ensure high reliability.

To minimize activation of the vessel, optimization of this discharge was performed in deuterium, followed by several shots at central tritium concentrations ~0.1% (obtained by the doping of two beam sources with 1% tritium). In the two full power pulses to which the final experiment was limited, tritium was introduced from two neutral beam sources, resulting in a central tritium concentration of ~10%. The performance of these discharges lay in the middle of the range of the hot-ion modes produced during the optimization experiments. Both D-T pulses were similar and each produced fusion power in excess of 1.5MW. Figure 47 shows the time development of the characteristic parameters for one of these pulses. All of these parameters increased throughout the H-mode phase of the discharge which starts at 12.4s and ended with a "carbon bloom" at 13.3s. The main plasma parameters at this time are listed in Table.7.

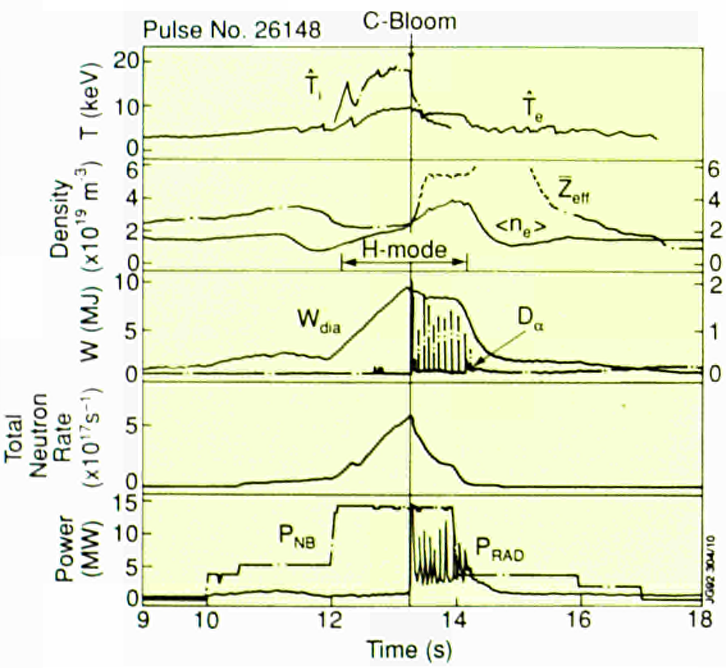


Fig.47: The time development of the central electron and ion temperatures, the volume-averaged electron density, the line-averaged Z_{eff}, the plasma diagmagnetic energy, the D_α emission, the total neutron rate, and the NB radiated powers as a function of time for one of the tritium pulses.

Table 7: Main Parameters of D-T Pulse No: 26148

Parameter	Value
NB power, (P_{NB})	14.3MW
Central (D + T) density, (n_{DT})	$2.4 \times 10^{19} m^{-3}$
Central electron temperature, ($T_e(0)$)	9.9 keV
Central ion temperature, ($T_i(0)$)	18.8 keV
Plasma energy, (W)	9.1 MJ
Confinement time, (τ_E)	0.9s
Fusion product, ($n_{DT}T_i(0)\tau_E$)	$3.8 \times 10^{20} m^{-3}$keVs
Density ratio T/(D+T), [$n_T/(n_D+n_T)$]	10%
Neutron emission ratio	$6.0 \times 10^{17} s^{-1}$
Fusion factor, Q	0.15

In planning and executing the deuterium-tritium experiment, the TRANSP simulation code was used to check the internal consistency of the measured data and to estimate the fraction of neutrons which were produced by thermal-thermal, beam-thermal and beam-beam reactions on the basis of the measured profiles of density, and ion and electron temperature and of the effective ion charge, Z_{eff}

The consistency of the data is demonstrated by the good agreement obtained between the measured and simulated emission of, predominantly, 14MeV neutrons (shown in Fig.48). The simulations showed that ≈50% of the neutrons were produced by thermal-thermal reactions while the remainder were mostly by beam-thermal reactions with only a small fraction by beam-beam reactions. The peak total neutron emission rate was 6.0×10^{17} neutrons per second with an accuracy of ±7%. The total fusion releases (α-particles and neutrons) were 1.7MW of peak power and 2MJ of energy. The actual fusion amplification factor, Q_{DT} was 0.15. With an optimum tritium concentration, this pulse would have produced a fusion power ≈ 5MW and a nominal $Q_{DT} \approx 0.46$. Analysis of these discharges is continuing to improve estimates of the tritium diffusion rate, to understand the MHD behaviour, and to evaluate transport and confinement. Overall, the experiment has produced invaluable scientific and technical data for the preparation of the full D-T phase of JET planned for 1996.

The introduction of a significant fraction of tritium into a tokamak plasma for the first time is the most significant accomplishment of the recent experimental programme. This experiment was noteworthy not only for the production of over 1.5MW of fusion power, but for several substantial technical achievements: the establishment of procedures for the monitoring and tracking of tritium in a tokamak environment; the demonstration of reliable techniques for the handling of tritium and its introduction into the tokamak; and

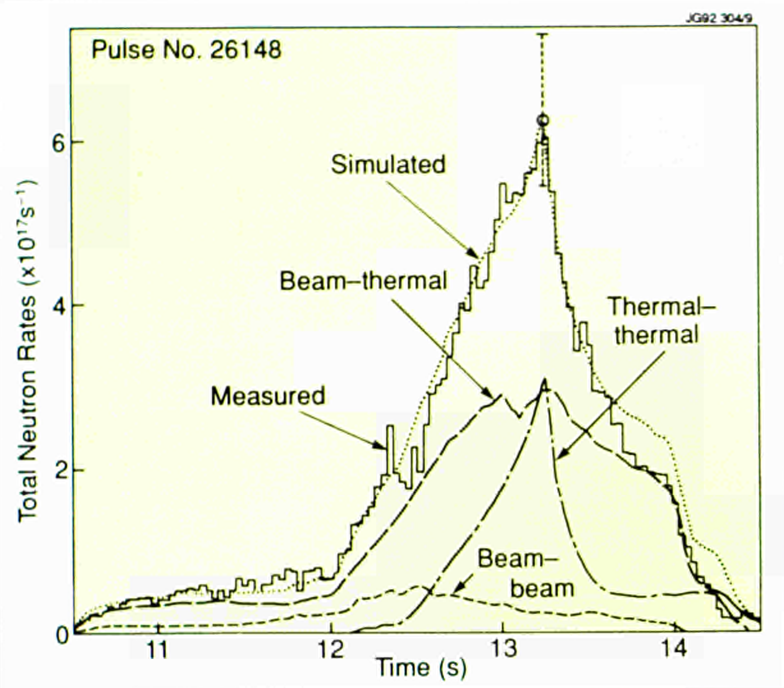


Fig.48: The measured and simulated total neutron rates (predominantly 14MeV neutrons) for one of the tritium pulses.

the first tritium neutral beam injection. Significant scientific information was also obtained in a number of areas: validation of simulation codes used to extrapolate from D-D to D-T performance; investigation of the transport of tritium in a tokamak plasma; and the study of rates of removal of tritium from the tokamak first wall and ancillary systems.

Summary of Scientific Achievements

Experiments during 1991 have encompassed several objectives: demonstration of improved tokamak performance in several areas; enhancement of fusion yield; improvement of understanding of the fundamental physics which determines plasma behaviour, and exploration of a number of issues central to the design of a Next Step tokamak. A wide range of experiments has been performed in addressing these issues and significant progress has been achieved. In addition, the foundation has been laid for a successful transition to the new phase of JET which will exploit the pumped divertor to address the key questions relating to the control of heat and particle exhaust and of impurity influxes.

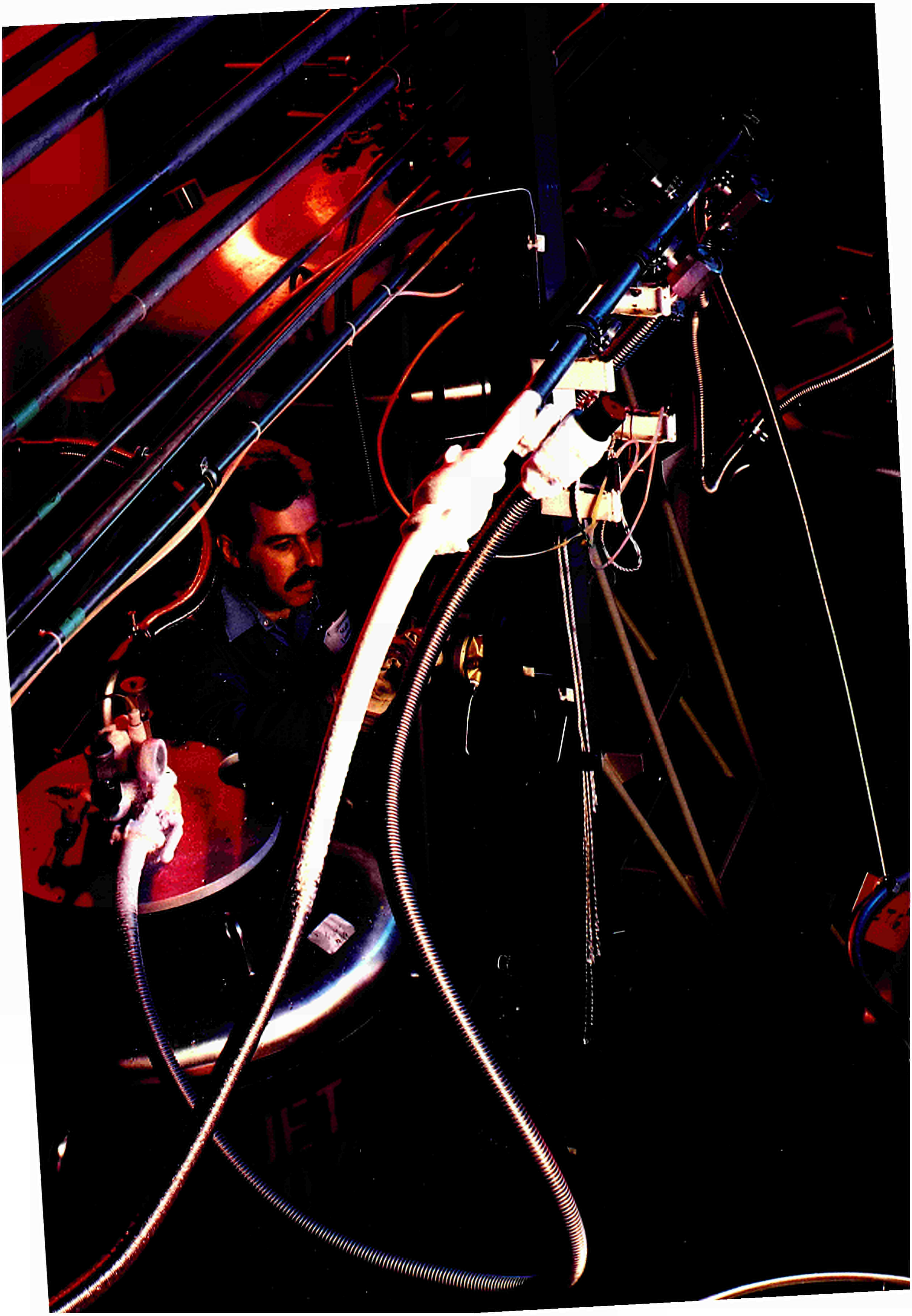
The general JET performance has improved, as follows:

- Investigation of the performance of diverted plasmas utilizing the newly installed toroidal X-point targets was a central activity throughout the campaign. With beryllium tiles for the lower target and carbon fibre composite (CFC) tiles for the upper target permitted a careful study of the relative merits of the two materials. In addition, two designs of CFC tile were examined and limitations arising from heating of tile edges were studied. The improved design of CFC target tile, which shielded the tile edges, yielded about a factor of two improvement in power handling capabilities. Comparisons of H-mode performance using CFC and beryllium targets showed that the two materials gave similar results at moderate to high densities, but the CFC targets permitted considerably better fusion yield to be achieved in the relatively low density hot-ion H-modes;

- Steady-state H-modes, with plasma parameters held constant by ELM activity, were produced in a 2MA plasma for a maximum duration of 18 seconds. The duration limit of the H-mode was set by technical, rather than plasma-related limitations. This regime was established by combining strong gas puffing and using NB and off-axis ICRF heating. Energy confinement of these H-modes was high, typically corresponding to twice the L-mode value.
- Evidence of the importance of local modifications of the current profile has been obtained from the first demonstration of fast wave current drive using the ICRF system antennas as a phased array. By varying the phase of the antenna array so as to reverse the relative directions of the predicted anti-parallel currents, sawteeth could be stabilized for periods of up to 2s, resulting in a significantly peaked temperature profile;
- Using ICRF power and lower hybrid current drive to enhance inductive drive, a discharge at 2MA plasma current was maintained and heated for 1 minute duration;
- Operation of the tokamak as an AC device offers an alternative route to quasi-continuous operation without the overheads entailed in external current drive systems. Reliable operation and additional heating of a two-cycle plasma was demonstrated in JET at 2MA and plasma parameters were found to be very similar in the two cycles. Although it was not possible to demonstrate a smooth transition between the two cycles with zero dwell time, dwell times as small as 50ms and as long as 6s were achieved without difficulty;
- The hot-ion mode was further enhanced in JET to produce the highest neutron yield in a deuterium plasma of $4.3\text{x}10^{16}s^{-1}$. The fusion product was $9.2\text{x}10^{20}m^{-3}$keVs, which corresponded to an equivalent fusion amplification factor Q_{DT}=1.14. This exceeded breakeven conditions and was within a factor of 5-8 of that required in a reactor;
- The most significant achievement of 1991 was the first tokamak discharges in D-T fuelled mixtures, which were undertaken

within limits imposed by restrictions on vessel activation and tritium usage. Tritium was introduced, into hot-ion H-mode discharges, from two neutral beam sources delivering 1.5MW within a total of 14.3 MW heating. This produced a tritium concentration of about 11% at the time of peak performance, when the total neutron emission rate was $6x10^{17}s^{-1}$. The integrated total neutron yield was $7.2x10^{17}$. The neutrons came about equally from beam plasma and thermal processes. These levels are equivalent to total fusion releases (α-particles and neutrons) of 1.7MW peak power and 2 MJ energy. The techniques used for introducing, tracking, monitoring and recovering tritium have been demonstrated to be effective. A valuable body of experience has been obtained which will be useful in planning the future more extensive D-T experiments on JET. However, these results were obtained in a transient state and could not be sustained in a steady state. Ultimately, the influx of impurities caused a degradation in plasma parameters.

In 1992, a New Phase of JET will start, aimed at demonstrating effective methods of impurity control in operating conditions close to those of a Next Step tokamak in an axisymmetric pumped divertor configuration. This is described in further detail in the following section.



Future Programme

Introduction

The initial JET objectives still remain valid and continue to provide the focus of the Project's plans. These original objectives were set out in the JET Design Proposal in 1978, as follows:

'The essential objective of JET is to obtain and study a plasma in conditions and dimensions approaching those needed in a thermo-nuclear reactor. These studies will be aimed at defining the parameters, the size and the working conditions of a Tokamak reactor. The realisation 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; and

iv) the study of α-particle production, confinement and consequent plasma heating.

The problems of plasma-wall interaction and of heating the plasma must, in any case, be solved in order to approach the conditions of interest.

An important part of the experimental programme will be to use JET to extend to a reactor-like plasma, results obtained and innovations made in smaller apparatus as a part of the general tokamak programme. These would include: various additional heating methods, first wall materials, the control of the plasma profiles and plasma formation.'

At the start of 1991, the JET Project entered the last year of its planned Phase III - Full Power Optimization Studies. The general objectives of the experimental programme were to optimise performance and to explore the domain of high performance plasmas, studying aspects of plasma physics and engineering including: profile and heating effects; exhaust phenomena; and divertor edge physics. Priority was given to study of the power and energy handling capability of newly installed plasma facing components in regimes relevant to the Next Step and to the New Phase of JET.

Objectives of JET

The essential objective of JET is to obtain and study plasma in conditions and with dimensions approaching those needed in a thermonuclear reactor. These studies will be aimed at:

1. Scaling of plasma behaviour as parameters approach the reactor range;

2. Plasma-wall interactions in these conditions;

3. Plasma heating; and

4. Alpha-particle production, confinement and consequent plasma heating.

Extensive studies had been made in the first and third areas of work of JET's objectives: reactor relevant temperatures (up to 30 keV), densities (up to 4 x $10^{20}m^{-3}$) and energy confinement times (up to 1.8s) had been achieved in separate discharges. The second area of work had been well covered in the limiter configuration for which JET was originally designed. However, the highest performance JET discharges had been obtained with a 'magnetic limiter', (or X-point configuration). The duration of the high performance phase of these discharges exceeded 1.5s; this was achieved by careful design of the targets and specific operation techniques, but is limited, ultimately, by an unacceptably high influx of impurities, characterized by a rapid increase in electron density, effective ion charge and radiated power (referred to as the 'bloom').

The fourth area of work had been started by earlier studies of energetic particles produced as fusion products or by ion cyclotron resonance heating (ICRH). It was addressed further during 1991 by the first tokamak plasma experiments in deuterium-tritium mixtures. The high performance achieved in deuterium discharges, together with the experience gained in making substantial modifications to JET in a beryllium environment and with significant vessel activation, gave confidence that an experiment with about 10% tritium in the plasma could be performed and would provide data that could be used to plan an effective campaign of deuterium-tritium experiments in 1996.

Following extensive discussions during 1991, the JET Council approved the policy of a step-wise approach to the introduction of tritium in advance of the full D-T phase of JET operations. As a first such step, having obtained all necessary regulatory approvals, JET successfully carried out a preliminary tritium experiment in November 1991 (as already described). A release of fusion energy in the Megawatt range in a controlled fusion device was achieved for the first time in the world.

The most recent experiments on JET achieved plasma parameters close to breakeven values for about a second, resulting in large bursts of neutrons. However, in spite of the plasma pulse continuing for many seconds after reaching peak plasma values, the neutron count fell away rapidly as impurities entered the plasma and lowered its performance. This limitation on the time for which the near-breakeven conditions could be maintained is due to the poisoning of the plasma by impurities (the 'bloom'). This has further emphasised the need to provide a scheme of impurity control suitable for a Next Step device.

At its meeting on 19th December 1991, the Council of Ministers adopted Decisions concerning the Euratom Fusion Programme in the period to the end of 1994 and a modification to the Statutes of JET which prolongs its statutory lifetime by four years until 31st December 1996. The extension is to allow JET to implement the new Pumped Divertor Phase of operation, the objective of which is to establish the effective control of plasma impurities in operating conditions close to those of the Next Step. This programme of studies will be pursued before the final phase of full D-T operations in JET.

During 1991, a large proportion of JETs effort was devoted to preparation for the new phase of operations. Intensive design and procurement activities were continued to ensure timely delivery of the many components of the pumped divertor and related modifications, to be installed during the 1992/93 shutdown.

JET Strategy

Present achievements show that the main objectives of JET are being actively addressed and substantial progress is being made. The overall aim for JET can be summarised as a strategy "to optimise the fusion product ($n_i T_i \tau_E$)". For the energy confinement time, τ_E, this involves maintaining, with full additional heating, the values that have already been reached. For the density and ion temperature, it means increasing their central values $n_i(0)$ and $T_i(0)$ to such an extent that D-T operation would produce alpha-particles in sufficient quantities to be able to analyse their effects on the plasma.

The enhancements to JET aim to build up a high density and high temperature plasma in the centre of the discharge (with minimum impurity levels) where alpha-particles could be observed, while maintaining an acceptably high global energy confinement time τ_E. The mechanisms involved are to decouple the temperature profile from the current density profile through the use of lower hybrid current drive and neutral beam injection to ensure that, at higher central temperatures, the current density in the centre does not reach the critical value that causes sawteeth oscillations.

This involves the following:

a) Increasing the Central Deuterium Density $n_D(0)$ by:
- injecting deuterium pellets and higher energy neutral beams to fuel the plasma centre and dilute impurities;
- injecting pellets to control the influx of edge material;
- stabilising the m=2, n=1 magnetic oscillations present at the onset of a disruption with magnetic perturbations produced from a set of internal saddle coils which will be feedback controlled;

b) Increasing the Central Ion Temperature, $T_i(0)$ by:

- lengthening the sawtooth period;
- controlling the current profile (by lower hybrid current drive in the outer region, and by counter neutral beam injection near the centre) to flatten the profile;
- on-axis heating using the full NB and ICRF additional heating power (24MW, ICRH, and 20MW, NB)

c) Increasing the Energy Confinement time τ_E by:

- increasing up to 6MA the plasma current in the full power, H-mode operation in the X-point configuration;

d) Reducing the impurity content, by:

- using beryllium as a first-wall material to decrease the impurity content;
- controlling new edge material by using the pumped divertor configuration.

In parallel, JET's preparations for the full D-T phase of operations have continued. In particular, JET has completed installation of all the main components of the active gas handling system in readiness for the system's commissioning programme. The JET Project is now in a position to enter the new phase of operations and next-step oriented studies in a pumped divertor configuration, leading to its final phase of full D-T operations. The following sections describe various developments which are underway on JET to implement these systems.

The New Phase of JET: Pumped Divertor

The following sections describe developments which are underway on JET to implement the proposed pumped divertor configuration in the New Phase.

Motivation and Status

Plasma dilution is a major threat to a reactor. The entrainment of impurities in a forced flow of plasma towards the divertor target plates is a candidate concept for impurity control in Next Step tokamaks. This form of active impurity control is the focus of the New Phase of JET, scheduled to start in JET in 1992. First results should become available in 1993 and the Project will continue to the end of 1996. The aim of the New Phase is to demonstrate, prior to the full use of tritium, effective methods of impurity control in operating conditions close to those of a Next Step tokamak, with a stationary plasma (10s-1 minute) of 'thermonuclear grade' in an axisymmetric pumped divertor configuration. Specifically, the New Phase should demonstrate:

- control of impurities generated at the divertor target plates;
- decrease of the heat load on the target plates;
- control of plasma density;
- an exhaust capability;
- a realistic model of particle transport.

The New Phase of JET should demonstrate a concept of impurity control; determine the size and geometry needed to realise this concept in a Next-Step tokamak; allow a choice of suitable plasma facing components; and demonstrate the operational domain for such a device.

Key Concepts

The key concept of the proposed JET pumped divertor is that since sputtering of impurities cannot be suppressed at the target plate of a divertor, these impurities should be confined in the vicinity of the target plate itself. This confinement can be achieved by maintaining a strong directed flow of plasma particles along the divertor channel towards the target plates, to prevent back diffusion of impurities by the action of a frictional force. An important feature of the configuration is the connection length, along the magnetic field lines, between the X-point region and the target plates. This distance should be long (~3-10m) to achieve effective screening of the impurities. In addition, the X-point should be well separated from the target plates and in JET these objectives can only be met by coils which are internal to the vessel.

The formation of a target plasma in the divertor channels is another essential feature of the pumped divertor. The cold (radiatively cooled) and dense plasma, which is expected to form in front of the target plates, plays a number of key roles:

- it radiates a significant fraction of the plasma input power, thus reducing the heat load on the target plates;
- it reduces impurity production by reducing the ion temperature;
- it reduces the probability of the impurities diffusing back to the plasma

In the vicinity of the outer target plate, a pumping chamber with a cryogenic pumping system is planned to control the main plasma density. It should be noted that only a small fraction of the hydrogenic neutrals generated at the target plates are expected to be pumped. Some of the neutrals will be able to recycle towards the X-point region, re-enter the scrape-off plasma there and enhance the plasma flow to the target plate. This local recirculation of hydrogenic particles should improve the impurity confinement. If required, gas can also be injected near the X-point to further increase the particle flow in the divertor channels, and be pumped away by the cryopump.

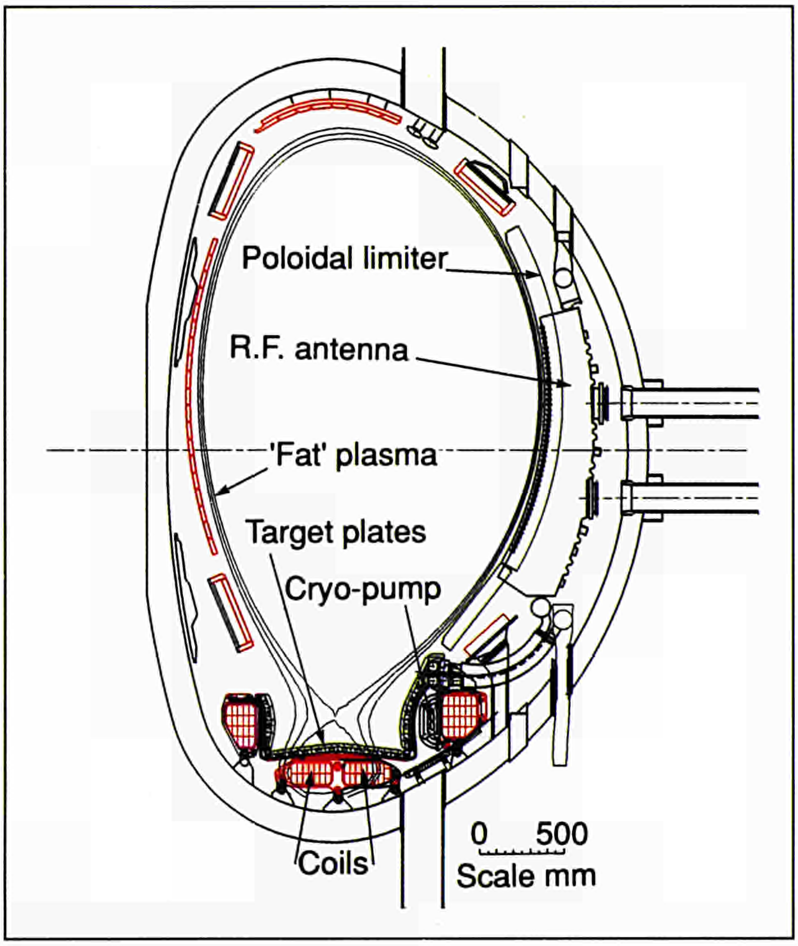


Fig.49: Configuration with so-called 'fat' plasma.

Evolution of engineering aspects

During 1991, there were no major changes to the pumped divertor concept or components, as had been set out in the 1990 Annual Report. Detailed design of all major components proceeded and procurement contracts were placed for all long delivery items. In addition, a significant effort was devoted to studying assembly procedures and design of the related assembly jigs and tools.

Magnetic Configuration and Divertor Coils

The multicoil configuration is a most flexible and versatile design, which allows investigation of a range of plasma and divertor configurations. The range of configurations is illustrated by two cases, the so called, fat and slim plasmas (Figs.49 and 50), which can be obtained at plasma currents up to 6MA. Since all four coils have independent power supplies, great operational flexibility can be achieved:

- A "fat" plasma can be made using only the central bottom coils. This has a large volume, moderate elongation but short connection lengths;
- A "slim" plasma can be produced by adding currents in the side coils. Longer connection lengths are achieved at the expense

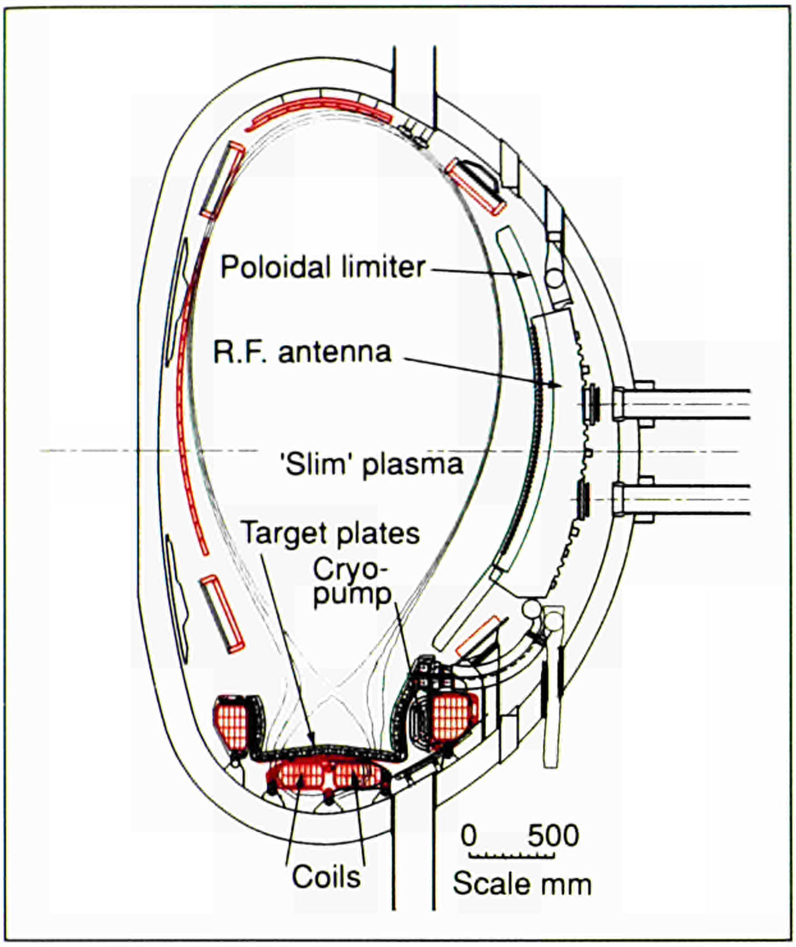


Fig.50: Configuration with so-called 'slim' plasma

of a small plasma volume. In addition, the elongation is larger and the vertical stability margin is reduced.

The "slim" plasma has a minimum connection length of 7m and the "fat" plasma has a minimum connection length of 3m, both at a plasma current of 6MA. The connection length increases at lower plasma currents. This gives the possibility of studying the effect of the connection lengths to be adjusted independently of the plasma current, and separately on the inboard and outboard sides of the X-point. In addition, the strike zone of the separatrix and scrape-off layer can be swept across the target plates to reduce the power deposition profile to an acceptable time averaged value.

There was no significant change to the coil design in 1991. The coils are of conventional construction (water cooled copper, epoxy glass insulation) and are contained in thin Inconel cases. The coils are supported from the vacuum vessel by hinged links, which allow the vessel to expand independently of the coils. The links are joined to the coils by Inconel clamps and to the vacuum vessel by welded on blocks. These blocks will be the first components to be welded onto the vessel and are designed to form a well aligned ring of supports.

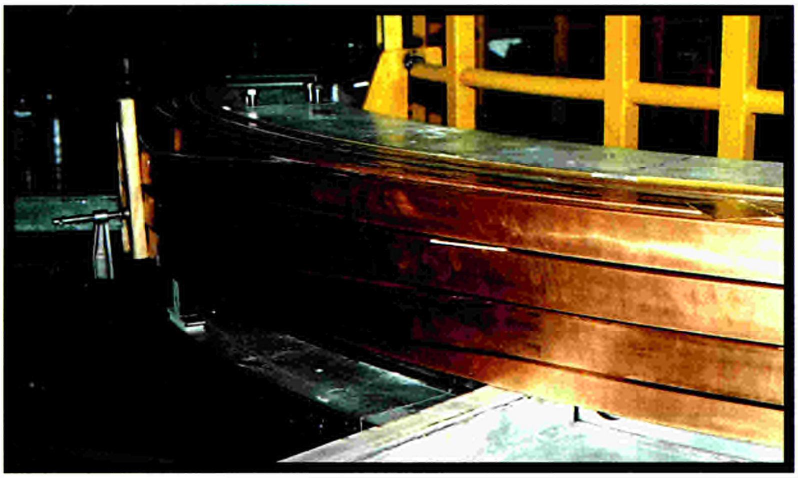


Fig.51: Pre-assembled divertor coil at Contractor's factory

During 1991, the conductors for the first two coils have been formed and pre-assembled (as shown in Fig.51). The press tools for the cases have been made and tested and the supports were almost completed. Considerable effort has been devoted to studying and planning the in-vessel assembly operations. Problems of in-vessel work include restricted space, restricted size of access port and restricted choice of materials. Lifting operations require special equipment, as no in-vessel crane is available. A number of possible solutions to this problem were still being examined at the end of 1991.

The topology of the coils and their terminals strongly affects the assembly sequence for the coils and cases. This sequence has been studied in detail using CAD models to check feasibility and determine the optimum. All in-vessel operations must be thoroughly pre-tested to ensure the reliability of the coils and to avoid delays or problems during assembly. For example, an epoxy resin impregnation test model is nearing completion (see Fig.52), which will also be used to check the case welding process.

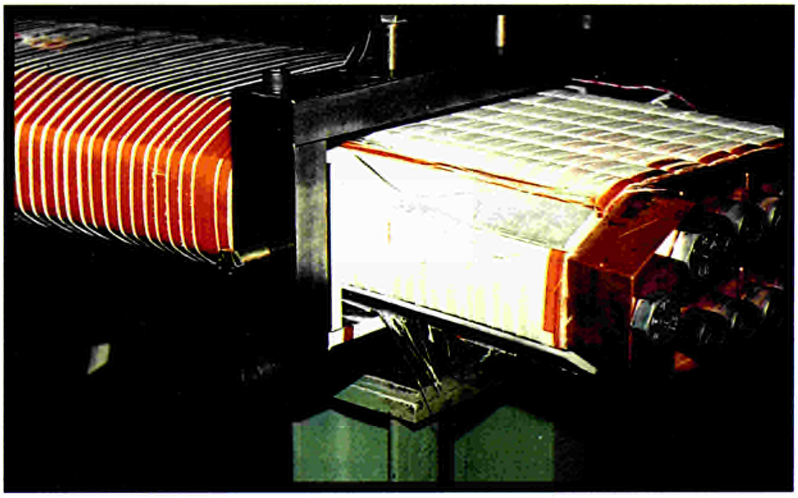


Fig.52: Divertor coil impregnation test model ready for insertion into case

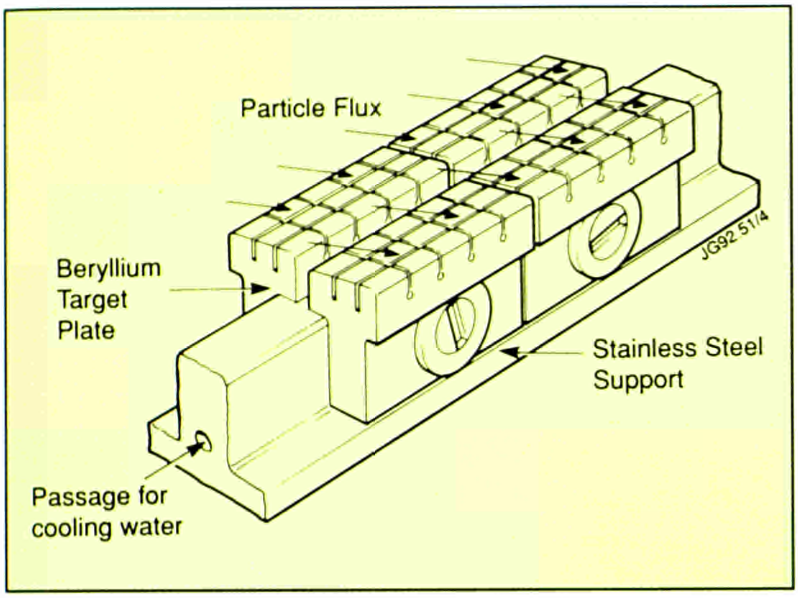


Fig.53: Beryllium tiles for target plate (Mark I)

Target Plates

The target plates feature three elements in a U-configuration to accommodate the new plasma and divertor contours. Horizontal plates at the bottom intersect the heat flux conducted along field lines. Therefore, these bottom plates are subjected to a severe power deposition. Vertical plates on either side intersect the power radiated from the divertor plasma, and receive only a modest heat load. The horizontal and vertical plates are split into 384 radial elements grouped into 48 modules of eight elements each.

In this configuration, the peak power density on the bottom target plates would be unacceptably high even for the most advanced heat sinks. Consequently, sweeping the field lines is essential. Then the load can be accommodated by hypervapotrons of the type used on the JET neutral beam systems. In contrast to the bottom elements, the side elements receive only modest radiative heat load from the divertor plasma, and should not cause difficulties.

Installation of the target plates will proceed in two steps. Step I will use radiatively-cooled tiles as target plates, while Step II will use water-cooled beryllium clad hypervapotrons. Step I provides a simple and robust design, which will allow the early exploration of the pumped divertor operation parameter space. Step II will be used for impurity control studies at high power and long pulse duration.

The conceptual design of the solid beryllium target plates is shown in Fig.53. Solid beryllium tiles are clamped onto water cooled stainless steel support beams. The front face of the tiles is castellated to reduce the intensity of thermal flux and limit the propagation of surface cracks. Preliminary stress analysis indicated that the size of castellations and the type of root of the castellation play a large role in the fatigue life of a solid target plate. In fact, for finer castellations, higher power densities were tolerated and no fatigue cracking was observed. A final series of tests is planned for 1992 to study the effect of start temperature on the fatigue life of Mark I solid beryllium tiles. In the final design, it is planned to use fine castellations.

The divertor target plates carry 7296 beryllium tiles. The tiles are attached in pairs to the beams and the pairs have been designed to be fully compatible with remote handling requirements. The surface of the tiles is saw-toothed, each tile surface being at an angle of 5° to the horizontal. This design ensures that the tile edges are shadowed as well as making allowances for possible misalignments in the system due to installation and manufacturing tolerances, as well as those caused by thermal displacements.

For the water cooled target plates (Mark II), all the copper-chromium-zirconium alloy (CuCrZr) for the target plates has been procured, and machining of the hypervapotron internal structure is well advanced. Final machining of the surface interfacing with beryllium tiles will be performed with the target plates attached to their rigid stainless steel support structures. This approach enables the overall dimensional accuracy of the divertor to be maintained within acceptable limits. The design of the assembly of individual target plates into modules has been completed.

During 1991, investigations have continued into bonding of beryllium to CuCrZr with the aim of achieving routine high strength bonding. Efforts have concentrated on minimising the build-up of brittle intermetallics at the beryllium-braze interface and improving the bond to beryllium surfaces. These have been demonstrated to be limiting factors in achieving adequate bond strength by operational tests of prototype elements and mechanical tests on small samples.

ICRF and LHCD System Developments

During 1991, the design of the A2 ICRF antennae has been largely completed. Many tests have been carried out to qualify manufacturing procedures, notably on the corrugated plates forming the side and back panels of the housing and the septum which forms the main structural element of the antenna. Manufacture is proceeding and the first until will be at JET in mid-1992. All beryllium screen elements and the mounting assemblies for these antennae have been delivered. Manufacture of four new vacuum transmission lines is well advanced and a contract has been placed for modification of the existing lines after removal from the torus.

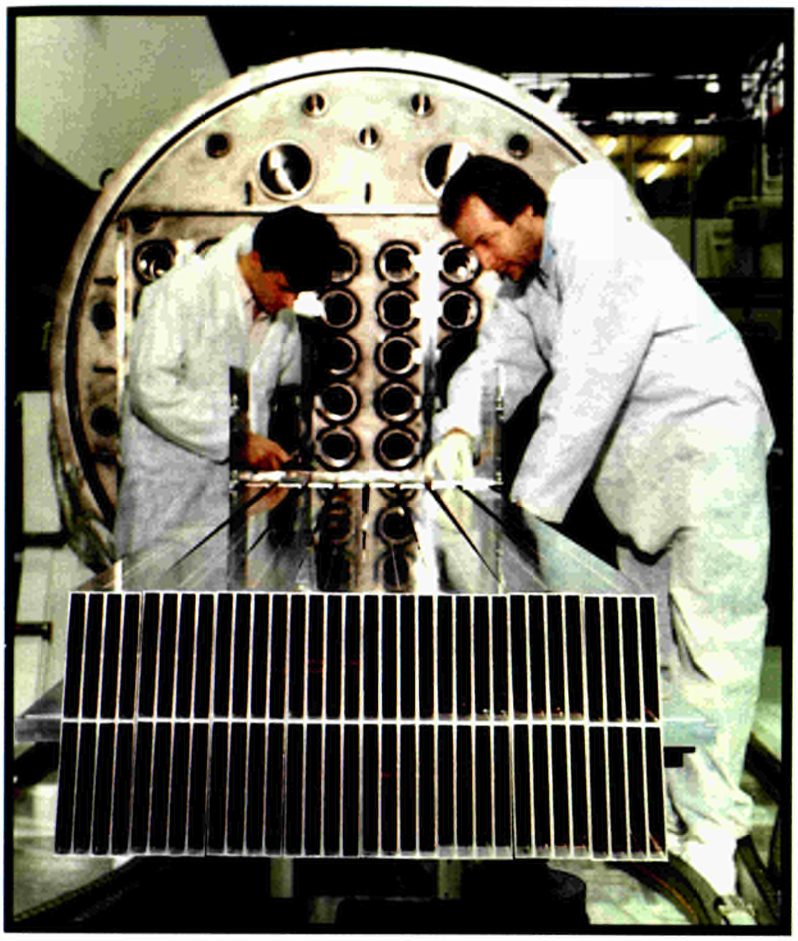


Fig.54 View of the LHCD Launcher (L1)

The antennae rely on distributed support from the torus to service disruptions. The design of these supports, which must allow thermal expansion, is also complete. With four of the antennae relocated in the torus, the corresponding main transmission lines need to be modified. The design of these is complete and the contract placed. High power tests on the model antenna have been largely completed. After refinement of the design, following initial tests, the full required performance (45kV for 20s) has been achieved.

The Lower Hybrid Current Drive (LHCD) launcher must be located close to the plasma to achieve good matching. The design of the LHCD launcher (L1) has been modified to position the grill mouth at the large plasma boundary and to match the profile of the adjacent A2 Ion Cyclotron Radio Frequency antenna. It is not possible to reach the small plasma boundary due to the tight fit of the launcher in the port. These changes have required some minor re-working of the external dimensions of the multijunctions, which is now complete. Preparation of the multijunction/ waveguide assemblies is well advanced and assembly of the launcher is proceeding. A view of the launcher grill mouth is shown in Fig.54.

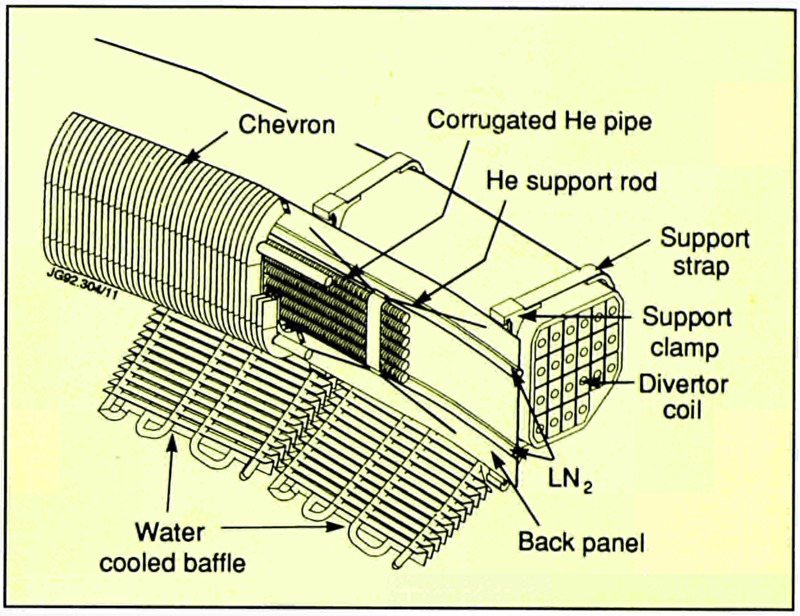


Fig.55: Divertor cryopump

The side protection and its support also require modification to match the new profile. The design work is complete and modification of the components will start after removal from the torus and decontamination. New remote handling tools are being prepared for these items. About 50 microwave windows have been received and each tested to full power. Waveguides for use in the launcher have also been tested.

Poloidal Limiters

The existing belt limiters will be replaced by a series of poloidal limiters, of which the main task will be to protect the new antennae. A belt limiter system cannot protect the antennae for all possible plasma configurations. These poloidal limiters have been designed to use the tiles, either beryllium or carbon, which have been used for the belt limiter.

The tiles will be reshaped and installed horizontally into the structure of the poloidal limiters. The eddy currents induced, in particular when beryllium tiles are inserted, will be substantially higher than in the present limiter configuration. Additional high forces can be expected from halo currents. The result is that strong supports are required which because they span the bellows, must be insulated. Prototypes of insulated pins have been successfully tested.

Additional requirements for the poloidal limiter system met in the design are: the ability to move the limiters radially to a different position when operating with slim plasmas and antennae that have been moved (which requires the lower section of the limiters to be hinged and moveable); the capability of finely adjusting their position after installation to ensure precise positional accuracy of the limiters in the radial direction, relative to the antennae and the

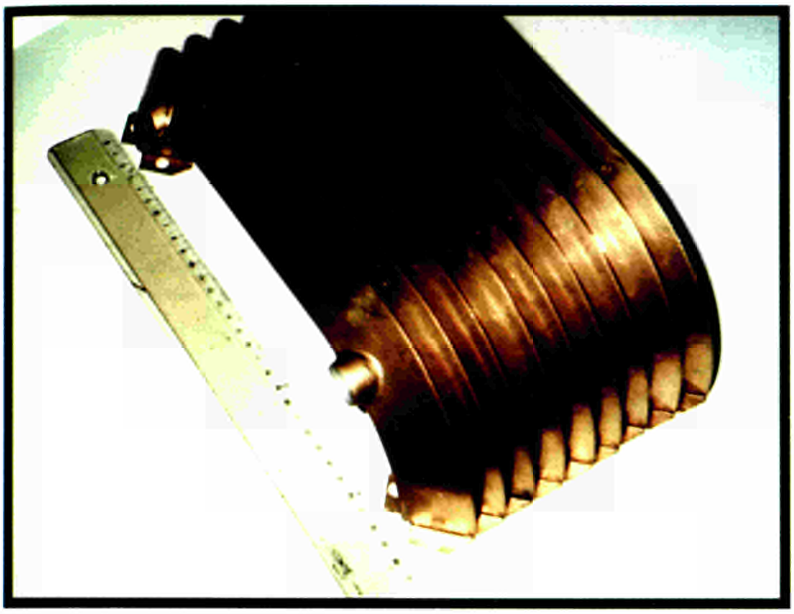


Fig.56: Cryopump chevron test assembly

magnetic centre of the machine; to accommodate the needs of numerous diagnostics, which require openings, holes and cut-outs in the limiter structures, as well as to allow the remote handling boom to enter the vessel unhindered. The latter requirements necessitated the design of two basic widths for the limiters and additional individual variations between them. Careful consideration has been made in the design relating to halo currents and grounding cables have been included at strategic locations.

Cryopump and Water Cooled Baffles

An integral part of the divertor is a large cryocondensation pump with a nominal pumping speed of $5x10^5 \ell s^{-1}$. The pump shown schematically in Fig.55 extends over the full toroidal length of the outermost divertor coil which also acts as the mechanical support. Further progress has been made with the procurement of the cryopump and water cooled baffle. In particular, a series of major technical problems to cope with the harsh environment for this equipment needed solutions. A suitable copper alloy for the chevron structure was required to provide the required mechanical strength necessary to cope with the extreme forces caused by eddy currents. At the same time, the material needed good thermal conductivity and also high electrical resistivity. All these requirements are basically contradictory and only by using a precipitation hardening copper alloy and by optimising the precipitation hardening process could a satisfactory solution be found.

A special blackening process for the chevrons was developed and qualified by using a plasma sprayed coat of Al_2O_3 and TiO_2. The main achievement was to find a suitable intermediate coat in order to

Table.8: Diagnostics for the New Phase of JET

System	Diagnostic	Function	Status
KE9D	LIDAR Thomson scattering	T_e and n_e profiles in divertor plasma	In-vessel design complete, procurement in progress.
KG6D	Microwave interferometer	$\int n_e dl$ along many chords in divertor plasma	In-vessel waveguide design complete, procurement in progress
KG7D	Microwave reflectometer	Peak n_e along many chords in divertor plasma	In-vessel waveguide design complete, ex-vessel microwave design and mockup experiments in progress
KK4D	Electron cyclotron absorption	$n_e T_e$ profile along many chords in divertor plasma	In-vessel waveguide design complete, ex-vessel microwave design and mockup experiments in progress
KD1D	Calorimetry of Mark1 divertor targets	Power balance of divertor plasma	Thermocouple installation awaiting delivery of target tiles
KC1D	Magnetic pickup coils	Plasma geometry in divertor region	Manufacture in progress
KY4D	Langmuir probes in divertor target tiles	n_e and T_e in the divertor plasma	Design completed
KY5D	Fast pressure gauges	Neutral flow in divertor region	Manufacture in progress
KT6D	Poloidal view visible spectroscopy of divertor plasma using a periscope	Impurity influx, 2-D emissivity profile of lines	Periscope and in-vessel components in manufacture. Design of other components and optics in progress
KT5D	Toroidal view visible spectroscopy of divertor plasma from Octant No:7 mid-plane	T_z and V_z, ion temperature and toroidal velocity of impurities	Design in progress. Optics components defined and procurement in progress
KT7D	VUV and XUV spectroscopy of divertor plasma	Impurity influx, ionization dynamics	Spectrometer in manufacture, mechanical design in hand, procurement of electronics and data acquisition in hand
KB3D	Bolometry of divertor region	Power balance of divertor plasma	Design of mechanical interface being finalized. Final tests of detector element in progress. Procurement of electronics and data acquisition in hand.
KY6	50kV lithium atom beam	Parameters of the scrape-off-layer plasma	Concept approved, detailed design in progress

make the coat compatible with both copper and the high temperature cycle for the combined precipitation hardening and brazing process of the chevrons. The first manufacturing prototypes for the cold formed chevrons have been successfully produced and assembled for brazing trials (see Fig.56).

Divertor Diagnostics

Development of divertor diagnostics is well advanced. Design integration and procurement of hardware with interfaces in the divertor region have received priority as these will be required first for installation. Magnetic geometry, electron density and temperature of the divertor plasma, radiative and conductive heat loading of the targets, and dynamics of impurity and hydrogen ions in the divertor plasmas are the main measurement goals. Current status in respect of design, procurement and installation is given in Table.8.

Assembly of the Pumped Divertor Configuration

The shutdown for installation of components relevant to the Pumped Divertor configuration will start in March 1992 and progress to Summer 1993. A new group, First Wall Installation

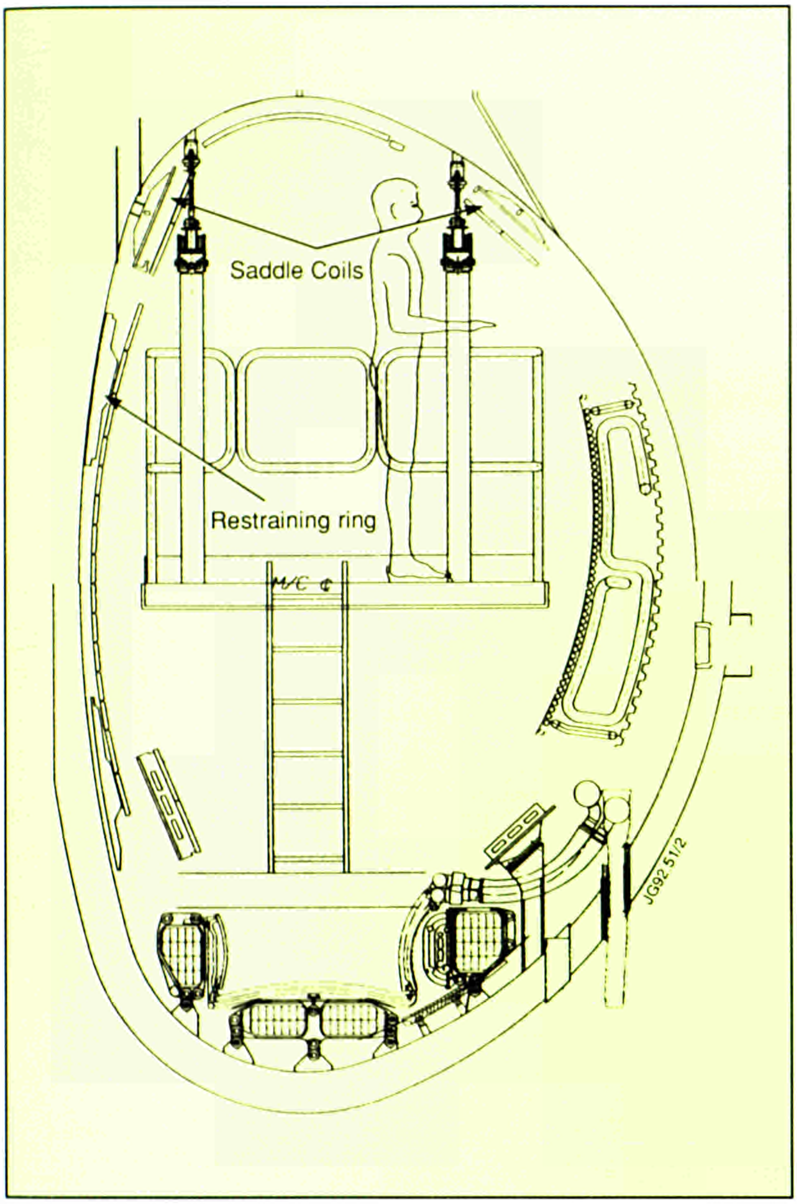
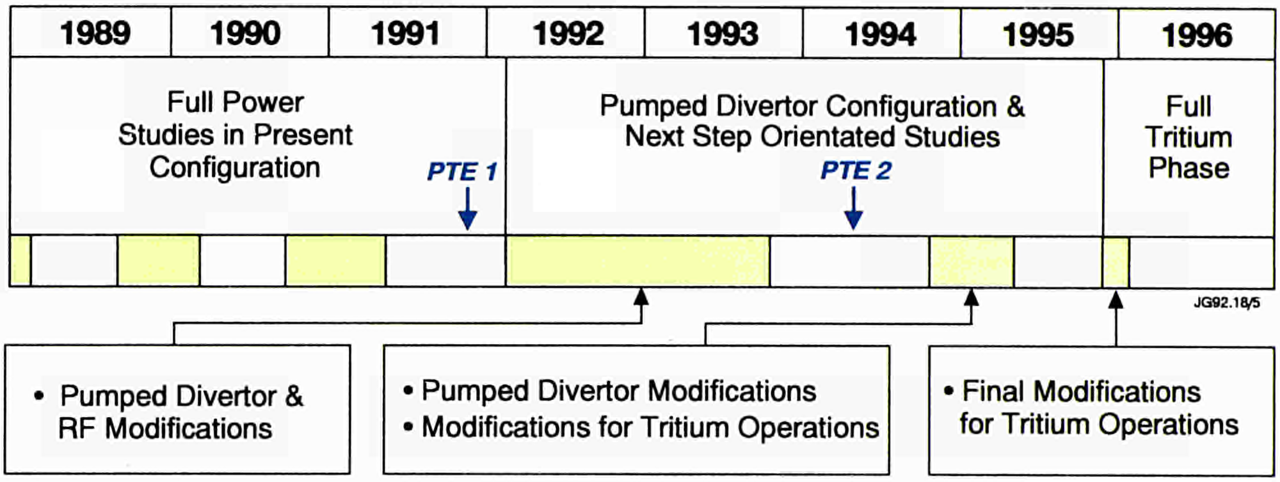


Fig.57: Divertor assembly arrangement

Group has been set up with the task of preparing the pumped divertor installation. To minimise the installation time and to optimise the working conditions inside the vessel (beryllium, tritium radiation and limited space) special procedures, jigs and equipment have been developed and suitable working methods and organisation has been set up.

Handling of components, through the ports and inside the vessel, has required a great deal of effort. In addition to the articulated boom and the TARM a special design toroidal travelling crane will be installed. A series of permanent bosses will be welded to the roof of the vessel. These will support the temporary lifting features (arched beams) required during the manufacturing of the divertor coils and subsequently a toroidal travelling crane and a mobile platform for installation of the other components (see Fig.57).

Table.9: JET Programme to 1996



PTE: **Preliminary Tritium Experiment**

The geometry and construction of the vacuum vessel does not lend itself to the installation of the divertor system to the level of accuracy required. Therefore special fixtures and jigs will be used to set the position of each component as closely as possible to the magnetic configuration of the machine using external datum. All inner wall components of the vacuum vessel will be stripped at the earliest stage possible in order to achieve decontamination of the vessel and, therefore, to require only minimal personal protection for the installation of the divertor. This will enhance the number of people working in the vessel to improve efficiency.

Future Plans

The JET Programme is divided into phases governed by the availability of new equipment and fitting within the accepted life time of the Project. The programme to 1996 is shown in Table.9. Phase I (Ohmic Heating Studies) was completed in September 1984, and Phase II (Additional Heating Studies) in October 1988. Phase III (Full Power Optimization Studies) ends in February 1992. The scientific aims of Phase III were to obtain maximum performance in (belt) limiter configuration (plasma currents up to 7MA) and to optimise X-point Operation (plasma currents up to 6MA) including a comparison of H-modes in X-point configuration using beryllium (lower X-point) with carbon (upper X-point) dump plates.

JET future plans are dominated by the insertion of a new phase of the Project (Phase IV Pumped Divertor Configuration and Next Step Oriented Studies), before the final full Tritium Phase, which is now scheduled to start in December 1995. This new phase has now been formally approved and has extended the lifetime of the Project by four years up to the end of 1996.

The aim of the new phase is to demonstrate, prior to the introduction of tritium, effective methods of impurity control in operational conditions close to those of a Next-Step Tokamak with a stationary plasma (10s-1m) of 'thermonuclear grade' in a single-null axisymmetric pumped divertor configuration. This configuration can only be achieved in JET by using divertor coils internal to the vessel.

Following the approval by the JET Council of the step-wise approach to the introduction of tritium in advance of the full Tritium Phase, a first preliminary tritium experiment (PTX1) was carried out and successfully completed in November 1991. A second tritium experiment (PTX2) is scheduled for the first half of 1994 at a point, yet to be determined, when divertor operation has been well established, but in time to allow the necessary period of radioactive decay before the following shutdown. The information derived from these preliminary tritium experiments will provide a safer approach to the full tritium phase and will help to optimise the active handling and waste management arrangements.

Further information on the Project's future phases are provided in the following paragraphs.

New Phase (first part): Pumped Divertor Configuration and Next Step Oriented Studies - Phase IVA (March 1992 - October 1994)

In March 1992, the Project will enter an extended shutdown, which will last until Summer 1993, in order to install the components relevant to the new pumped divertor phase. This will involve intensive in-vessel work to install the following equipment:

- lower divertor structure with Mark 1 beryllium target plates (inertially cooled Be blocks)
- pumping chamber and cryopump;
- internal divertor coils and necessary power supplies;
- poloidal limiters;
- new ICRF heating antennae (A2);
- full lower hybrid current drive (LHCD) system with modified launcher;
- divertor diagnostics;
- high-speed pellet launcher (for plasma core injection);
- centrifuge pellet launcher (for plasma edge injection);
- disruption control system using internal saddle coils.

The single-null X-point pumped divertor configuration should enable JET to progress towards extended high power operation with 40MW additional heating using neutral beam and ICRF power (e.g. plasma currents of 6MA for up to 3s, 3MA for up to 5s). The control of disruptions using saddle coils system and the control of sawteeth using the full power LHCD systems should also be studied.

The first operating period should focus initially on establishing reliable operation in this new configuration. Subsequently, attention should be devoted to the study of performance and effects of the pumped divertor in controlling impurities, plasma density and exhaust, and power loading on the target plates.

Preparations for D-T operations will also continue during this period, including finalisation of Remote Handling tests and commissioning of the Active Gas Handling System (AGHS) with tritium gas (subject to consent by the approving bodies).

New Phase (Second Part): Pumped Divertor Configuration and Next Step Oriented Studies - Phase IVB (Nov. 1994 - Nov. 1995)

The proposed shutdown (~6 months) in late 1994 would be used to replace the Mark 1 Be target plates by the Mark 2 version (water cooled Be clad hypervapotrons) and to provide an opportunity, depending on information from the experimental programme and elsewhere, to implement other modifications to the pumped divertor. In addition, it should be possible to install other enhancements aimed at improving performance in the new configuration or for the D-T Phase (eg. enhanced pellet injection or fuelling system, modifications to LHCD or additional heating systems, etc).

The primary objective of this period of the new phase (to end 1995) would be to provide information needed to demonstrate steady divertor operation with 40MW power and to establish with confidence the key design features of the Next Step in relation to:

- impurity control;
- fuelling;
- helium transport and exhaust of ashes.

Another objective would be to optimise reliability and plasma performance in the divertor configuration in preparation for D-T operations.

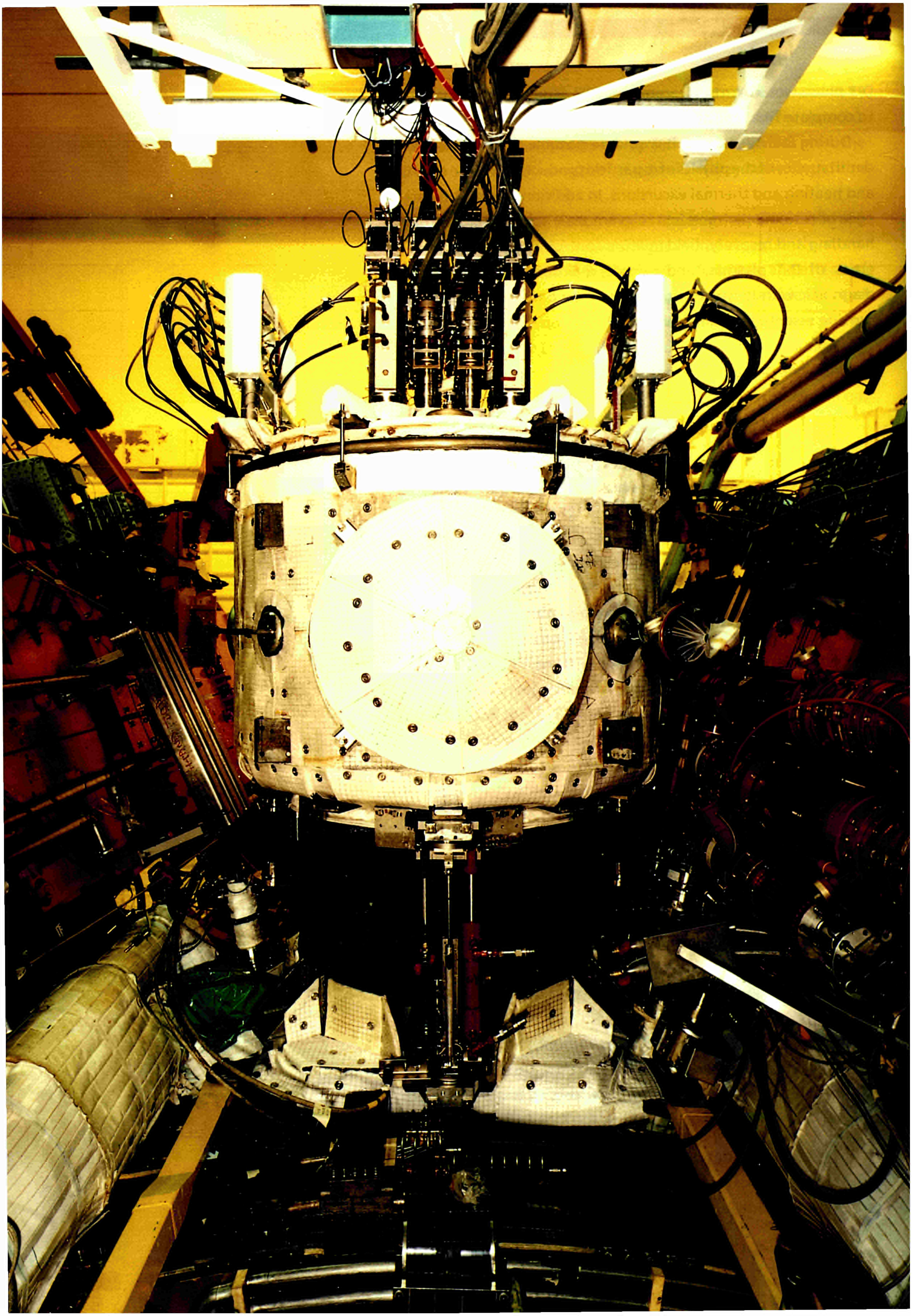
In parallel, the tritium commissioning of the Active Gas Handling System would be completed and the main tritium modifications to the machine and its associated systems (including remote maintenance and waste handling) would be implemented.

Full Tritium Operation - Phase V (Dec. 1995 - Dec. 1996)

Subject to the approval of the JET Council and to necessary official consents and when general levels of system reliability justify it, the

D-T phase would start in December 1995, after a short shutdown to complete final adjustments required for active operations.

During D-T operations, it would be possible to undertake in-depth studies of the physics of α-particle production, confinement and heating and thermal excursions. In addition, the real experience of tritium operation in a relevant scale tokamak (ie. tritium handling and recovery, field mixture control, confinement properties of D-T plasmas, and remote maintenance and plasma diagnostic with large neutron and gamma backgrounds) should provide essential information for a Next Step device.



Members and Organisation

Members

The JET Joint Undertaking has the following Members:

The European Atomic Energy Community (EURATOM);

The Belgian State, acting for its own part ('Laboratoire de Physique des Plasmas of the École Royale Militaire') and on behalf of the Université Libre de Bruxelles' ('Service de Chimie-Physique II de l'ULB'); and of the 'Centre d'Études de l'Energie Nucléaire (CEN)/'Studiecentrum voor Kernergie' (SCK);

The Centro de Investigaciones Energéticas Medioambientales y Tecnológicas (CIEMAT), Spain;

The Commissariat à l'Énergie Atomique (CEA), France;

The Comitato Nazionale per la Ricerca e per lo Sviluppo dell'Energia Nucleare e delle Energie Alternative (ENEA), Italy;

The Hellenic Republic, Greece;

The Forskningscenter Risø (Risø), Denmark;

The Grand Duchy of Luxembourg, Luxembourg;

The Junta Nacional de Investigaçao Cientifica e Tecnológica (JNICT), Portugal;

Ireland;

The Kernforschungsanlage Jülich GmbH (KFA), Federal Republic of Germany;

The Max-Planck-Gesellschaft zur Förderung der Wissenschaften eV - Institut für Plasmaphysik (IPP), Federal Republic of Germany;

The Swedish Natural Science Research Council (NFR), Sweden;

The Swiss Confederation, Switzerland;

The Stichting voor Fundamenteel Onderzoek der Materie (FOM), The Netherlands;

The United Kingdom Atomic Energy Authority (UKAEA), Host Organisation.

Management

The JET Joint Undertaking is governed by Statutes which were adopted by the Council of the European Communities on 30 May

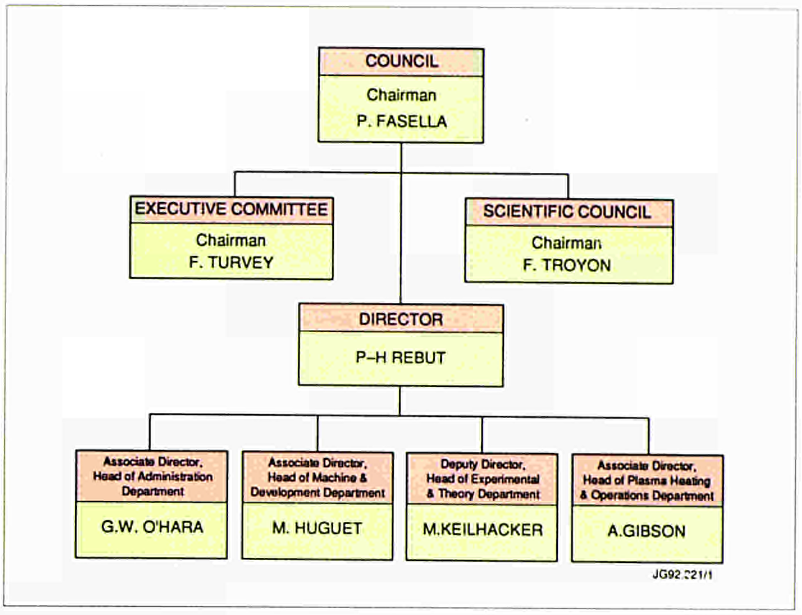


Fig. 58: Overall Project Structure

1978. The organs of the Joint Undertaking are the JET Council and the Director of the Project. The JET Council is assisted by the JET Executive Committee and is advised by the JET Scientific Council (see Fig.58).

JET Council

Each member of the Joint Undertaking is represented on the JET Council, which is required to meet at least twice yearly. The Council is responsible for the management of the Joint Undertaking and for:

- Nomination of the Director and Senior Staff of the Project with a view to their appointment by the Commission or the Host Organisation as appropriate;
- Approval of the annual budget, including staffing, as well as the Project Development Plan and the Project Cost Estimates;
- Ensuring collaboration between the Associated Laboratories and the Joint Undertaking in the execution of the Project, including the establishment of rules on the operation and exploitation of JET.

Four meetings of the JET Council were held during the year on 14th-15th March, 12th-13th June, 26th September and 17th-18th October1991. The membership of the JET Council is shown in Appendix I.

JET Executive Committee

The JET Executive Committee is required to meet at least six times a year. Its functions include:

- Advising the JET Council and the Director of the Project on the status of the Project on the basis of regular reports;
- Commenting and making recommendations to the JET Council on the Project Cost Estimates and the Draft Budget, including the establishment of staff, drawn up by the Director of the Project;
- Approving, in accordance with the rules on the award of contracts established by the JET Council, the tendering procedure and the award of contracts;
- Promoting and developing collaboration between the Associated Laboratories and the Joint Undertaking in the execution of the Project.

The membership of the JET Executive Committee is shown in Appendix II. The Committee met six times during the year on 21st-22nd February, 18th-19th April, 4th-5th July, 12th-13th September, 14th November and 12th December 1991.

JET Scientific Council

The JET Statutes confer the following functions on the JET Scientific Council:

- Upon the request of the JET Council, to advise on scientific and technical matters, including proposals involving a significant change in the design of JET, its exploitation, and its long-term scientific implications;
- To perform such other tasks as the JET Council may request it to undertake.

The membership of the JET Scientific Council is shown in Appendix III. The Scientific Council met three times during the year on 26th February, 22nd-23rd May and 26th-27th November.

The main work of the JET Scientific Council in 1991 was to assess and advise the JET Council on:

- The JET experimental programme and planning
- The technical status of the pumped divertor
- Diagnostics for the pumped divertor;
- The Preliminary Tritium Experiment;

In addition, it reviewed the results of the 1991 campaign and discussed the experimental programme to February 1992.

Host Organisation

The United Kingdom Atomic Energy Authority, as the Host Organisation for the JET Joint Undertaking, has made available to the Joint Undertaking, the land, buildings, goods and services required for the implementation of the Project. The details of such support, as well as the procedures for co-operation between the Joint Undertaking and the Host Organisation, are covered by a 'Support Agreement' between both parties. In addition to providing staff to the JET team, the Host Organisation provides support staff and services, at proven cost, to meet the requirements of the JET Project.

Project Team Structure

The Director of the Project

The Director of the Project, Dr. P-H. Rebut, is the chief executive of the Joint Undertaking and its legal representative. He is responsible to the JET Council for the execution of the Project Development Plan, which specifies the programme, and for the execution of all elements of the Project. The Project Development Plan covers the whole term of the Joint Undertaking and is regularly updated. The Director is also required to provide the JET Scientific Council and other subsidiary bodies with all information necessary for the performance of their functions.

Internal Organisation

The internal organisation of the Project consists of four Departments and the Coordinating Staff Unit. The four Departments are:

- Plasma Heating and Operation Department;
- Experimental and Theory Department;
- Machine and Development Department;
- Administration Department.

The overall Project Structure is shown in Fig.58.

Directorate

The Heads of the Departments report to the Director of the Project and together with the Director form the JET Directorate. Various special functions are carried out by the Director's Office. The Internal Audit Office monitors the financial activities and provides advice on accounting and control procedures as well as maintaining links with the Court of Auditors. The Project Control Office is responsible for financial planning and for the preparation of the Project Development Plan and Project Cost Estimates. The JET Council Secretariat provides Secretarial Services to the JET Council and to the Executive Committee and also to the JET Project Board.

In addition, there are three groups. One contains Scientific Assistants who assist and advise the Director on scientific aspects of JET operation and future development. Another group contains the Technical Assistant who assists and advises the Director on organisational and technical matters related to JET operation and who also acts as Leader of the Publications Group. A third section contains the Press and Public Relations Group.

Plasma Heating and Operation Department

The Plasma Heating and Operation Department is responsible for heating the plasma, the organisation of experimental data and the day-to-day operation of the machine. The main functions of the Department are:

- heating of the plasma and analysis of its effects;
- centralising the interpretation of experimental results and investigating their coherence;
- organising data acquisition and computers;
- preparing and co-ordinating operation of the machine across the different Departments.

The Department is composed of four groups (Machine Operations Group, Physics Operations Group, Data Management Group and Health Physics Group) and three Divisions:

(1) Control and Data Acquisition System Division (CODAS), which is responsible for the implementation, upgrading and operation of the computer-based control and data acquisition systems;

(2) Neutral Beam Heating Division, which is responsible for the operation of the neutral beam injection system. The Division also participates in studies of the physics of neutral beam heating;

(3) Radio Frequency Heating Division, which is responsible for the design, construction, commissioning and operating the RF heating and Lower Hybrid (LH) systems during the different stages of their development to full power. The Division also participates in studies of the physics of RF heating and Lower Hybrid Current Drive.

Experimental and Theory Department

The main functions of the Department relate to the measurement and validation of plasma parameters and the theory of tokamak physics. The major tasks are:

- to conceive and define a set of coherent measurements;
- to be responsible for the construction of necessary diagnostics;
- to be responsible for the operation of the diagnostics, the quality of measurements and the definition of the plasma parameters;
- to play a major role in the interpretation of data;
- to follow the theory of tokamak physics.

The Department consists of two Groups (Diagnostics Engineering Group and Data Processing and Analysis Group) and three Divisions:

(1) Experimental Division One (ED1), which is responsible for specification, procurement and operation of approximately

half of the diagnostic systems. ED1 undertakes electrical measurement, electron temperature measurements, surface and limiter physics and neutron diagnostics;

(2) Experimental Division Two (ED2), which is responsible for specification, procurement and operation of the other half of the diagnostic systems. ED2 undertakes all spectroscopic diagnostics, bolometry, interferometry, the soft X-ray array and neutral particle analysis;

(3) Theory Division, which is responsible for prediction of JET performance by computer simulation, interpretation of JET data and application of analytic plasma theory to gain an understanding of JET physics.

Machine and Development Department

The Machine and Development Department is responsible for the performance capability of the machine as well as for equipment for the active phase, together with enhancements directly related to it (excluding heating) and the integration of any new elements on to the machine. In addition, the Department, which is composed of three Divisions, is responsible for maintenance and operation of the coil systems, structural components and machine instrumentation. The three Divisions are:

(1) Magnet and Power Supplies Division is responsible for the design, construction, installation, operation and maintenance of the tokamak electromagnetic system and of plasma control. The area of responsibility encompasses the toroidal, poloidal and divertor magnets, mechanical structure, methods for controlling plasma position and shape and all power supply equipment needed for magnets, plasma control, additional heating and auxiliaries;

(2) First Wall Division, which is responsible for the vital area of plasma-wall interactions. Its main tasks include the provision and maintenance inside the vacuum vessel of conditions leading to high quality plasma discharges. The Division develops, designs, procures and installs first wall systems and components, such as limiters, wall protections, internal pumping devices and pellet injection systems. The area of responsibility encompasses the vacuum vessel as a whole, together with its associated systems, such as pumping, bakeout and gas introduction;

(3) Fusion Technology Division, which is responsible for the design and development of remote handling methods and tools to cope with the requirements of the JET device, and for maintenance, inspection and repairs. Tasks also include the design and construction of facilities for handling tritium.

Administration Department

The Administration Department is responsible for providing Contracts, Finance and Personnel services to the Project.

Coordinating Staff Unit

The Coordinating Staff Unit is responsible for the provision of engineering services to the whole project and for the implementation of specific coordinating tasks at the Project level.

It comprises five Groups:

- Technical Services Group;
- Safety Group;
- Drawing Office;
- Quality Group;
- Health Physics and Safety Group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Administration

Introduction

The three main aspects of JET's administration - Finance, Contracts and Personnel - are reported on in this section. In addition, there are contributions on Safety, Public Relations, and Publications Group.

Finance

The initial budgets for 1991 were approved at 97.57 MioECU for Commitments and 102.63 MioECU for both Income and Payments. The Commitments and Payments Budgets each are divided into two phases of the Project - the Extension to Full Performance and the Operational Phase; subdivisions distinguish between investment, operating, and personnel costs, each with further detailed cost codes.

The Commitments Budget included 2.00 MioECU and the Payments and Income Budget included 1.50 MioECU for specific work for the European Fusion Programme.

During the year, the Project continued its efforts to achieve economies, particularly in operating and staff expenditure, and will pursue this policy in coming years to ensure successful completion of its programme. However, the scope of the programme might be restricted due to the imposed budgetary cuts.

Commitments

Of the total appropriations in 1991 of 129.20 MioECU (including 31.63 Mio ECU brought forward from previous years), 110.35 Mio ECU was committed and the balance of 18.85 MioECU was available for carrying forward to 1992. The details of the commitment appropriations available (Table.10) and of the amounts committed in each Phase during the year (Table.11) are summarised as follows:

- In the extension to Full Performance Phase 5.58 MioECU was committed leaving 7.55 MioECU commitment appropriations not utilised at 31 December 1991, to be carried forward to 1992.

Table.10: Commitment Appropriations for 1991

	MioECU
Initial Commitments Budget for 1991	97.57
Amounts brought forward from previous years	31.63
	129.20
Commitments made during the year	110.35
Balance of appropriations at 31 December 1991 available for use in 1992	18.85

Table.II: Commitments and Payments for 1991

	Commitments		Payments	
Budget Heading	Budget Appro-priations MioECU	Outturn MioECU	Budget Appro-priations MioECU	Outturn MioECU
Phase 2 Extension to Full performance				
Title 1 Project Investments	13.13	5.58	9.95	6.83
Phase 3 Operational				
Title 1 Project Investments	19.55	16.47	11.54	8.23
Title 2 Operating Costs	39.66	34.12	36.42	35.31
Title 3 Personnel Costs	56.86	54.18	58.91	46.42
Total Phase 3	116.07	104.77	106.87	89.96
Project Total - all phases	129.20	110.35	116.82	96.79

- In the Operational Phase 104.77 MioECU was committed leaving a balance of 11.30 MioECU to be carried forward to 1992.

Income and Payments

The actual income for 1991 was 99.27 MioECU to which was added 2.35 Mio ECU available appropriations brought forward from previous years giving a total of 101.62 MioECU. This total includes 0.05 MioECU arising from Specific Fusion Research work against a budget of 1.50 MioECU. The shortfall in income of 1.45 MioECU was offset by a corresponding reduction in the Payments Budget. The excess of

Table.12: Income and Payments for 1991

	MioECU	
Income		
Budget for 1991		102.63
Income received during 1991		
(i) Members' Contributions	96.28	
(ii) Bank Interest	2.91	
(iii) Miscellaneous	0.03	
(iv) Unused Appropriations brought forward from 1989	2.35	
(v) Income for Specific Fusion Research	0.05	
Total Income		101.62
Income in excess of budget carried foward for off-set against Members' future contributions	0.44	
Shortfall of Income for Specific Fusion Research reducing available payment appropriations	(1.45)	
Payments		
Budget for 1991	102.63	
Amounts available in the Special Account to meet outstanding commitments at 31 December 1990	14.18	
Reductions in appropriations corresponding to shortfall of income for Specific Fusion Research	(1.45)	
Total Available Appropriations for 1991		115.36
Actual payments during 1991	96.79	
From Special Account transferred to income	0.84	
Unutilised appropriations at 31 December 1991		97.63
carried foward in the Special Account to meet outstanding commitments at that date		17.73

other income over budget totalling 0.44 MioECU is carried forward to be offset against future contributions of Members. The total payment appropriations for 1991 of 116.81 MioECU were reduced by 1.45 Mio ECU due to the shortfall in income from Specific Fusion Research work; the payments in the year amounted to 96.79 MioECU and 0.84 MioECU was transferred from the Special Account to income. The balance of 17.73 MioECU was transferred to the Special Reserve Account to meet commitments outstanding at 31 December 1991. (Payments are summarised in Tables 11 and 12).

Contributions from Members

The budget for Members' contributions was 96.28 MioECU funded as follows:

Table.13: Percentage Contributions to JET for 1991

(based on the Euratom Participation in Associations' Contracts for 1990)

Member	%	MioECU
Euratom	80.0000	77.02
Belgium	0.1974	0.19
CIEMAT, Spain	0.1725	0.17
CEA, France	1.9389	1.87
ENEA, Italy	1.9930	1.92
Risø, Denmark	0.0849	0.08
Luxembourg	0.0016	0.00
JNICT	0.0354	0.03
KFA, Germany	0.6821	0.66
IPP, Germany	2.1707	2.09
KfK, Germany	0.8770	0.85
NFR, Sweden	0.2009	0.19
Switzerland	0.4814	0.46
FOM, Netherlands	0.3951	0.38
UKAEA	10.7691	10.37
	100.0000	96.28

- 80% from the general budget of the European Atomic Energy Community (Euratom);
- 10% from the UK Atomic Energy Authority as Host Organisation;
- 10% from members who have Contracts of Association with Euratom in proportion to the previous year's contribution from Euratom towards the cost of their Association Contracts.

Table.13 gives contributions from Members for 1991.

Bank Interest

During the year, funds are normally received on a quarterly basis in respect of Members' contributions and intermittently in respect of other items. Therefore, the Project has funds which are not immediately required for the discharge of its commitments; these funds

Table.14: Summary of Financial Transactions at 31 December 1991

	MioECU
Cumulative commitments	1227.6
Cumulative payments	1171.0
Unpaid commitments	56.6
Amount carried forward in the Special Account	17.7
Amount available from 1990 and 1991 for set off against future contributions from Members	4.2

are placed on deposit accounts at market interest rates. During 1991, earned interest amounted to 2.91 MioECU.

Appropriations from Earlier Years

The unused payment appropriations and excess income over budget of 2.35 MioECU arising in 1989 and held for reduction of Members' future contributions were transferred to income in 1991.

Summary

Table.14 summarises the financial transactions of the JET Joint Undertaking as at 31 December 1991, which have yet to be audited. The final audited accounts will be published in due course.

Contracts Service

Contracts Activity

164 tender actions covering supply, service and personnel requirements were issued in 1991 and 15,656 contracts were placed. Of the 95 major contracts (value > 75,000 ECU) placed, a significant proportion were for pumped divertor related plant and for services. 3,880 minor contracts (value between 500 and 75,000 ECU) were issued in 1991. 11,674 Direct Orders (value between 0 and 500 ECU) were issued in the year which amounted to 75% of the total orders placed whilst their aggregate value amounted to only 3.1% of the total value of all orders in that period.

Many of the larger contracts involve advance and retention payments for which bank guarantees are required by JET. The total value of guarantees held as at 31 December 1991 was 7.8 MioECU.

Imports and Exports Services

Contracts Service is also responsible for the import and export of JET goods. 1,310 imports were handled in 1991 while the total exports amounted to 279. There were also 1,046 issues of goods to UK firms. The total value of issues to all countries for the year was 8.01 MioECU.

Stores Organisation

The bulk of JET material is procured on a "just in time" basis and the Stores Organisation provides a receipts and delivery service for this material to the Project. The total number of such receipts in 1991 amounted to 16,694.

Administration of Contracts

The distribution of contracts between countries is shown in Tables 15 and 16. Table.15 includes all contracts with a value of 10,000 ECU and above placed prior to 1984, together with all contracts placed during the period 1984-91. Table.16 is an allocation of "high-tech" contracts, which is based on the figures shown in Table.15 but excludes

Table.15: Allocation of JET Contracts

Country	Total of kECU Values	% of Total
UK	445,327	53.39
Germany	151,913	18.22
France	77,595	9.31
Italy	52,137	6.25
Switzerland	40,855	4.90
Denmark	12,477	1.50
Netherlands	15,628	1.87
Belgium	10,139	1.22
Sweden	6,258	0.75
Ireland	394	0.05
Others	21,143	2.54
TOTALS	833,866	100.00

Table.16: Allocation of JET "High-Tech" Contracts

Country	Total of kECU Values	% of Total
UK	132,332	28.29
Germany	138,233	29.54
France	68,394	14.61
Italy	47,730	10.20
Switzerland	32,961	7.04
Denmark	7,409	1.58
Netherlands	14,705	3.14
Belgium	4,869	1.04
Sweden	4,229	0.90
Ireland	330	0.07
Others	16,780	3.59
TOTALS	467,972	100.00

all contracts below 5,000 ECU and contracts concerning civil works, installation, pipework, consumables (includes gases), maintenance operations and office equipment (including PCs).

Personnel Service

During 1991, the work of the JET Personnel Service was again concentrated in three main areas of recruitment and staffing, training and conditions of service, and general administration.

Recruitment and Staffing

During the first half of 1991, the policy of recruiting young scientists and engineers was maintained. These are personnel who are able to make a positive contribution to the work of the Project up to the end of 1996 and capable of taking a leading role in international fusion research after JET. Later in the year, this policy had to be curtailed as the Project came close to achieving its full complement of staff for the first time in the history of JET, and due to budgetary reductions which were imposed by the JET Council in order to meet the financial constraints set by the Commission. As a result, no further recruitment action was taken except where it was considered to be essential for the Project's immediate programme. Nevertheless, the links which had been established last year with European academic institutions to attract students and JET Fellows were maintained and opportunities for them to work at JET continued to be available during the year.

Team Staff

The recruitment campaign at the start of the year resulted in 27.5 vacant team posts being filled, but after allowing for staff departures, the net increase in the number of team staff was 5.5 (2 Euratom and 3.5 UKAEA). The number of posts filled against the official complement is shown in Table.17. Figure 59 shows the composition of team staff by nationality at 31 December 1991.

Three JET Fellows were also recruited at the end of the their studies in line with the policy of recruiting young scientists to the Project.

Students

87 student appointments were made during 1991, which represented a reduction of 6% compared with 1990, reflecting the need to make financial savings. To ensure that the number of students appointed was kept within the financial budget, a quota was allocated to the three Technical and Scientific Departments, with the

Table.17: Posts Filled against Complement
(situation as at 31st December 1990)

Team Posts	Complement	Posts Filled
Temporary Euratom Staff	191	182
UKAEA Staff	260	250
DG XII Fusion Programme Staff	19	8
TOTAL	470	440

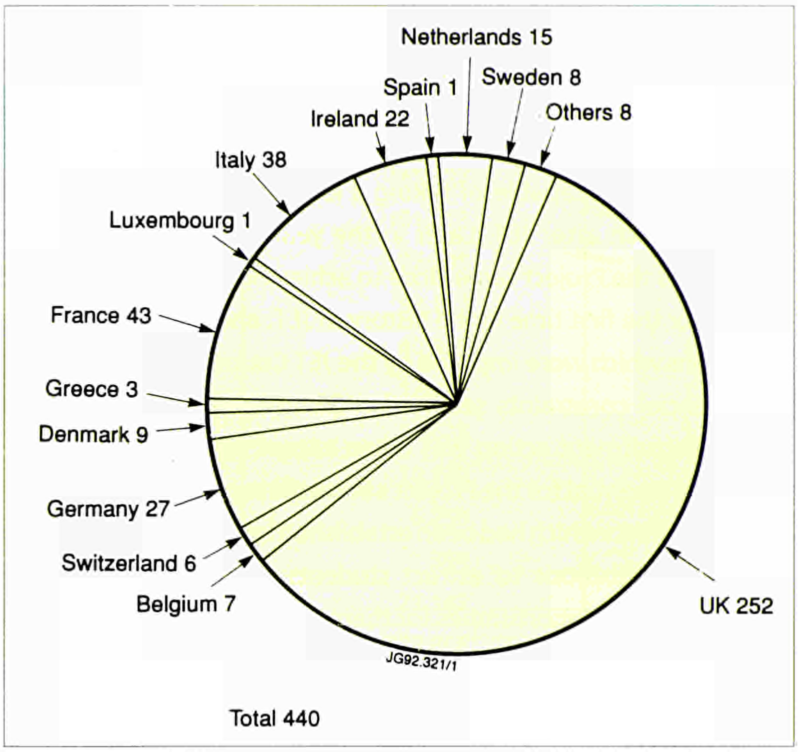

Fig.59: Composition of Team Staff by nationality

additional aim of increasing the proportion of engineering students. This resulted in more than twice as many engineering students being appointed compared with science students. Students have been recruited from all of the partner states except from Luxembourg.

JET Fellows

During 1991, 28 Fellows worked at JET on post-graduate and post-doctoral research, the majority being scientists. A further six candidates have been offered grants for 1992, which should maintain the number of JET Fellowships close to the complement of 25.

Assigned Associate Staff

The work of the project continued to be supported during the year by staff from the Associations, assigned to JET under the Assigned Associate Staff scheme. The contribution under this scheme had gradually decreased in recent years. This is a trend which was not unexpected, as much of the work of building and installing equipment and diagnostics, on which Assigned Associate Staff have been mainly employed, has now been largely completed.

The contribution for 1991 was 27.8 man-years compared with 20 man-years in 1990 and 1989 and about 30 man-years at the peak of the scheme in 1986.

Tables 18 and 19 show the contributions under the scheme from the Associations in 1991, together with the distribution of the Assigned Associate Staff within the project.

Table.18: Contributions from Association Laboratories during 1991

	Associate Laboratory	Man-Years
UKAEA	(UK)	14.0
IPP	(Federal Republic of Germany)	3.3
NFR	(Sweden)	2.4
Risø	(Denmark)	2.0
ENEA	(Italy)	1.9
KfA	(Federal Republic of Germany)	1.7
FOM	(The Netherlands)	1.1
CIEMAT	(Spain)	1.0
JNICT	(Portugal)	0.4
	TOTAL	27.8

Table.19: Assigned Associate Staff within the Project during 1991

Division	Man-Years
Experimental Division 1	12.1
Experimental Division 2	8.8
Theory Division	3.0
Fusion Technology	1.7
Machine and Development Dept.	0.9
Magnet and Power Supplies	0.8
CODAS	0.5
TOTAL	27.8

Training and Conditions of Service

Training

Training has been provided in several key areas, mainly for JET team members and also to an increasing extent for contractors' staff. Much of this training is carried out in-house, the aim being to help new staff to integrate quickly into the JET team and to become familiar with the basic safety requirements of working at JET. Altogether, 700 persons (including contractors) received such training during the year. In addition, as part of the preparation for operating the JET machine with tritium, 260 persons (including contractors) received training in tritium safety. In compliance with the EEC directive concerning the use of VDU equipment, training in office ergonomics was given to 100 staff, who use VDU's for more than 50% of their working time, to help them avoid possible health hazards. Other courses have been arranged to keep staff up to date with technological advances and to provide professional training.

A major change was made in the provision of language tuition at JET as a consequence of financial constraints. After consultation with the Staff Representatives Committee, it was decided that language tuition should be confined to the English language, as the official language of the Project, and tuition in non-English languages should be withdrawn by mid-1992, after staff had completed courses already in progress.

Conditions of Service

Staff conditions are determined by the conditions of service of the two employers. Where possible these are modified in consultation with the employers to meet the unique conditions of JET.

The Retention of Experience Allowance payable to UKAEA team members, which was introduced in 1987, has continued to be effective in retaining UKAEA staff in the Project. Payments of up to 15% of salary were made at the end of the year to 225 staff qualified for the allowance. This figure is about 88% of the total UKAEA strength. UKAEA shift workers have continued to receive the special shift allowance for Satuday working, introduced in 1989 in recognition of the atypical pattern of JET shift working compared with that worked by other UKAEA staff. Bonus payments under the special bonus scheme for beryllium workers, which was introduced last year, were paid this year to 23 UKAEA team members (the same number as last year). These bonus payments are based on the nature of the work carried out in beryllium controlled areas, which requires the wearing of irksome protective clothing.

In addition to questions relating to conditions of service, which necessarily involve consultation with the two employers, other matters relating to the working conditions at JET were considered jointly by the JET management and the Staff Representatives Committee at three meetings during the year. Topics covered included: the review of team posts; operation of the JET promotion procedures; catering facilities; and the EEC directive on the health and safety requirements for VDU users; .

General Administration

Many of the services previously provided by Culham Laboratory continued to be reviewed by the JET Personnel Service with the aim of rationalising the services in order to achieve economies. The increase in the overall numbers of staff working at JET resulted in the need to review the use of existing accommodation and to identify alternative solutions for provision of additional accommodation economically. As a result of this review, the need for additional accommodation was limited to the provision of two new Portacabins, which were erected to house mainly staff who were previously members of the Culham/JET joint services until their transfer to the JET team.

Changes in catering services were implemented in the middle of the year, with the experimental introduction of commercial catering companies providing a lunchtime snack service, in addition to that provided by the Culham Restaurant. This has proved successful.

A new on-site radiopaging system was installed in Autumn 1991. This is the first step towards the new telecommunications system which was in the final planning stage at the end of 1991.

Safety

Organisation and Committees

The JET Director is responsible for safety and is required by the JET Statutes to undertake all organisational measures to satisfy relevant safety requirements. JET continues to meet all the requirements of relevant UK legislation and, in accordance with the Host Support Agreement, JET complies with the safety regulations of the Host Organisation. The JET Safety Group provides a general safety service, including training, monitoring, co-ordination and planning. Special attention was paid to training with particular emphasis on the Preliminary Tritium Experiments. The JET Health Physics Group provide a comprehensive radiological protection and occupational hygiene service, dosimetry service, beryllium analysis and environmental monitoring, both on and off-site.

There are currently four committees on safety related matters:

- The JET Safety Policy Board, chaired by the Director, meets once a year to review safety policy and define new actions;
- The JET Health and Safety Committee, chaired by the Head of Administration Department, consists of representatives of management and staff and reviews all matters which affect the health and safety of all employees on the JET site. It receives reports of Safety Audits. Inquiries into accidents, and accounts of activities of the other JET Safety Committees;
- The JET Fusion Safety Committee, chaired by the Head of Coordinating Staff Unit and includes non-JET members, keeps under review the safety aspects of the Project during design, commissioning and operation, which arise from the use of tritium;
- The JET Safety Working Group, which is chaired by the Head of the Coordinating Staff Unit with members drawn from JET and the Host Patrol Service, has continued to review all aspects of day-to-day safety.

Radiation and Beryllium Data

The collective radiation dose for in-vessel work during the 1990/91 shutdown was 0.066 man-Sv, of which 55% (0.036 man-Sv) was accrued during 1991. A short torus intervention in September led to an additional 0.005 man-Sv resulting in a collective dose of 0.042 man-Sv. The in-vessel radiation exposure amounted to 67% of the total collective dose for the Project during 1991, which was 0.062 man-Sv.

The preliminary tritium experiment (PTE) was carried out during 1991 without any radiological incidents or occurrences. No detectable increase in personal or collective doses resulted from the PTE. This was due to effective containment/confinement facilities and other radiological protection measures. The enhanced neutron activation within the vacuum vessel can be expected, during the 1992 shutdown, to involve rotation of in-vessel workers at an earlier stage than during the 1990/91 shutdown.

Measurements of airborne beryllium continue to demonstrate compliance with the Control of Substances Hazardous to Health Regulations 1988, and indicate negligible personal exposures. 99.9% of exposure assessments made during 1991 were less than $2\mu g/m^3$ - (8 hour TWA) - the internationally recognised standard.

Press and Public Relations

Press and Public Relations activities during the year were dominated by two major events; public Open Days in June and JET's Preliminary Tritium Experiment in November.

During the last weekend in June, members of the public were invited to the JET Open Days, at which all major aspects of the Project were on display. The event attracted about 3500 visitors. The tour included the Control Room, Diagnostics Area, Assembly Hall, Torus Hall, Tritium Processing Plant and the Power Supplies areas. Apart from the JET tokamak, the major attractions were the working remote handling equipment and the spare machine octant, through which visitors could walk (see Fig.60). A new 16-page colour brochure was prepared for the Open Days.

The Preliminary Tritium Experiment, carried out on Saturday 9th November, attracted extensive press interest worldwide. A television crew in the Control Room recorded the event and within hours the news item was broadcast throughout Europe and the United States. This was followed by wide coverage in national newspapers, journals, and radio and television programmes. A Press Conference, held in London on the following Monday, attracted 37 journalists, including television and radio reporters. This news of the first generation of a significant amount of fusion power from a magnetic

Fig.60: Visitors on Open Days viewing the JET Spare Octant

confinement system resulted in 25 radio and television interviews and numerous Press queries, requests for further technical information and public lectures.

Throughout the year, the Project continued to attract a wide range of visitors from the scientific and industrial communities, political circles, the media and the general public. On the diplomatic front, there were visits to JET by the Dutch Ambassador, Mr. Hoekman, the French Ambassador, M. B. Dorin and fifteen members of the London Diplomatic Science Club. Amongst the politicians visiting the Project were the UK Secretary of State for the Environment, Mr. M. Heseltine, six UK Members of Parliament (including four Ministers), the Irish Minister for Energy, Mr. R. Molloy, the Dutch Minister for Education and Science, Mr. J. Ritzen, a Member of the European Parliament, Dr. R. Linkohr, and twelve Japanese Members of Parliament. From industry, JET received Lord Prior, Chairman of GEC-Alsthom, a group of senior French nuclear engineers, a group of Italian industrialists and members of the Power Division of the UK Institution of Electrical Engineers. In addition, 52 journalists were given briefings on JET. Throughout the year, groups of visitors from universities, schools, professional bodies and the general public (about 1750 people in 100 groups) made tours around JET.

Publications Group

The Publications Group provides a Graphics, Phototypesetting, Photographic and Reprographics service for the Project. The Group is lead by the Publications Officer, who is also responsible for the clearance, production and distribution of all JET documents. In addition, the Group arranges JET attendance at major International Conferences, and prepares papers and posters for these Conferences and Meetings.

Conferences

The number of major fusion Conferences held during 1991 was fewer than in 1990, but JET still provided major contributions to a number of meetings, as follows:

- 18th European Conference on Controlled Fusion and Plasma Physics, Berlin, Germany 3rd-7th June 1991 (3 Invited Papers, 7 Oral Contributions and 43 Posters);
- 14th Symposium on Fusion Engineering (SOFE), San Diego, USA, 30th September - 3rd October 1991 (2 Invited Papers and 14 Posters);
- 4th Topical Meeting on Tritium Technology, Albuquerque, USA, 29th September - 4th October 1991, (2 Oral Contributions and 8 Posters);
- 33rd Meeting of the APS Division of Plasma Physics, Tampa, Florida, USA, 4-8 November 1991, (1 Invited Paper and 13 Posters).

In total, the Group prepared 139 Papers and 78 Posters for presentations to 20 different Conferences throughout the world. Arrangements were also made by the Group for 126 participants to attend these major meetings during the year.

JET Open Days

The Group made a major contribution to the JET Open Days which were held for two days on 29th and 30th June. A selection of 117 illustrated display panels and 408 display photographs were produced for the event. In addition, 53 different information sheets and a sixteen page illustrated colour brochure were prepared as handouts for the public.

Publications

The Publications Office is responsible for the clearance and production of all JET presentations (including Journal Papers, Reports, Conference Papers, Poster Contributions, Lectures, etc.). Throughout 1990, over 400 publications were cleared for external presentation.

During the year, 284 documents were published from the Project and the full list is included as an Appendix in the JET Progress Report, 1991. This total included 14 JET Reports, 66 JET Preprints, 12 JET Internal Reports, 4 JET Technical Notes and 5 JET Divisional Notes. All these JET documents are produced and disseminated by the Group on a wide international distribution. In total, the Group produced 3218 illustrations and figures and 3260 photographs for publications and other disseminated material.

APPENDIX I
The JET Council

Member	Representative
The European Atomic Energy Community (EURATOM)	P. Fasella (Chairman) C. Maisonnier
The Belgian State acting for its own part (Laboratoire de Physique des Plasmas - Laboratorium voor Plasmafysica, Ecole Royale Militaire - Koninklijke Militaire School) and on behalf of the Université Libre de Bruxelles (Service de Chimie-Physique II of the UBL); and of the 'Centre d'Étude de l'Energie Nucléaire' (CEN)/'Studie-centrum voor Kernenergie' (SCK)	P.E.M. Vandenplas T. van Rentergem
The Centro de Investigaciones Energéticas Medioambientales y Tecnológicas (CIEMAT), Spain	A. Grau Malonda
Commissariat à L'Énergie Atomique (CEA), France	J. Tachon R. Aymar
Comitato Nazionale per la Ricerca e per lo Sviluppo dell'Energia Nucleare e delle Energie Alternative (ENEA), Italy	R. Andreani C. Mancini
The Hellenic Republic (Greece)	A. Katsanos
The Forskningscenter Risø (Risø), Denmark	H. von Bülow J. Kjems
The Grand Duchy of Luxembourg (Luxembourg)	J. Hoffmann J.P. Zens (to November) Mrs. S. Lucas (from November)
The Junta Nacional de Investigação Cientifica e Tecnológica (JNICT), Portugal	J.T. Mendonça (to October) C. Varandas (from October) Mrs. M.E. Manso
Ireland	M. Brennan F. Turvey
The Kernforschungsanlage Jülich GmbH*, Federal Republic of Germany (KfA)	A.W. Plattenteich (to November)
The Max-Planck-Gesellschaft zur Förderung der Wissenschaften e.V. Institut für Plasmaphysik (IPP), Federal Republic of Germany	K. Pinkau
The Swedish Natural Science Research Council (NFR), Sweden	M.O. Ottosson H. Wilhelmsson
The Swiss Confederation	F. Troyon P. Zinsli
The Stichting voor Fundamenteel Onderzoek der Materie (FOM), The Netherlands	M.J. van der Wiel K.H. Chang
The United Kingdom Atomic Energy Authority (UKAEA)	J.R. Bretherton D.R. Sweetman

Secretary: J. McMahon, JET Joint Undertaking

* name changed to Forschungszentrum Jülich GmbH in January 1990.

APPENDIX II
The JET Executive Committee

Member	Representative
The European Atomic Energy Community (EURATOM)	J.P. Rager P.J. Kind
The Belgian State acting for its own part (Laboratoire de Physique des Plasmas - Laboratorium voor Plasmafysica, Ecole Royale Militaire - Koninklijke Militaire School) and on behalf of the Université Libre de Bruxelles (Service de Chimie-Physique II of the UBL); and of the 'Centre d'Étude de l'Energie Nucléaire' (CEN)/'Studie-centrum voor Kernenergie' (SCK)	R. Vanhaelewyn
The Centro de Investigaciones Energéticas Medioambientales y Tecnológicas (CIEMAT), Spain	F. Manero
Commissariat à L'Énergie Atomique (CEA), France	C. Gourdon (Vice-Chairman) R. Gravier
Comitato Nazionale per la Ricerca e per lo Sviluppo dell'Energia Nucleare e delle Energie Alternative (ENEA), Italy	A. Coletti M. Samuelli
The Hellenic Republic (Greece)	A. Theofilou
The Forskningscenter Risø (Risø), Denmark	F. Øster (to June) Mrs. L. Grønberg (from June) V.O. Jensen
The Grand Duchy of Luxembourg (Luxembourg)	R. Becker
The Junta Nacional de Investigação Científica e Tecnológica (JNICT), Portugal	J. Bonfim F. Serra
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The Max-Planck-Gesellschaft zur Förderung der Wissenschaften e.V. Institut für Plasmaphysik (IPP), Federal Republic of Germany	K. Tichmann
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Secretary: J. McMahon, JET Joint Undertaking

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APPENDIX III
The JET Scientific Council

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