JOINT EUROPEAN TORUS JET

JET JOINT UNDERTAKING

ANNUAL REPORT 1994

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EUR-JET-AR17

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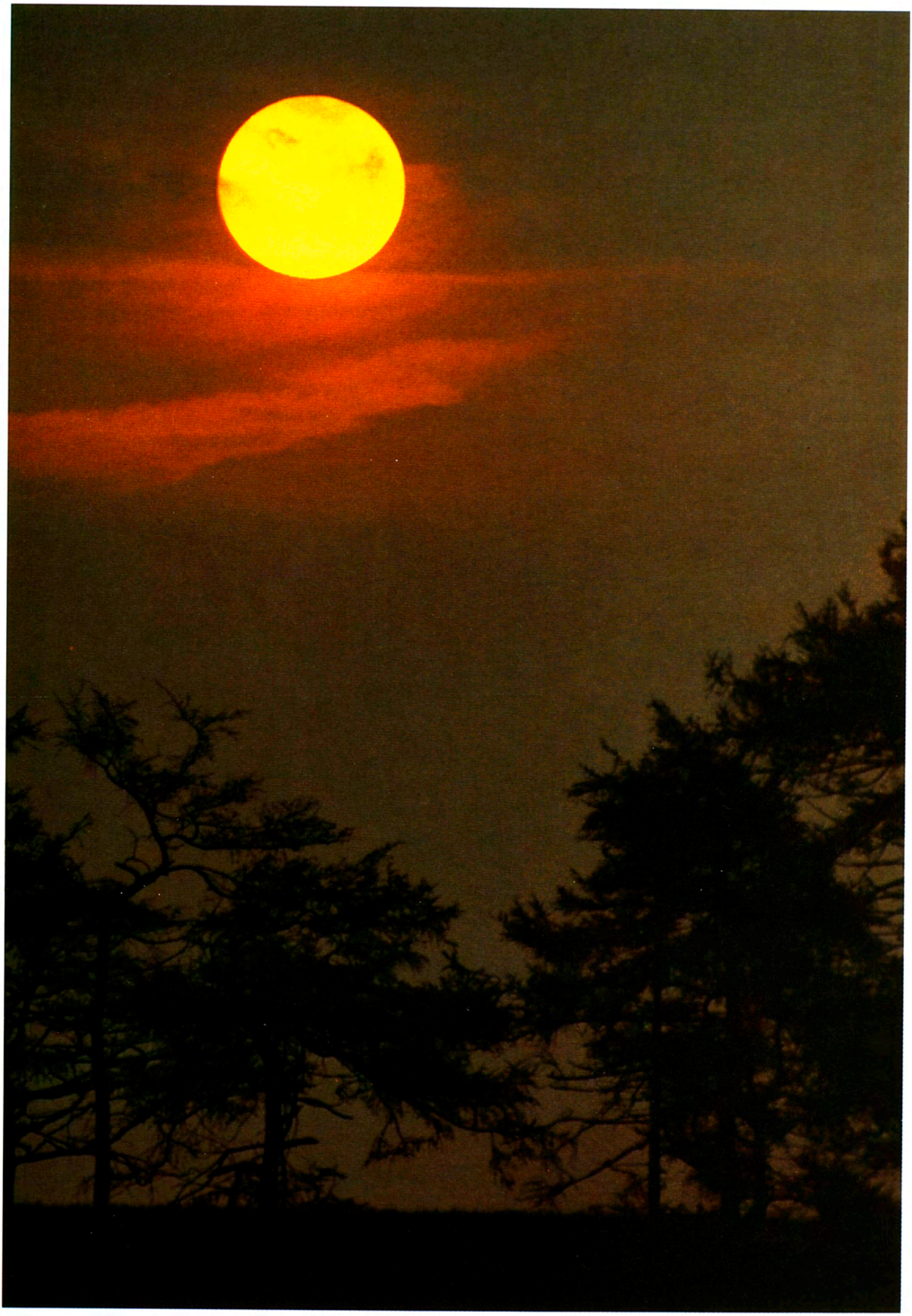


Preface

1994 was a year of significant achievement for the JET Project. The longest and most complex shutdown since the machine was first assembled was successfully completed at the start of the year. By the end of January, the components for the new pumped divertor phase had been installed, which entirely transformed the interior of the torus. The experimental programme began, as planned, early in 1994 with virtually a new machine.

In 1991, JET had been extended to the end of 1996 to enable the Project to establish reliable methods of plasma purity control and plasma exhaust in operational conditions relevant for the Next Step Tokamak, and to prepare for the final phase of JET with deuterium - tritium plasmas. Plasma impurities have always been a major obstacle to steady state operation which is desirable in a fusion reactor. The pumped divertor has been installed in the JET machine in order to control power exhaust and to screen plasmas from impurities which cool the plasma and reduce the number of energy producing reactions.

In February 1994, the first plasmas were produced successfully in the new machine configuration, ushering in the experimental programme. Initial experiments were focused on establishing and characterising plasma behaviour and on confirming reliable operation in the new configuration. The careful design of the divertor target plates and sweeping of the plasma over the target tiles permitted the high power handling capability of the divertor to be demonstrated. As a result, "blooms" of carbon impurities which had previously terminated high performance, could now be eliminated. Furthermore, the Torus cryopump has proven to be effective for the improvement of impurity and density control of the plasma.

The scientific results obtained during the year under review showed improving plasma performance. Energy confinement times of more than one second have been consistently achieved. In the high mode of plasma energy confinement (H-mode), this containment has been maintained throughout a pulse length of 20 seconds. The triple fusion product of plasma density, plasma temperature and confinement time, obtained simultaneously, has reached "near break-even conditions", which compares with the best values reached in JET before the machine was modified. Higher values are expected to be achieved during the continuing campaign in 1995.

The JET machine, as the most powerful fusion experiment in the world, is designed to operate with large plasma currents and high power heating systems. In the new configuration, the plasma current - a major factor in the performance of a tokamak - was gradually increased again to 5 Megamperes. This value greatly exceeds the current employed in any other fusion experiment presently operating in the world.

During the past year, experiments were also conducted with total power from the Neutral Beam Injectors and Radio Frequency Heating systems of over 24 Megawatts in the plasma. This will be further increased during 1995. The experiments showed that reliable operations at these high currents, similar to those required in a fusion reactor, can be sustained in a divertor configuration without major problems in operation.

JET's present experimental programme concentrates on the study of reactor relevant problems and the provision of further important information for the design of the International Thermonuclear Experimental Reactor (ITER) Project.

ITER is a collaborative agreement between Euratom and the Governments of Japan, the Russian Federation and the United States of America. Its basic objective is to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes. ITER would accomplish this objective by demonstrating controlled ignition and extended burn of a deuterium - tritium plasma, with steady-state as an ultimate goal.

The present Engineering Design Activities (EDA) of ITER have identified further research and design issues which need to be addressed, and these can be investigated in JET. A three-year ITER-EDA Support Phase of JET would provide further data of direct relevance to ITER before JET entered into its final phase of D-T operation. The JET Council has therefore decided to seek a further prolongation of the Project until the end of 1999. The prolongation would enable JET to make essential contributions to the development and demonstration of a viable divertor concept for ITER by carrying out experiments using D-T plasmas in an ITER-like configuration. This would broaden the basis for the D-T operation of ITER. JET's proposed experimental programme would allow a co-ordinated, timely and cost effective investigation of the various options for an ITER divertor. Hopefully, the Council of Ministers will approve the JET extension later this year.

Due to temporary blocking by the European Parliament of part of the Community Fusion Programme Budget for 1994, the original budget could not be adopted until August. As a consequence, JET had to operatein the meantime within a constrained budget and rigorous expenditure restrictions were enforced within the Project.

It was with great sadness that we learned of the death of Jean Teillac in Paris on 10 March 1994. Professor Teillac chaired the Interim JET Council established in

1977 and was Chairman of the JET Council from 1978 to 1984. The JET Project is deeply appreciative of the role that he played and the devotion to JET that he demonstrated during the crucial formative years of the JET Joint Undertaking. Those of us who knew him will also fondly recall his humanity, courtesy and dignity. His immense contribution will also be remembered when the history of JET is recorded.

I should like to express my sincere gratitude to the members of the JET Council for their unfailing support during my term as Chairman, which ended in March 1995. Professor Francis Troyon, who has succeeded me, is a long standing member of the JET Council and was Chairman of the JET Scientific Council from 1988 to 1992. I wish him every success in his new role.

I would also like to record my belated congratulations to Dr Robert Aymar, who resigned from the JET Council and was appointed Director of ITER last autumn, and to thank him for his valuable contribution to the Project.

The JET Executive Committee and the JET Scientific Council continued to provide excellent support and advice to the JET Council throughout the year, for which we on the JET Council are deeply grateful.

In conclusion, I congratulate the JET Director, Dr Martin Keilhacker and all the staff of the Project for their outstanding success in completing a demanding and complex programme. I wish the Project every success.

H von Bulow
Chairman of the JET Council
March 1995.

Introduction, Summary and Background

Introduction

The Joint European Torus (JET) 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.

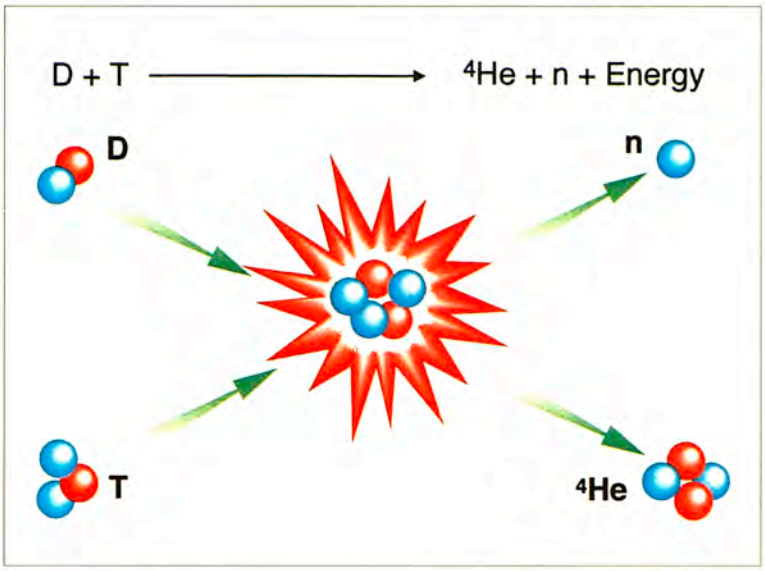
A more detailed and comprehensive description of the technical aspects of the JET Project can be found in the JET Progress Report.

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 introduction, provides an overview of the planning of the Report and sets the background



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 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.

to the Project. This is followed by a description of JET and its experimental programme and explains its position in the overall Euratom and International Fusion Programmes. In addition, it 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 the technical status of JET including: the completion of technical changes during the longest shutdown in JET's history to install the new divertor configuration; preparations for future shutdowns and progress on systems for future operation. This is followed by a section on scientific achievements during 1994. It sets out progress towards reactor conditions and compares the performance between JET and other tokamaks. It shows the substantial achievements made by JET since the start of operations in 1983. 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. It describes the administration of JET, in which it details the budget situation; contractual arrangements; and sets out staffing arrangements and complement.

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.

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, an international design team started work in the

UK, and by mid-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 Joint Undertaking was originally established for a period of 12 years, beginning on 1st June 1978. The device would be built on a site adjacent to 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 in the United Kingdom.

The Members of the Joint Undertaking are Euratom, its Associated Partners in the framework of the Fusion Programme, including Sweden 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, UKAEA pays ten per cent, with the remainder 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 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 approved by the Council of Ministers in December 1991. The extension is to allow JET to implement the new Pumped Divertor Phase of operation, the objective of which is to establish effective control of plasma impurities in operating conditions close to those of the Next Step. An extension of the JET programme to 1999 is currently being proposed in support of the ITER divertor while satisfying the requirements of JET D-T operations.

Objectives of JET

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

Conditions for Fusion

Fusion reactions can only take place if the nuclei are brought close to one another. However, all nuclei carry a positive charge and therefore repel each other. By heating the gaseous fuels to very high temperatures, sufficient energy can be given to the nuclei that the repulsive force can be overcome and they to fuse together. In 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 is put in to heat the fuels and run the system. Reactor power output depends on the square of the number (n) 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 ($n_i \tau_E T$), exceeds the figure of $5 \times 10^{21} m^{-3} keV$. Typical values for the parameters that must be attained simultaneously for a reactor are:

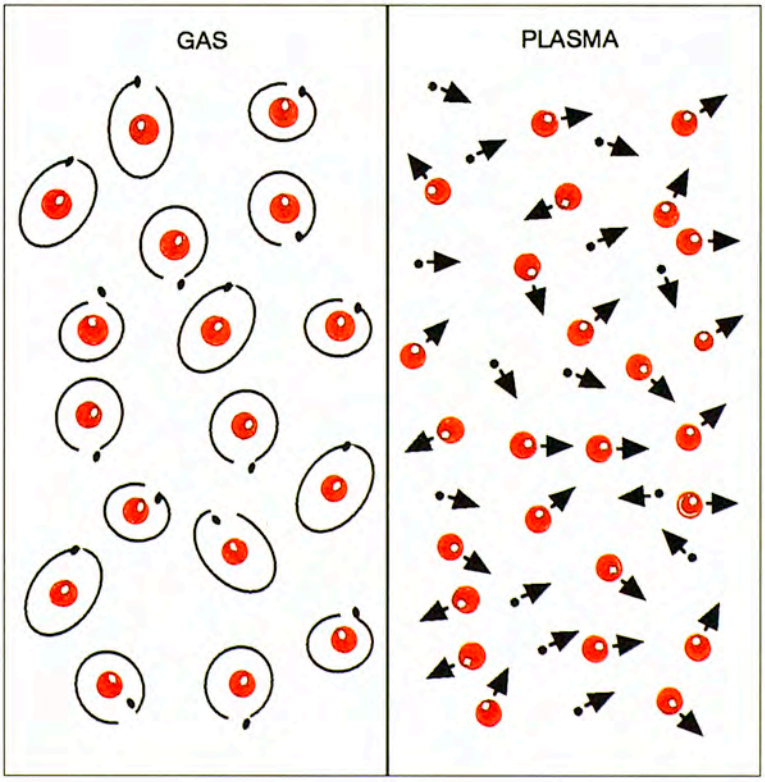
Central ion temperature, T_i
10-20keV

Central ion density, n_i
$2.5 \times 10^{20} m^{-3}$

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).

Fig.1: Aerial view of the JET Joint Undertaking, situated near Oxford in the United Kingdom

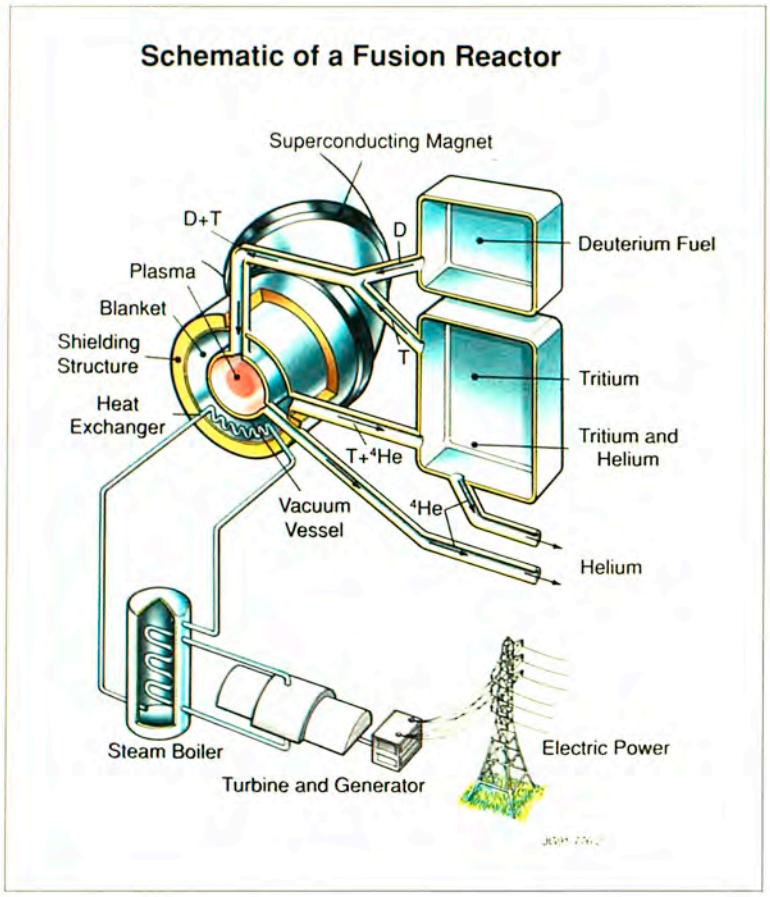


Plasma

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 USSR.



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:

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

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

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

'... 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 key technologies required in fusion reactors: the use of tritium and remote handling 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.

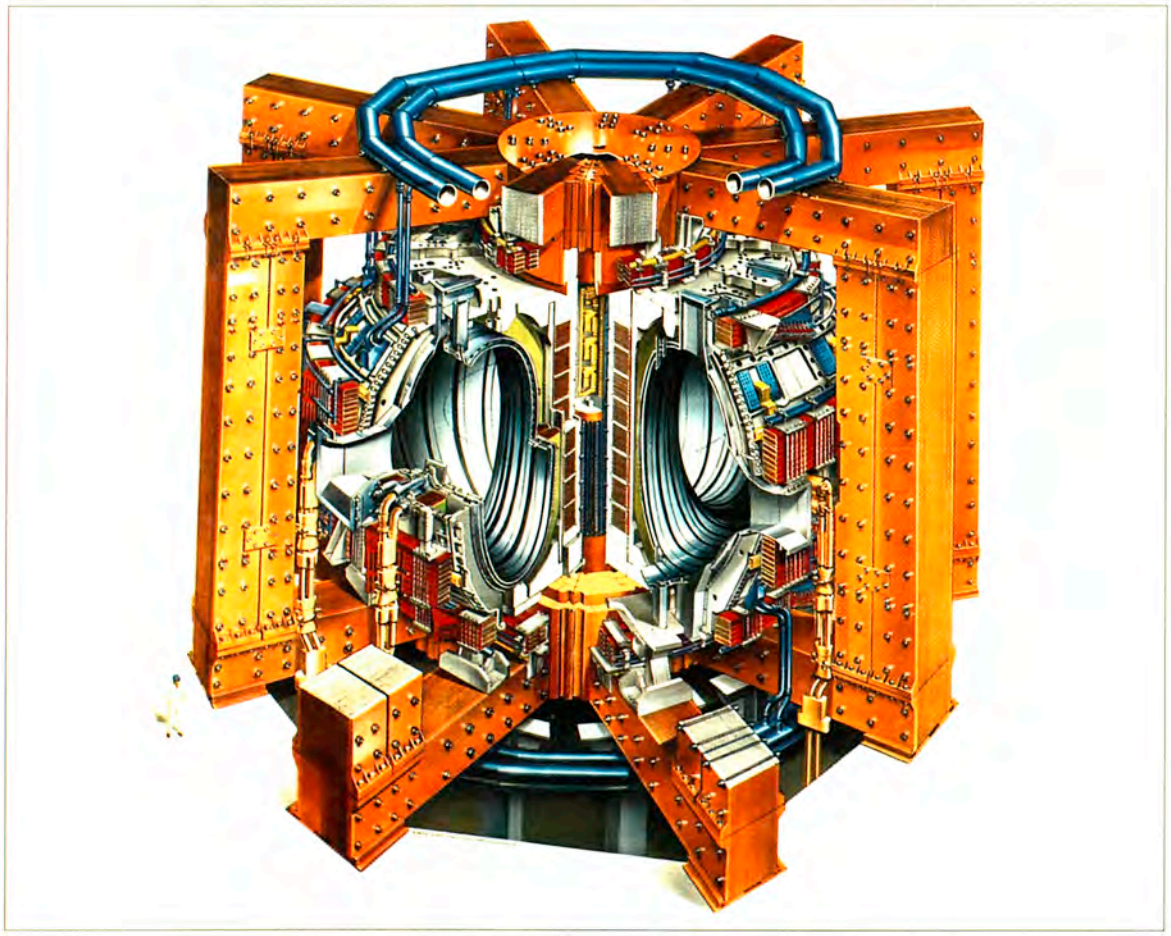
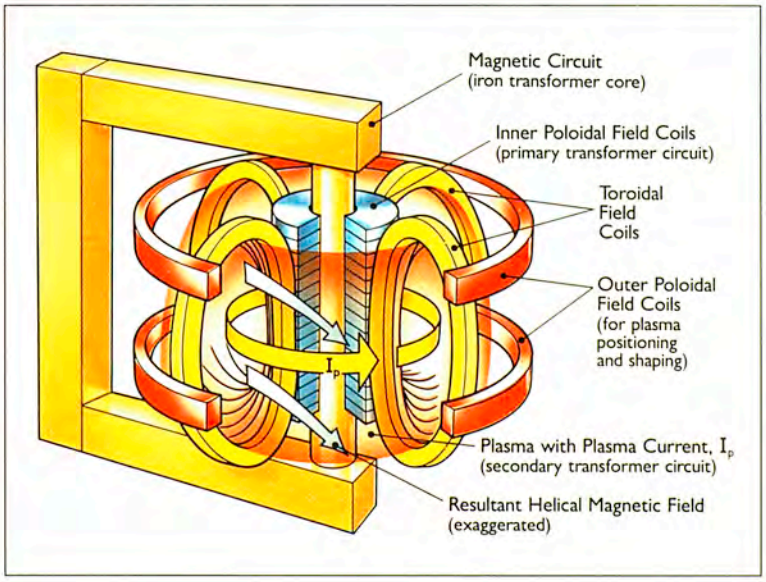


Fig.2: Illustration of the JET Apparatus

PARAMETER	SIZE
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 TO DRIVE PLASMA CURRENT	34Vs
ADDITIONAL HEATING POWER	25MW

Table I: Original Design Parameters of JET

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



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.

positioning, 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.

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 to twenty minutes, and each one can last for up to 60 seconds in duration. 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 gramme.

The construction phase of the Project, from 1978 to 1983, was completed successfully within the scheduled 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. 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 carried out mainly using hydrogen or deuterium plasmas, although during 1991, experiments were performed in helium-3 and helium-4 and a preliminary experiment was performed using 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 produce significant heating of the plasma. During this phase, the machine structure will become radioactive to the extent that any repairs and maintenance would have to be carried out using remote handling systems.

The Community Fusion Programme

Objective and Near-term Programme

The long-term objective of the programme, embracing all activities undertaken in Member States (plus Switzerland) in the field of controlled thermonuclear fusion by magnetic confinement, is "the joint creation of safe, environmentally sound prototype reactors, which should result in the construction of economically viable power stations, which will meet the needs of potential users. In this context, particular attention will be given to the constraints imposed by the requirements of power utilities" (Council Decision 94/268 Euratom of 26 April 1994 concerning the Euratom Framework Programme 1994 to 1998, OJ No L 115, 6.5.94). The long timespan and the large human and financial efforts needed to attain this objective call for a concentration of Community action on the objective, the complete cohesion of the network of organizations associated in the Community action and the full exploitation of cooperation with the major fusion programmes outside the Community.

Safety and environmental issues will play a central role in the realisation of the large devices, which, after JET, are included in the strategy leading towards a prototype reactor. This strategy includes, in particular:

- an experimental reactor, Next Step, the overall objective of which is to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes;
- a demonstration reactor, DEMO, capable of producing significant quantities of electricity.

For the period 1994-98, the priority objective is to establish the engineering design of the Next Step within the framework of the quadripartite cooperation between Euratom, Japan, Russia and the USA on the Engineering Design Activities for the International Thermonuclear Experimental Reactor (ITER-EDA). Specialized studies are also needed to look at possible improvements to concepts in plasma physics and plasma engineering, as well as to carry out the long term technology developments required for progressing towards the exploitation of fusion as an energy source. The results of such studies will be of benefit both in the operation of ITER and, in the longer term, in the conceptual definition of DEMO.

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.

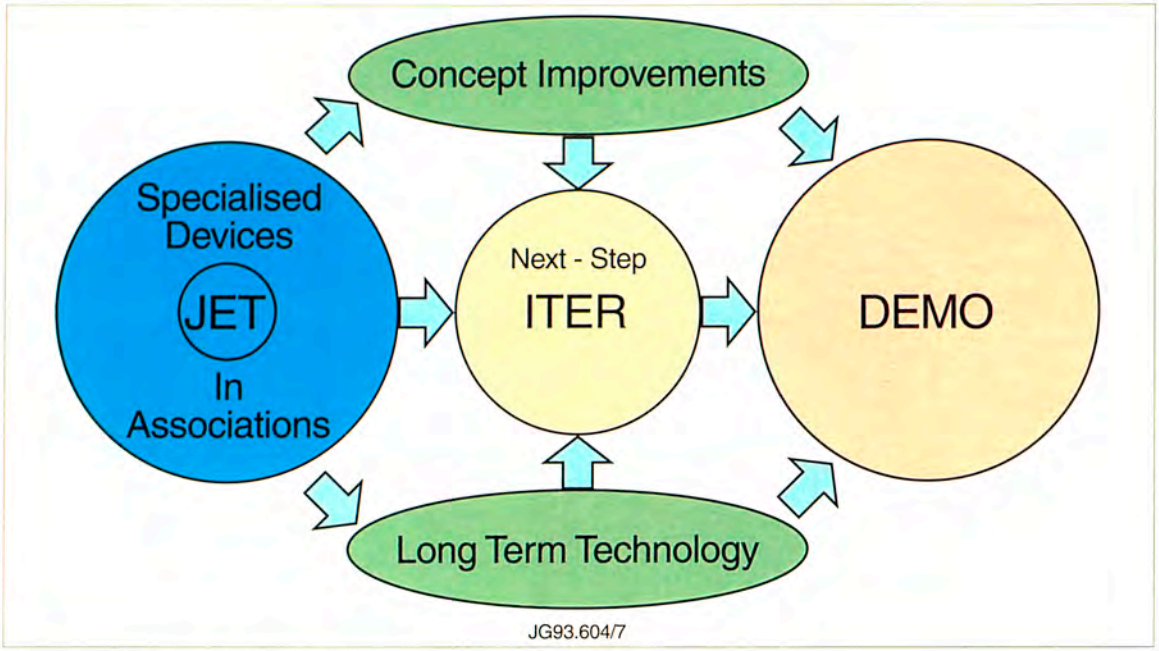


Fig.3: Interactions of European and International Activities.

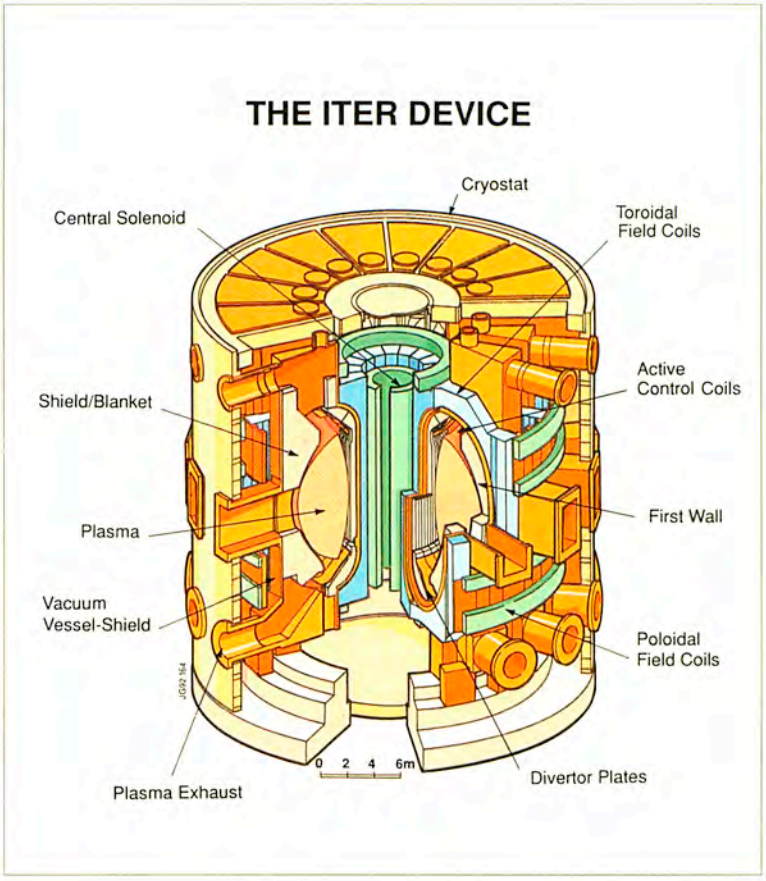
The proposed strategy calls for the simultaneous development of three areas of activity (see Fig.3), on which efforts will be concentrated mainly by means of shared-cost actions:

- Next Step Activities: design proper, and R&D supporting design, construction and operation of the Next Step;
- Concept Improvements: Next Step and DEMO oriented R&D on plasma physics and engineering;
- Long Term Technology: DEMO and reactor-oriented R&D on technology.

1994 Achievements

In the frame of the 1990-1994 Fusion Programme (Council Decision 91/678/Euratom of 19 December 1991, OJ No. L 375, 31.12.1991), a large fraction of the activities in 1994, including those on JET and within the Associated Laboratories, was in support of the Next Step. In particular, the major upgrading of the JET device with the installation of a pumped divertor to establish reliable methods of plasma purity control and essential for supporting the Next Step design.

The Next Step design has progressed in the framework of the quadripartite Agreement (Euratom, Japan, Russia and USA) on cooperation in the ITER-EDA. The overall programme objective of ITER is "to demonstrate the scientific and technological feasibility of fusion



Principal Parameters of ITER

Major Radius	*8.0m*
Minor Radius	*3.0m*
Elongation	*1.6*
Plasma Current	*24MA*
Toroidal Field	*5.7T*
Fusion Power	*1.5GW*
Burn Time	*1000s*

energy for peaceful purposes." ITER would accomplish this objective by "demonstrating controlled ignition and extended burn of deuterium-tritium plasmas, with steady-state as an ultimate goal, by demonstrating technologies essential to a reactor in an integrated system, and by performing integrated testing of high heat-flux and nuclear components required to utilize fusion energy for practical purposes". The Agreement further specifies that the ITER-EDA - to be conducted by the four Parties under the auspices of the IAEA (International Atomic Energy Agency) and carried out by a Joint Central Team (JCT) located in three internationally staffed joint work sites in San Diego (USA), Naka (Japan) and Garching (EU) and by four Home Teams - shall be implemented by two or more Protocols. The ITER Council, in January 1994, considered an Outline Design report produced by the ITER JCT as "an acceptable basis for consideration by the Parties" to proceed, Protocol 1 was completed and Protocol 2 (covering the period up to the scheduled completion of the EDA in July 1998) was signed on 21 March 1994.

The Next European Torus (NET) Team pursued its Next Step related activities, in particular in assisting the Euratom Home Team

(HT) Leader in the performance of his duties and responsibilities in the ITER frame. An industrial grouping was selected to contribute to the overall design activities for the Next Step. Lists of qualified firms have been established for the R&D and supply of prototypes for the Next Step in fifteen technologies specific to fusion.

The Safety and Environmental Assessment of Fusion Power (SEAFP) was performed as a collaborative effort by the NET Team, the Euratom-UKAEA Association and a temporary grouping of industries, and with the participation of other Associations and the Joint Research Centre (JRC). The report will be issued in early 1995.

The conceptual design of a powerful source of high-energy neutrons for testing fusion-relevant materials has started in the multilateral framework (presently Euratom, Japan, USA) of an Implementing Agreement in the frame of the IEA (International Energy Agency, Paris): the development and characterisation of radiation-resistant and low-activation materials is underway. R&D on tritium breeding blankets was pursued, with the aim of building DEMO-relevant blanket modules to be tested in ITER.

In the Associated Laboratories, specialised devices have made contributions to the development of the physics base for ITER and DEMO. They explore, in support of the main objectives of the programme, the accessible parameter range and the possible modes of operation for the different confinement concepts and configurations. Furthermore, these devices provide testbeds for the development of new concepts and techniques for plasma engineering and wall technology. These also serve for fundamental fusion physics studies, for the development of diagnostics, for the preparation of collaboration on larger devices, for innovative studies and for the training of young professionals, as well as being the link which allow the incorporation of university research into theoretical, numerical or diagnostics activities.

Progress in the understanding and control of confinement, plasma-wall interaction, fuelling and exhaust, as well as heating and current drive was achieved on the specialised Tokamaks (TORE SUPRA, ASDEX-Upgrade, TEXTOR, FTU, TCV, COMPASS, RTP, ISTTOK, etc) and in the accompanying programmes.

Studies on alternative lines in toroidal magnetic confinement were pursued: absence of disruptive density limitation on the W7-AS

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.

Stellarator; operation of the RFX Reversed-Field Pinch to higher currents and further diagnostics developments (e.g., polarimetry); start of operation of the EXTRAP-T2 Reversed-Field Pinch and of the TJ-1U Stellarator; and engineering design and prototype development for a possible large Stellarator (W7-X).

In technology, activities include, in particular: superconducting coils development, plasma facing components, fuel cycle, remote handling, and operational and environmental safety.

The scientific and technical achievements of the Community Fusion Programme place Europe in the forefront of world fusion research. Concentration on the most successful toroidal magnetic confinement line, the tokamak, and on a few promising lines akin to it - while keeping a watching brief on inertial confinement fusion and on other approaches - continues to be justified by the results.

Management of the Programme

All magnetic fusion research is integrated into one Community Fusion Programme which presents itself as a single body in its relations with other world fusion programmes. The European Commission, assisted by a Consultative Committee for the Fusion Programme (CCFP) composed of national representatives, is responsible for the implementation of the Fusion Programme. The Programme operates principally through: contracts of Association with Member States (plus Switzerland) or organisations within Member States, the JET Joint Undertaking, the NET Agreement which takes into account Euratom participation in the ITER-EDA, the JRC, contracts of limited duration (in particular with organisations in Member States without Association) and industrial contracts. Through the multipartite Agreement for "Promotion of Staff Mobility", the mobility of scientists and engineers was developed. In coordination with the "Human Capital and Mobility" programme, fellowships were awarded. Dissemination of knowledge and exploitation of results was performed through laboratory reports, publications in scientific journals, workshops and conferences. An itinerant exhibition was further developed and run by the "Fusion EXPO" consortium.

Community financial participation continued to be about 25% of the running expenditure of the Association, 45% of capital cost of projects having been awarded priority status, and 80% of JET

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.

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.

expenditure. Currently, Community funding of fusion research is about 200 MioECU per annum: when funding by national administrations and/or bodies is taken into account, the expenditure on fusion from all sources in Europe amounts to about 450 MioECU per annum. About 1,750 professional scientists and engineers are currently engaged in fusion research in Europe.

The 1994 - 1998 Fusion Programme

The March 1994, assessment by the CCFP of the evolution of the Programme in the medium-term was instrumental in the preparation of the 1994-1998 Fusion Programme Decision (Council Decision 94/799/Euratom of 8 December 1994, OJ No. L 331, 31.12.94). Following this decision, the envelope for fusion activities amounted to 840 MioECU (of which 46 MioECU was for fusion activities at the JRC). Such a figure could possibly be increased if the overall budget for the two Framework Programmes (European Union and Euratom), presently established at 12,300 MioECU, were to be increased by up to 700 MioECU in 1996.

In the frame of this Programme, the following actions have been undertaken or initiated: a proposal to extend JET until 1999 to perform specific tasks in support of ITER, is being considered; proposals for upgrading existing devices (ASDEX-Upgrade, TORE-SUPRA) and for the construction of new ones (Wendelstein 7-X, MAST) are being assessed; a proposal for joining the activities of three Associations into a "Trilateral Euregio Cluster" is under consideration.

International Collaboration

The Community approach has led to an extensive collaboration between the fusion laboratories. For example, most Associations undertake work for other Associations. The Associations are partners in JET, NET and ITER and carry out work for them through various contracts and agreements. The Programme has built across Europe a genuine scientific and technical community of large and small laboratories, readily able to welcome newcomers (as it is the case with imminent signature of a contract of Association between Euratom and TEKES-Finland and with the exploratory discussions concerning a new Association in Austria), and directed towards a common goal. The leading position of the Community Fusion

Programme has also made Europe an attractive partner for international collaboration. Apart from the most far-reaching collaboration illustrated by the ITER project, bilateral Framework Agreements have been concluded (with USA and Japan) or are to be concluded (with the Russian Federation). Also, a Memorandum of Understanding with the Government of Canada, which is ready for signature, will cover inter alia the involvement of this country in the Euratom contribution to the ITER-EDA. Finally, there are nine Implementing Agreements in the frame of the IEA (the latest one, on "Nuclear Technology of Fusion Reactors", was signed in 1994).

Disruptions

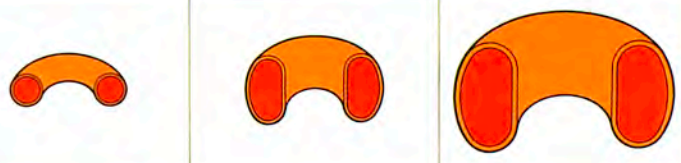
There is a maximum value of density which can be contained with a given plasma current. If this value is exceeded a disruption occurs when the plasma confinement is suddenly destroyed and the plasma current falls to zero in a short period of time. Under these conditions high mechanical and thermal stresses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

Large International Tokamaks

In 1994, JET, the largest tokamak in the world, restarted operation in its new divertor configuration. It soon demonstrated the high power handling capabilities of its new divertor targets and that the severe impurity influxes ("carbon blooms"), which previously terminated high performance plasmas, were eliminated.

Further, the large TFTR tokamak at PPPL, USA, has made a major contribution to the fusion effort with its extensive tritium programme. This has now lasted well over a year and has recently produced in excess of 10MW of peak fusion power.

In addition, the large Japanese tokamak JT-60U continues to improve its performance by using boronisation of its vacuum vessel. It



		TFTR	JET	ITER
MINOR RADIUS	a	0.85m	1.25m	3.0m
MAJOR RADIUS	R	2.5m	2.96m	8.0m
ELONGATION	κ	1.0	1.8	1.6
TOROIDAL FIELD	B	5.2T	3.45T	5.7T
INPUT POWER	P	32MW	36MW	30-200MW
FUSION FACTOR	Q_{DT}	0.3	1.1	30 - IGNITION
PLASMA CURRENT	I	3MA	7MA	24MA

Fig.4: Operating parameters of three large tokamak designs

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	1.00	1.8	2.96	7.0	3.5	42	1983
JT-60U	JAPAN	0.85	1.6	3.2	4.0	4.2	40	1991
TFTR	USA	0.85	1.0	2.50	2.7	5.6	40	1982
TORE-SUPRA	FRANCE	0.80	1.0	2.4	2.0	4.2	22	1988
T-15	CIS	0.70	1.0	2.4	2.0	4.0	-	1989
DIII-D	USA	0.67	2.5	1.67	3.0	2.1	22	1986
ASDEX-U	GERMANY	0.5	1.6	1.65	1.2	2.7	7	1991
FT-U	ITALY	0.31	1.0	0.92	1.2	7.5	-	1988

Table 2: Large Tokamaks operating around the World

has achieved the highest fusion product ($n\tau_E T_i$) of $12.3 \times 10^{20} m^{-3}$keV at an ion temperature of 41keV, with an equivalent Q_{DT} value of ~0.6. It has also reached quasi-steady-state conditions in an ELMy H-mode in deuterium with a high fusion product of $4.5 \times 10^{20} m^{-3}$keV and an equivalent Q_{DT} of 0.25-0.36

Now that the ITER Engineering Design Activity (EDA) is underway, achievements in tokamak research and, particularly, for the largest tokamaks (Fig.4), have become even more relevant. Table 2 sets out an overview of the larger tokamaks worldwide, together with their main parameters and starting dates. Considerable progress has been made by these tokamaks, and these are detailed below.

MACHINE	JET	JT-60U	TFTR	DIII-D
ELECTRON TEMPERATURE				
T_e(keV)	11	10	11.5	6
ION TEMPERATURE				
T_i(keV)	18	41	44	5.5
DURATION (s)	1.5	0.6	0.5	0.5
FUSION PRODUCT				
$n_i T_i \tau_E$ ($\times 10^{20} m^{-3}$keVs)	10	12.3	6.6	5

Table 3: Fusion Products in Large Tokamaks

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 outer layer of 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.

Peak Fusion Performance

The highest peak performance as measured with the triple product of deuterium density, energy confinement time and central ion temperature $n_D \tau_E T_i$ is now taken over by JT-60U with a value of $12.3 \times 10^{20} m^{-3} keV s$ at a central ion temperature of 41keV. This is calculated to yield a fusion efficiency Q_{DT} of 0.6. This is still surpassed by the Q_{DT} of JET which was close to unity in 1991. During 1994, JET has also reached high values in the divertor configuration, with triple products of $7.4 \times 10^{20} m^{-3} keV s$. DIII-D and TFTR in the USA have both maintained considerable progress and have doubled their previous records this year.

Peak values of the fusion triple product obtained in the four largest tokamaks are shown in Table 3.

Technological Developments

The heat load to the divertor and the particle removal rate both pose crucial problems that must be solved before a tokamak reactor can be built. Several forms of advanced divertors are emerging, generally involving closed divertors with large active cooled areas. This allows for impurity retention as well as a high divertor density, so that most of the power entering the divertor can be radiated before it strikes the target plates. This power can then be absorbed over a much larger area.

In JET, divertor plasmas heated with up to 26MW have been thermally stable and maintained at high density with more than 70% of the power radiated. However, the discharge returned to the low confinement (L-mode) regime. ASDEX-U has achieved similar results with high radiation and detached divertor in the H-mode regime.

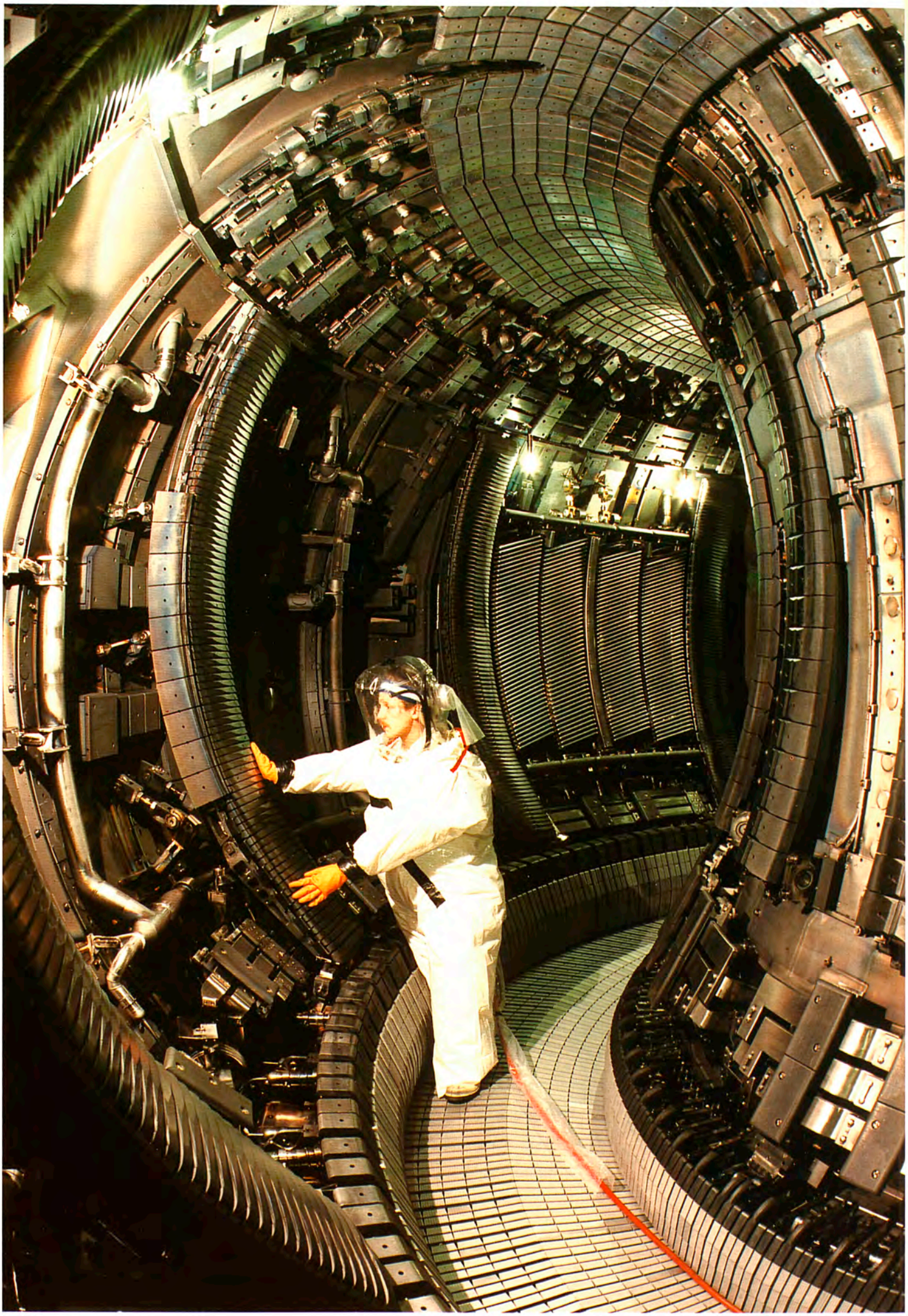
In the JET and DIII-D machines, divertor cryopumps have proven to be highly successful in maintaining quasi-steady-state density regimes at high fuelling and pumping rates. In addition, TORE-SUPRA in France has demonstrated that an ergodic divertor with impurity puffing can efficiently radiate away incoming power and retain those impurities.

Tritium Experiments

After the successful first tritium operation carried out in JET during 1991 with a D-T mixture containing 11% T, tritium experiments have been carried out in TFTR, USA, using a 50:50 D-T mixture. JET has plans to carry out further tritium experiments during its 1996 campaign.

TFTR operation has been producing fusion power exceeding 10MW. It has carried out many D-T experiments over a period of about a year and has processed 380kCi of tritium in that period. In total, 680MJ of fusion energy has been produced. In addition, TFTR has detected the first signs of alpha-particle heating. Plasmas of 2.7MA current in a 5.5T toroidal field, in the so-called supershot regime, have been heated by 39MW of neutral beam power of 50% deuterium and 50% tritium neutrals. Even though the Toroidal Alfven Eigenmode (TAE) activity increased with the obtained fusion power, no major effect on the alpha-particles was observed. The helium (cold alpha-particles) transport was found to be roughly the same as for the hydrogenic species.

An extension of the JET Programme to the end of 1999 is currently being proposed. Two periods of D-T operation are foreseen for the JET programme to the end of 1999, subject to the necessary approvals. JET will make important contributions to D-T physics for ITER (including H-mode threshold, ELM and confinement behaviour, and some radio frequency heating studies). The first period (DTE-1), scheduled for the second half of 1996, would check whether the more favourable confinement found during D-T operation in TFTR extends also to the ITER-relevant divertor and operating conditions in JET. In addition, it should demonstrate long pulse fusion power production. JET alone can investigate these topics in its divertor configuration. The second period (DTE-2), scheduled for 1999, should permit a thorough study of D-T plasmas with enhanced levels of alpha-particle heating, capitalising on the performance improvements achieved in the preceding experimental campaigns with deuterium.



Technical Status of JET

Introduction

JET entered 1994 nearing the end of the longest and most extensive modification since the initial assembly of the device. Two faulty toroidal magnetic field (TF) coils had been replaced and the new pumped divertor configuration had been installed. The interior of the torus had been completely rebuilt to include four new divertor coils, an inertially-cooled divertor target structure and a cryogenic vacuum pump.

The main objective had been to install the components of the pumped divertor and undertake its associated system modifications, which were aimed at establishing in deuterium plasmas "reliable methods of plasma purity control under conditions relevant for the Next Step Tokamak". The main components of the divertor are, as follows:

- the four poloidal coils can create various magnetic configurations, with the X-point far away from the target plates, and includes the capability of X-point sweeping to spread the thermal load. The coils are contained in a 1.2mm thick inconel casing, on which the rest of the divertor components are installed;
- the target plates, arranged in a U-shaped contour, collect power released from the plasma. These consist of an inertially water-cooled structure, which support carbon fibre composite (CFC) tiles, shaped to maximise the power wetted surface. The CFC will be replaced in 1995 with beryllium tiles, for comparison of plasma performance;
- the toroidal cryopump controls plasma density in the divertor region. It is anchored to the outer coil casing and its main components are a water-cooled baffle, a liquid nitrogen cooled copper backed panel, a set of helium cooled pipes and a chevron structure.

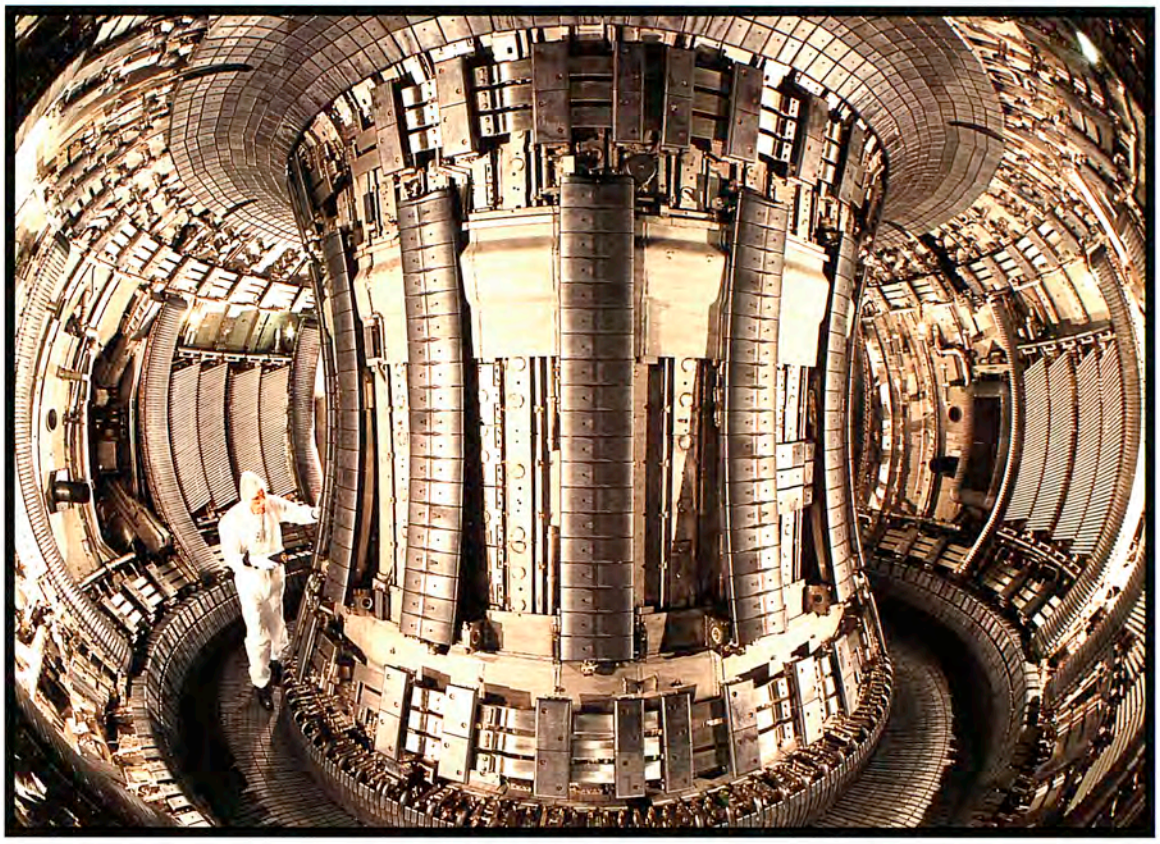


Fig.5: Inside of vessel after completion of installation of components

The heating and current drive systems were also modified and now comprise an upgraded neutral beam (NB) system to inject 7.8MW of power at 140keV and 13.6MW at 80keV, new Ion Cyclotron Resonance Frequency (ICRF) antennae (designed to couple 20MW) and a new Lower Hybrid (LH) launcher (designed to couple 10MW). Other new installations included a system for plasma current and shape control, a Fast Radial Field Amplifier for vertical position control, eight Saddle Coils inside the vacuum vessel for disruption control and for the study of Toroidal Alfvén Eigenmodes (TAEs) and an extensive array of new diagnostics, especially for divertor measurements.

The shutdown was successfully completed with pumpdown of the torus in mid-January 1994. The inside of the vessel on completion of the shutdown is shown in Fig.5. The first plasma in the new pumped divertor configuration was produced in mid-February and by mid-March successful 2MA divertor plasmas had been established. During 1994, the plasma current was increased to 5MA, the total heating power to 26MW, the stored energy to 11.3MJ and the neutron rate to 4×10^{16} neutrons per second.

The high power handling capability of the Mark I divertor target has been demonstrated and the severe impurity influxes (carbon "blooms"), which previously terminated high performance plasmas, have been eliminated. The cryopump reduces recycling, eliminates the effects of wall saturation (observed in previous long pulse operation), allows effective particle control, and generally allows higher performance.

The two NB injectors have routinely provided up to 19MW and 140MJ has been injected during a 20s ELMy H-mode. 13MW of ICRF power has also been coupled, but this is limited by a combination of unsatisfactory control electronics, unequal coupling of the straps of the antennae array and low power transfer to the plasma under some phase conditions. New control electronics systems are currently being installed and will be tested with plasmas in early 1995. Further antenna modifications are scheduled for the beryllium tile exchange shutdown and the Mark II divertor shutdown planned to start in March and June 1995, respectively. Combined NB and ICRF heating powers up to 26MW have been injected and close to 3MA has been driven non-inductively with up to 6MW of LH power.

The saddle coils have been used for initial experiments on TAE modes and the disruption feedback stabilisation system is in the final stages of commissioning. Only the lower saddle coils are now available for experiments, since the upper saddle coils were disabled in September 1994 after being damaged.

During 1994, three manned in-vessel interventions were carried out in full air-suits (since beryllium evaporation had been used). In late March 1994, following first experiences with the restart of operations, an inspection showed that no significant in-vessel damage had occurred. In September/October 1994, damage to the upper saddle coils was stabilised and a further diagnostic system was installed. In early November 1994, the beryllium evaporator heads were removed, strengthened and reinstalled.

The following sections detail the technical achievements made during 1994.

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 a 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.

Technical Achievements

Commissioning of Divertor Coils

The purpose of the JET coil system is to establish, maintain and control the tokamak plasma configuration (Fig.6). It includes: the toroidal coils,

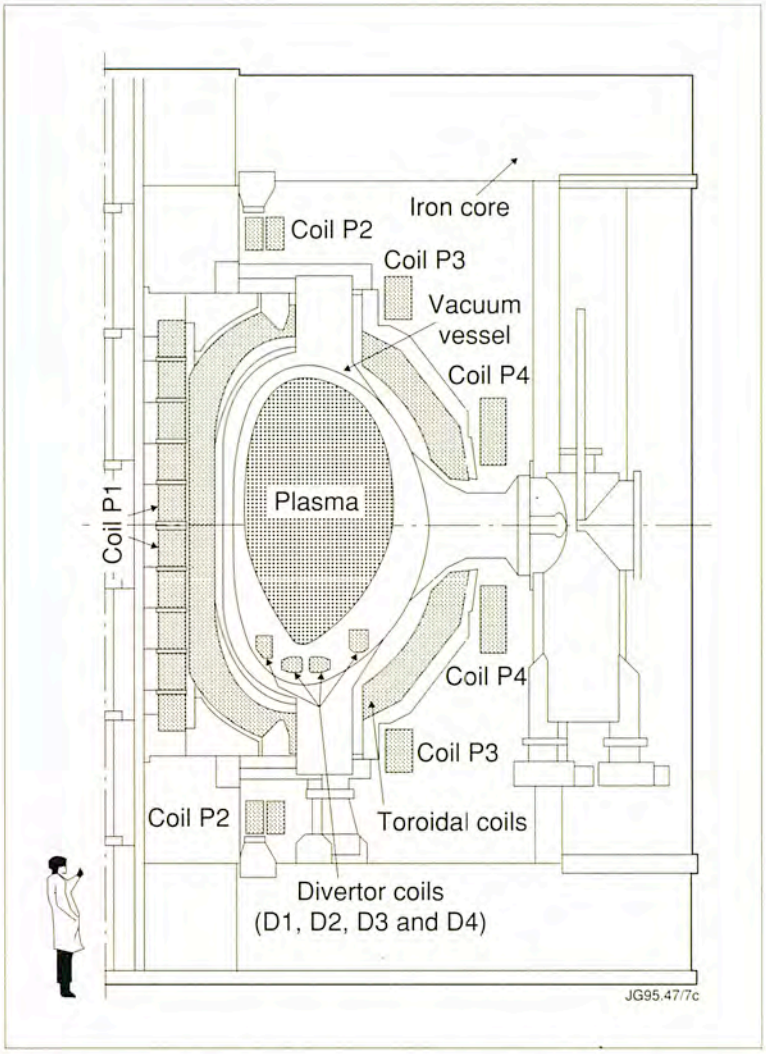


Fig.6: Cross-section of JET showing toroidal, poloidal and divertor coils

which establish the toroidal magnetic field; the poloidal coils P1, acting as primary windings of the tokamak transformer; and coils P2, P3 and P4 to control plasma radial position, vertical position and shape. The divertor coils D1, D2, D3 and D4 were installed inside the vacuum vessel, and now form an integral part of the poloidal system. In conjunction with existing coils, these coils establish and control the new divertor magnetic configuration.

In early 1994, the divertor coils were progressively commissioned in line with experimental requirements, which was most activity centred around the thermal performance of the coils and cooling system. The coils are heated from two sources: the ohmically generated heat due to the coil current; and the heat transfer from the surrounding vacuum vessel and in-vessel components. The coils are cooled by freon pumped through the conductor cooling channels. The cooling system is designed

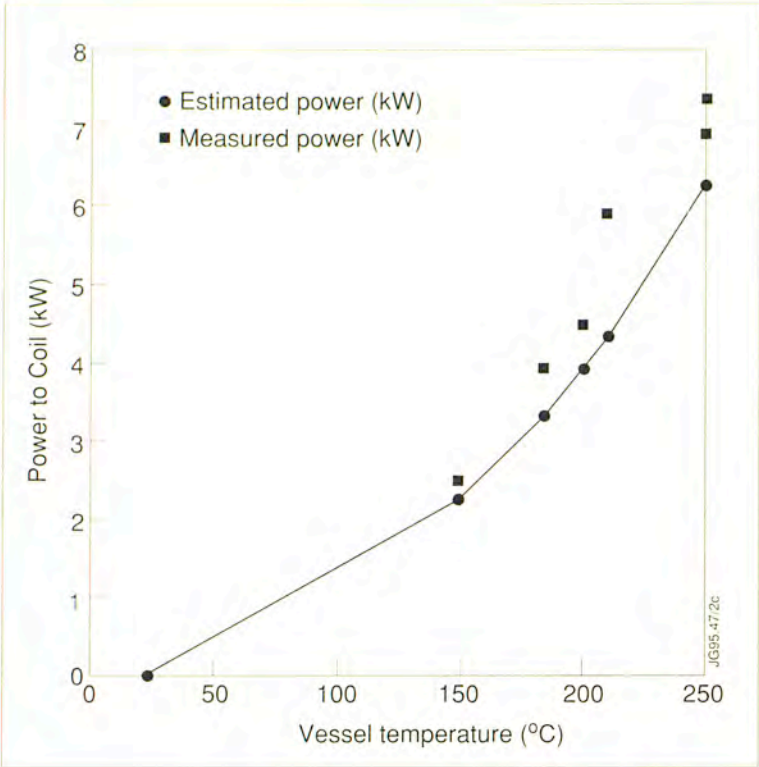


Fig.7: Heating power to Coil D4 with target plates uncooled

to maintain the measured temperature difference across the coils at 15°C. This limits the thermally developed shear stress in the inter-turn insulation to 10MPa.

The thermal performance of the coils has been monitored. The aim is to estimate whether any part of the epoxy-glass insulation might be subjected to temperatures which could degrade its properties as an insulating or structural material. Since direct measurements of the epoxy temperature are not possible, the heat power load to the coils, and the temperatures on the inconel casing have been measured. The present Mark I divertor target plate assembly is supported on the divertor coils, and its alignment ultimately relies on the mechanical stability of the epoxy. Calculations were also made to evaluate faults such as loss of vacuum or failure of the coil cooling system. From the thermal point of view, the epoxy is a matter of concern. The temperature of the casings is monitored, at four points per coil, by thermocouples welded to the casing. To assess the power scaling and the main heat transfer mechanism to the coils, measurements were performed with the vacuum vessel at several temperatures in steady state conditions. Tests were performed, to assess the effect of a vacuum loss in the vessel. H_2 and N_2 gases were injected up to 10mbar pressure, with the vessel at

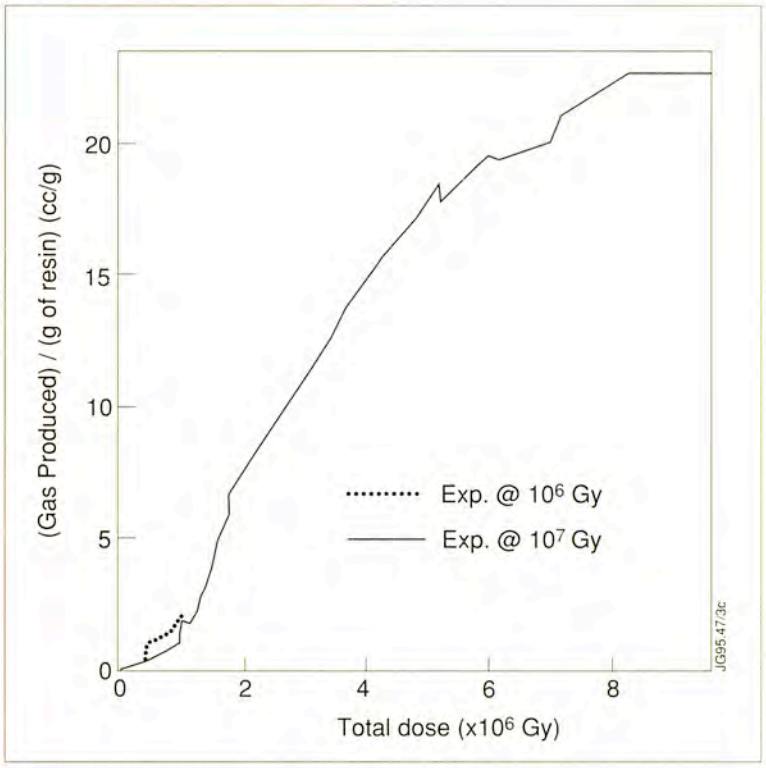


Fig.8: Gas released under irradiation

150°C and 250°C, respectively, and the increased heat power measured (see Fig.7).

Molecular conduction appeared to be the main means of heat transfer across the inter-space, between casing and epoxy insulation. This interspace, which is pumped down to about 1 millibar was formed during resin polymerisation. The epoxy layer under coils D2 and D3 was most critically affected by the vessel temperature, due to the larger than expected, heat conduction from the target plate graphite tiles to the support structure. As a consequence, baking and operation with the target plate uncooled is now restricted to a maximum vessel temperature of 200°C per hour, and with target plates cooled to 320°C.

Irradiation Tests on Divertor Coil Insulation

In the D-T phase, the divertor coils will operate under neutron and gamma fluxes which may affect the characteristics of the electrical insulation. To assess the effect of radiation damage, test samples were irradiated in conditions close to those during the active phase. Irradiation conditions achieved were fast neutron fluence rate corresponding to a dose rate of $6Gys^{-1}$ and a gamma rate of $5Gys^{-1}$.

Three irradiations were performed at total doses of 10^5, 10^6 and 10^7Gy, respectively.

Under irradiation, covalent bonds of the epoxy resin atoms were broken forming free radicals, which recombined in new compounds with alteration of physical, mechanical and chemical properties. The material lost tensile, shear and impact strength and became brittle. Moreover, freed protons and methilic groups recombined producing hydrogen (mostly), methane and ethane. Up to $4x10^5$Gy, gas was released during the three exposures, with a slight regular increase with increasing doses. At higher doses, the production rate strongly increased with increasing doses, sometimes with abrupt changes, as shown in Fig.8. This behaviour can be explained by the presence of cavities in the resin, either formed during impregnation and/or formed due to material degradation under irradiation. The gas produced filled the cavities, then it was slowly released.

It was concluded that:

- loss of the insulation mechanical properties preceded loss of electrical properties. The resin started to become brittle at doses higher than 10^6Gy, but at a dose of 10^7Gy, the electrical resistivity value was still acceptable for insulation purposes;
- radiolysis was the actual insulation life limiting phenomenon;
- gas release was not significant up to 10^5Gy, but at higher doses, it caused swelling and a pumping system would be required to evacuate the gas from the divertor coils cases.

New Coil Protection System

Due to complexity of the new magnetic coil system, a new Coil Protection System (CPS) has been introduced to protect the coils against: electrical faults; and electrical, mechanical or thermal over-stressing due to operation outside safe limits. Electrical faults are detected by comparing the actual electrical performance of the coils with a circuit model run in real time. Electrical, mechanical and thermal stresses are computed from the currents flowing and voltages applied to the coils. If predetermined values for any of these parameters are exceeded, the pulse is terminated. The protection system is implemented fully digitally using a high performance digital signal processor.

The radial and vertical forces on the outer poloidal coils P2, P3 and P4 and the vertical forces on the divertor coils are computed using fluxes measured by coil mounted flux loops and measured ampere-turns. It was not possible to fit vertical flux loops on the divertor coils, so that radial forces acting on these coils are computed analytically. The tensile and shear stress of each coil is computed as a linear combination of vertical force, radial force and temperature. A simple model is used to estimate the temperatures of the epoxy insulation and copper windings of the divertor coil. The inputs to the model are the vessel temperature, coil case temperature, coils currents, coolant flow and coolant inlet and outlet temperatures.

The system became operational in May 1994 and was progressively commissioned and upgraded to include new requirements. Statistics for the four-months of July to October 1994 have shown that the protective actions were mainly caused by parameters being outside limits, due to induced currents and voltages during disruptions. Early experience indicates that the system will be useful and reliable. Due to its flexibility, protection functions not included in the original design have already been implemented and others are planned.

Plasma Control

Due to the enhanced asymmetry of the plasma in the new divertor configuration and to the strong coupling with the divertor coils, reconsideration of essential plasma and machine control and protection was necessary. 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 is large at high plasma current and when larger shaping currents are applied, such as in single or double-null configurations. The vertical instability produces potentially dangerous forces on in-vessel elements such as protection tiles.

The previous analogue vertical stabilisation system has been replaced by one which employs a Fast Radial Field Amplifier (FRFA) and a fast digital signal processing and controller unit. The stabilisation is based on proportional feedback to the vertical speed of the current centroid, weighed with the plasma current. The average FRFA current is kept near zero on average by a slower

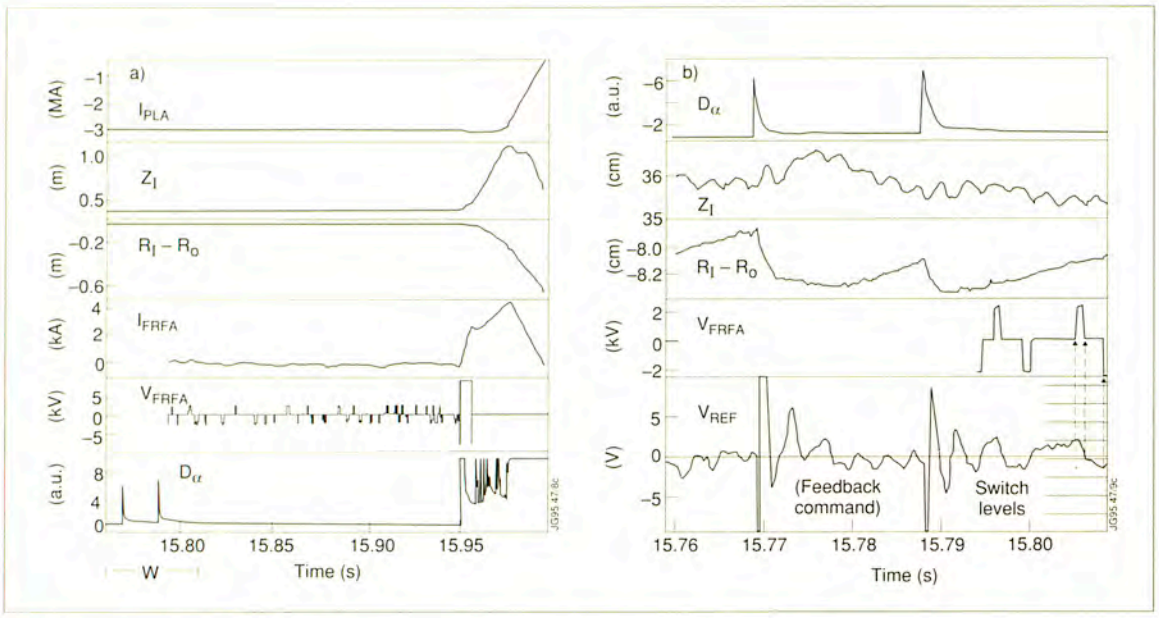


Fig.9: (a) Plasma and FRFA signals before and during a disruption; (b) D_α plasma position and FRFA signals during ELMs

proportional-integral current feedback. The system is designed to stabilise plasmas with instability growth rates up to $1000s^{-1}$.

After first operation with plasma, a number of modifications had to be made. The switching frequency was unexpectedly high and the highest switching level ±10kV was often invoked at a high rate, causing "sparkling" at in-vessel components and poor plasma behaviour. This problem was due to larger interference at 600Hz caused by the divertor coil power supplies on the plasma feedback signal, and by the high noise sensitivity of the original FRFA hysteretic controller.

The interference was suppressed by abandoning compensation of the feedback signal for the unwanted contribution from the divertor currents and also by omission of the signal contributions from the lower set of poloidal field pick up coils. The FRFA hysteretic controller algorithm was altered to avoid multiple switching by noise and to achieve a closer similarity with a linear voltage control. Hysteresis is now set to one switch level (2.5kV), independent of the instantaneous output level. Practically all disruptions and also giant ELMs caused a voltage saturation and a current limit trip of the FRFA, leading to a vertical instability and large vertical forces on the vessel. Figure 9(a) illustrates a typical case of sudden saturation, probably caused by a giant ELM (at t=15.95s). Stabilisation is not regained and the plasma becomes unstable and disrupts. Figure 9(b) indicates the response on smaller ELMs.

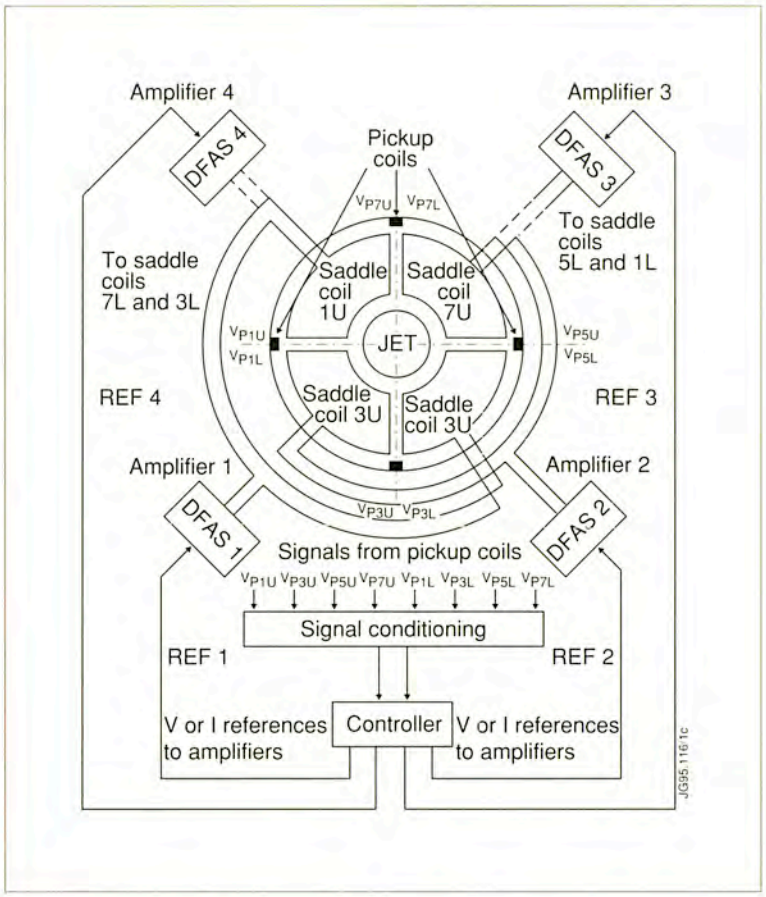


Fig.10: Schematic of disruption feedback stabilization system

The feedback signal VREF shows synchronous perturbations even though the vertical current centroid location z_I remains almost constant as opposed to the radial position R_I. This behaviour suggests that the resaturation of the originally intended plasma current derivative method could be advantageous. As an alternative, the implementation of soft X-ray cameras is proposed for feedback stabilisation. This measurement should be free from magnetic noise and insensitive to ELMs.

Disruption Feedback Stabilization System

The aim of the disruption stabilization system, shown schematically in Fig.10, is the detection and suppression of the m=2, n=1 MHD modes, which are commonly observed to precede, and ultimately cause, major disruptions in tokamaks. While extension of the tokamak operating regime by suppression of disruptions would be a major achievement, the capability of working more closely to the existing operational limits with a greater margin of safety would, in itself, represent a significant advance. Thus, the system is potentially

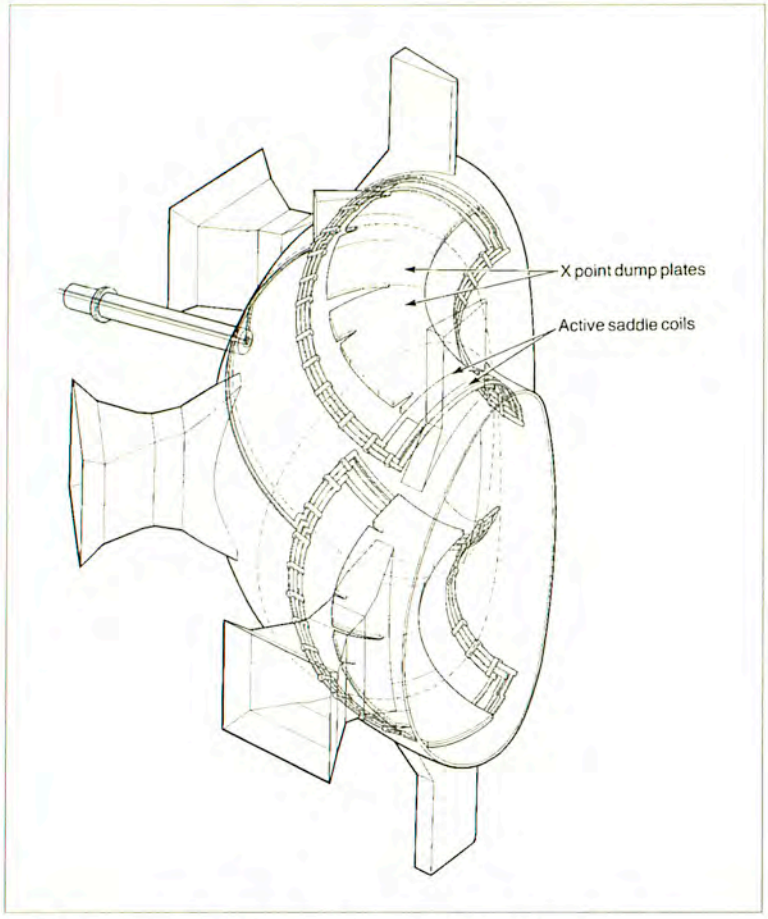


Fig.11: Overview of saddle coils in one Octant of the vessel

of great value to tokamak operations, in particular, in the context of steady-state operation of a reactor, where a major disruption may have both technical and economic consequences.

The disruption feedback stabilization system consists of:

- four disruption feedback amplifiers (DFAS) in the range 0-10kHz;
- eight saddle coils in the vacuum vessel;
- saddle coil protection system, including saddle coil crowbars (SCC);
- disruption feedback controller (DFC) with associated magnetic detection system.

Installation of the eight saddle coils was completed during the shutdown. Each saddle coil covers one quadrant of the vacuum vessel in toroidal direction and approximately 60° in poloidal direction, as shown in Fig.11, which shows a schematic overview of part of the system.

The disruption feedback controller, which is the key component in the system, receives input from four magnetic pick-up coils inside the torus, from which the amplitude, phase and frequency of the precursor are determined. The necessary magnetic correction signals

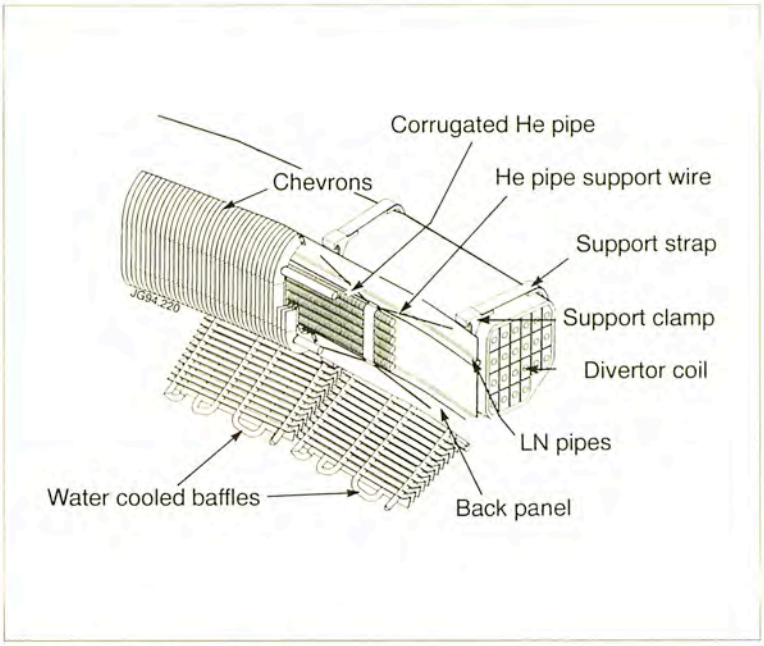


Fig.12: Divertor cryopump

are then calculated and control signals sent to the DFAS, which operates in current control. This produces an appropriate magnetic correction field, via the saddle coils, to stabilize the mode growth.

Commissioning of the system is underway and initial open loop power experiments have been performed in the presence of plasma. Although the principal aim of the system is stabilization of disruption precursors, a wide range of further experiments is possible. Non-rotating modes due to error fields are of major concern in the design of ITER and did limit the operation regime of JET in the past. The DFSS system will allow the physics of such modes to be investigated in detail. Several aspects of the physics of the m=2, n=1 mode can also be addressed. In addition, the influence of error fields and MHD activity on the core and edge plasmas can be investigated.

Cryopumps and Cryoplant

An integral part of the divertor is a large cryocondensation pump shown schematically in Fig.12, which extends over the full toroidal length of the outermost divertor coil, and which also acts as the mechanical support.

The supercritical helium loop used to supply the pumped divertor cryopump in shown schematically in Fig.13. Prior to connection to the in-vessel pump, the loop and the new refrigerator were commissioned "off-line" using a spare quadrant of the cryopump

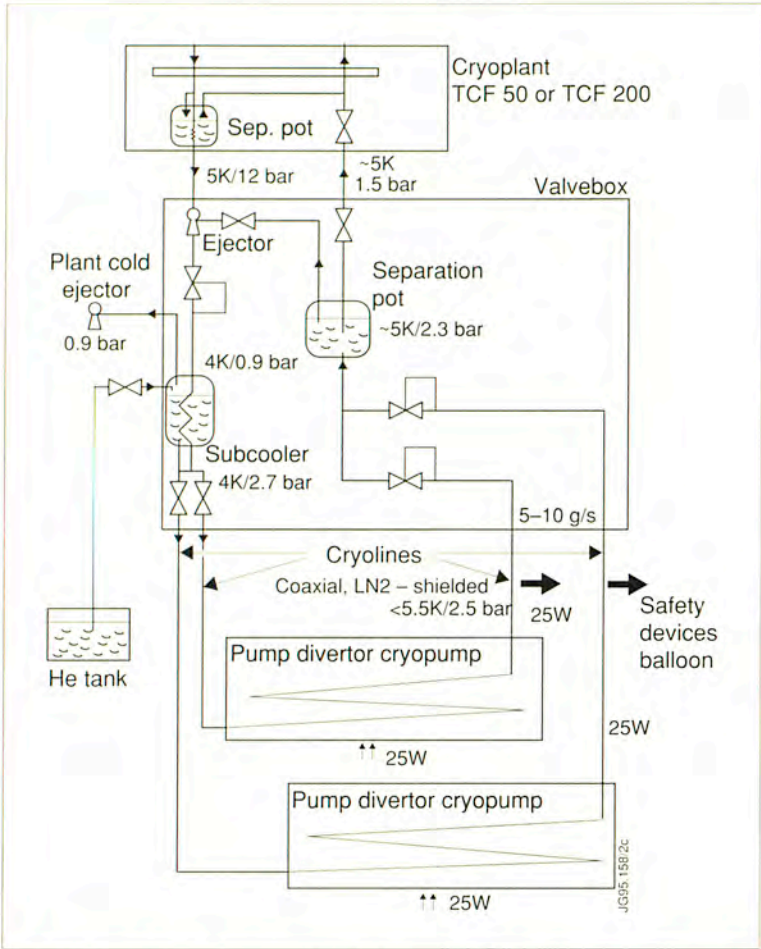


Fig.13: Schematic of supercritical helium loop for divertor cryopump

installed adjacent to the cryoplant. The in-vessel pump was connected to the cryoloop in July and, following a relatively short period of commissioning, entered routine operation, with an effective pumping speed of $2 \times 10^5 l/s^{-1}$ for the divertor structure plus the cryopump. In operation, the pump has proved to be resilient to all plasma operation scenarios and is unaffected by plasma disruption.

Although helium gas is not pumped by cryocondensation, it can be pumped using cryosorption onto a pre-deposited layer of argon frost. This technique has previously been used successfully in the neutral beam injectors and will be applied to the torus cryopump using an argon spray system integrated into the liquid nitrogen cooled chevron structure of the pump. Commissioning of the dedicated argon gas introduction system is well advanced.

Following preliminary operation to confirm satisfactory performance, a Lower Hybrid Current Drive (LHCD) pump was installed on the torus at the beginning of the year and brought into routine operation by mid-

1994. The pump has proved to be a necessary prerequisite to obtaining high power operation of the LHCD system. The performance of the pump, measured during the limited time available for acceptance tests, conforms to the design values, with a pumping speed near $10^5 l/s^{-1}$. The pump has demonstrated continuous and stable operation for periods of several weeks, including overnight helium glow discharge cleaning.

Neutral Beam Heating

During the major shutdown, the neutral beam systems had been modified and upgraded. Some injectors had been converted to the new high current 80kV configuration originally developed for the Preliminary Tritium Experiment. Eight of these were installed on the beam system at Octant No. 4. The Octant No. 8 beam system operates at 140kV. The injectors were recommissioned during the early part of the year and brought up to full power. The high voltage injector routinely delivered up to 7.8MW to the plasma (~5% above the original design value) whereas the injected power level of the high current injector was progressively increased up to 11.5-12MW (50% above the original design value).

Commissioning also took place of the new remotely operated vertical and horizontal steering systems for each of the positive ion neutral injectors (PINIs) which provide the eight beams integrated within each injector. The new vertical steering facility allowed the power deposition profiles to be optimised for the upwardly shifted plasma equilibria, characteristic of the new pumped divertor.

In addition, the new extended protection assemblies for both the neutral beam drift injection ducts were commissioned together with their extensive thermocouple instrumentation. This extended protection allowed safe injection of neutral beams down to plasma currents of ~1MA, whereas in previous campaigns, injection had been limited to $\geq$1.5MA. This proved invaluable in high β_p experiments.

These tasks were accomplished successfully and the neutral beam system operated on nearly every day of the 1994 campaign. The overall statistics for power injected into JET with the neutral beam system are shown in Fig.14. The system was more heavily used in 1994 than any previous year and the proportion of pulses at high power (one quarter of all pulses) exceeded those in previous campaigns. Figure 14 shows that the spectrum of injected power was nearly uniform up to 16MW,

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 were designed to operate at accelerating voltages up to 80kV and in 1990 one system was substituted with units capable of operating up to 140kV. In addition, this box was also converted to operate with helium (^{3}He and ^{4}He) 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 12MW of power can be directed.

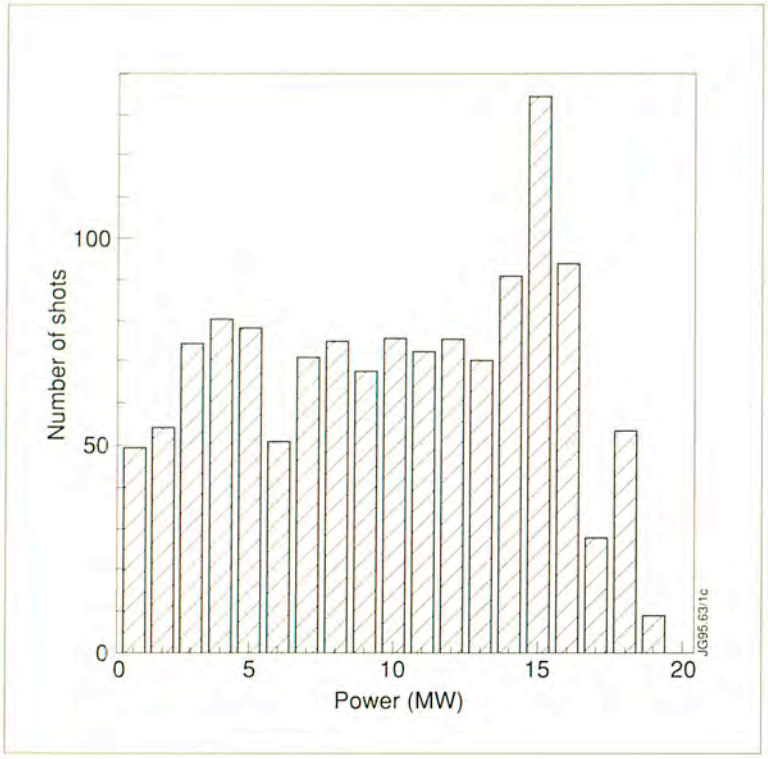


Fig.14: Histogram of injected neutral beam power during 1994

reflecting the wide variety of experiments, in which the system was used. Above 16MW, the number of pulses was reduced as the system only attained these power levels reliably from October onwards.

Radio Frequency Heating

The ion cyclotron resonance frequency (ICRF) heating system is used for high power centralised heating of the plasma, with increased emphasis on Fast Wave Current Drive (FWCD) studies with the new antennae. The localisation depends mainly on the magnetic field and is insensitive to parameters such as density and temperature. Wide-band operation (23-57MHz) allows variation in both the choice of minority ion species heated (H or He^3 at present, D in the future D-T phase) and the localised position of the heating. With the previous antennae, up to 3.5MW on one antenna and 22.7MW total coupled power for 2s were achieved.

New ICRF antennae have been installed and optimised to the geometry of the divertor plasmas. Their location in the torus has been revised to give four arrays of two adjacent antenna. Each array has four RF radiating conductors (see Fig.15) or straps, which provide an enhanced radiated spectrum. Variation in the relative phase of the RF currents in the straps allows this spectrum to be varied for both heating

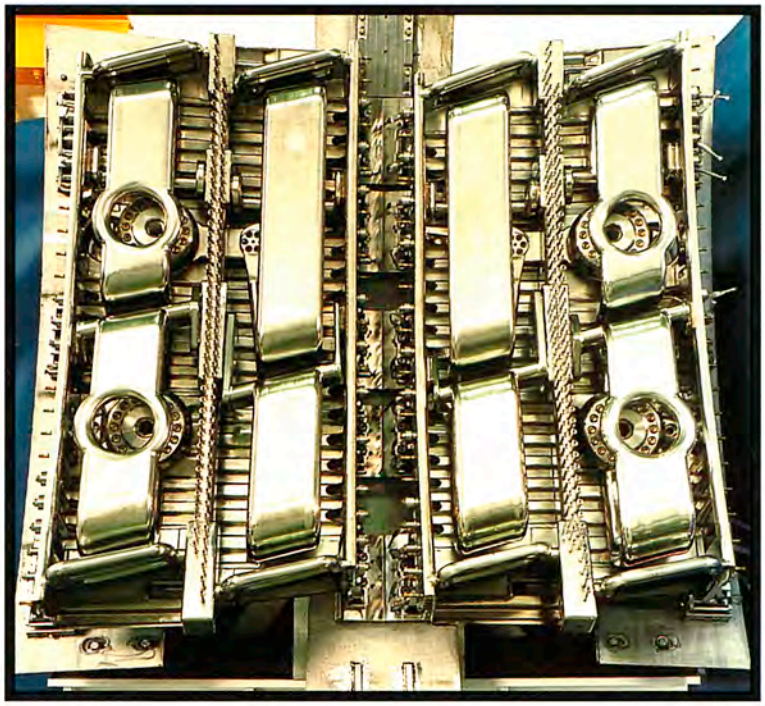


Fig.15: The new ICRF antenna

and current drive experiments. In addition, the control electronics have been completely rebuilt to allow operation with four straps closely coupled and to improve the reliability of the ICRF plant by reducing the cross-talk between modules. The start of operations was made difficult by the fact that both the control electronics and the antennae are new systems and have required considerable commissioning time. Several problems have been identified and remedial actions has been taken.

Two major improvements have been made to the power transmission. Longer matching trombones have been added to many of the transmission lines to cope with the requirements of tuning four antenna straps, and conjugate box decouplers are installed on three modules to compensate for power coupled across straps of an antenna when phased for current drive operation. In spite of early difficulties, a record power of 13.5MW had been launched in a divertor plasma, compared to 12MW in X-point plasmas in the preceding campaign.

Lower Hybrid Current Drive

The Lower Hybrid Current Drive (LHCD) system, operating at 3.7GHz, is capable of driving a significant fraction of the toroidal current in the plasma. This is achieved by launching an RF wave predominantly in one toroidal direction. This wave accelerates the high energy electrons in the plasma and so drives a current. It may be used to stabilise sawtooth

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. The eight RF generators produce a maximum output power of 32MW. The net power coupled to the plasma has reached 22.7MW, compared with theoretical limit of 24MW.

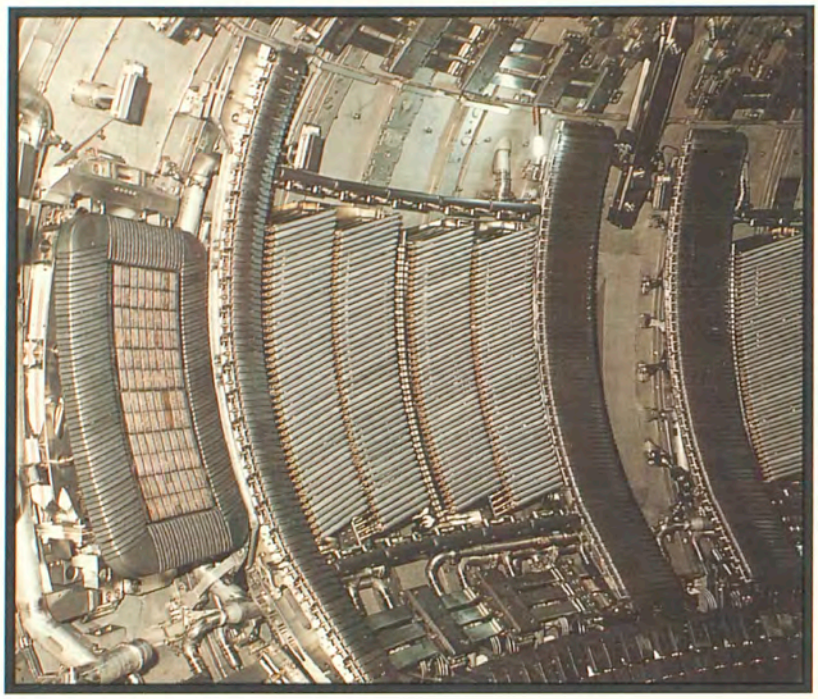


Fig.16: The LHCD launcher (left) with the ICRF antenna (right)

oscillations, thereby increasing central electron temperatures. It is this system for controlling the plasma current profile which is considered to be the main tool to stabilise high beta poloidal plasmas with a large bootstrap current (the so-called advanced tokamak scenarios).

With a prototype launcher, up to 2.3MW of lower hybrid waves have been coupled to JET plasmas. The Lower Hybrid generator consists of 24 klystron amplifiers. The total generator power is 15MW in pulses up to 10s and 12MW up to 20s. The power is transmitted from the generator through waveguides to a new upgraded LHCD launcher, which is installed on the main horizontal port of Octant No:3. In the launcher, the power is progressively split to feed a phased array of 384 waveguides at the grill mouth facing the plasma. The front end of the launcher as seen from inside the vessel is shown in Fig.16. Three vertically superposed areas of the waveguide array can be phased independently. This allows for high flexibility in the wave spectrum composition.

The control system for the generator includes a number of automatic software routines to simplify operation of the plant. One of these is used to condition the launcher vacuum waveguides, without plasma load. This routine pulses the generator modules at the maximum rate allowed and automatically adjusts the power to the maximum that the launcher can handle. A second software routine optimises the power within an LH pulse into plasma to the

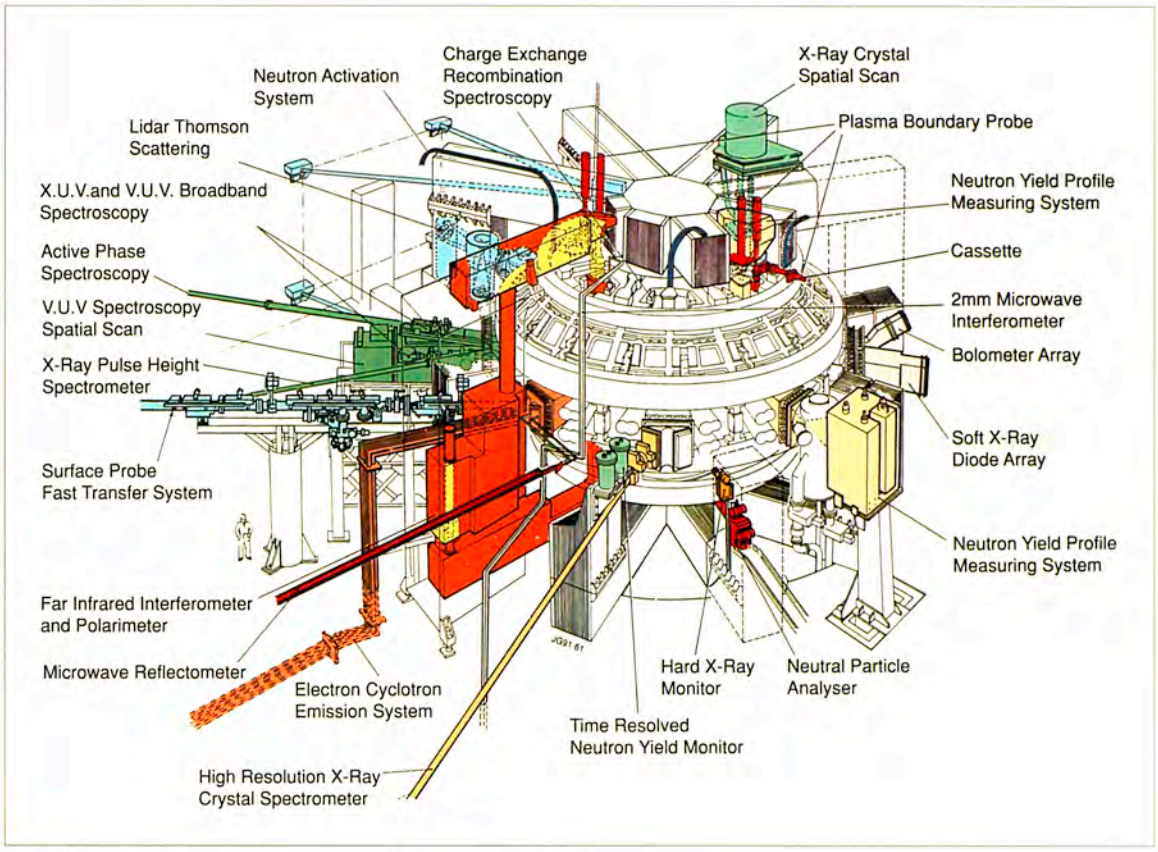


Fig.17: General layout of diagnostics in the machine

maximum possible by ramping up the power until there is an arc in the launcher. The power is then removed for a pre-set time and reapplied at a level 20kW below that which caused the arc.

Full current drive conditions at 2MA plasma current were achieved in the first dedicated LH experiments in mid-year, and at 3MA with about 6MW LH power coupled towards the end of the year. The variation of LH power deposition profiles was studied with extensive code modelling of the experimental data in preparation for profile control experiments. Strong broadening of the current profile was obtained with off-axis LH current drive (LHCD). In addition, sawteeth were stabilised with near-axis LHCD. Performance can still be increased up to, the maximum design launched power of 10MW.

Diagnostics Systems

The status of JET's diagnostic systems at the end of 1994 is summarized in Tables 4 and 5 and their general layout in the machine is shown in Fig.17. The staged introduction of the diagnostic systems onto JET has proceeded from the start of

SYSTEM	DIAGNOSTIC	PURPOSE	ASSOCIATION
KB1	Bolometer array	Time and space resolved total radiated power	IPP Garching
KC1	Magnetic diagnostics	Plasma current, loop volts, plasma position, shape of flux surface, diamagnetic loop, fast MHD	JET
KE3	Lidar Thomson scattering	T_e and n_e profiles at 4Hz and 8Hz for short periods	JET and Stuttgart University
KF1	High energy neutral particle analyser	Ion energy distribution up to 3.5MeV	Purchased from Ioffe St Petersburg
KG1	Multichannel far infrared interferometer	$\int n_e ds$ on four vertical chords and four horizontal chords	CEA Fontenay-aux-Roses
KG3	O-mode microwave reflectometer	n_e profiles and fluctuations	JET and FOM Rijnhuizen
KG4	Polarimeter	$\int n_e B_p ds$ on eight chords	JET and CEA Fontenay-aux-Roses
KH1	Hard X-ray monitors	Runaway electrons and disruptions	JET
KH2	X-ray pulse height spectrometer	Monitor of T_e, impurities, LH fast electrons	JET
KK1	Electron cyclotron emission spatial scan	$T_e(r,t)$ with scan time of a few milliseconds	NPL, UKAEA Culham and JET
KK2	Electron cyclotron emission fast system	$T_e(r,t)$ on microsecond time scale	FOM Rijnhuizen
KK3	Electron cyclotron emission heterodyne	$T_e(r,t)$ with high spatial resolution	JET
KL1*	Plasma viewing	Monitor plasma in visible light	JET
KL3	Surface temperature	Surface temperature of target tiles	JET
KM1	2.4MeV neutron spectrometer	Neutron spectra in D-D discharges, ion temperatures and energy distributions	UKAEA Harwell
KM3	2.4MeV time-of-flight neutron spectrometer		NFR Studsvik
KM7	Time-resolved neutron yield monitor	Triton burnup studies	JET and UKAEA Harwell
KN1	Time-resolved neutron yield monitor	Time resolved neutron flux	UKAEA Harwell
KN2	Neutron activation	Absolute fluxes of neutrons	UKAEA Harwell
KN3	Neutron yield profile measuring system	Space and time resolved profile of neutron flux	UKAEA Harwell
KN4	Delayed neutron activation	Absolute fluxes of neutrons	Mol
KR2	Active phase neutral particle analyser	Ion distribution function, $T_i(r)$	ENEA Frascati
KS1	Active phase spectroscopy	Impurity behaviour in active conditions	IPP Garching
KS3	H-alpha and visible light monitors	Ionisation rate, Z_{eff}, impurity fluxes from wall and limiter	JET
KS4	Charge exchange recombination spectroscopy (using heating beam)	Fully ionized light impurity concentration, $T_i(r)$, rotation velocities	JET
KS5	Active Balmer α spectroscopy	T_D, N_D and $Z_{eff}(r)$	JET
KS6*	Bragg rotor X-ray spectrometer	Monitor of low and medium Z impurity radiation	UKAEA Culham
KS7*	Poloidal rotation	Multichannel spectroscopic measurement of poloidal rotation	UKAEA Culham
KT2*	VUV broadband spectroscopy	Impurity survey	UKAEA Culham
KT3	Active phase CX spectroscopy	Full ionized light impurity concentration, $T_i(r)$, rotation velocities	JET
KT4*	Grazing incidence+visible spectroscopy	Impurity survey	UKAEA Culham
KX1	High resolution X-ray crystal spectroscopy	Central ion temperature, rotation and Ni concentration	ENEA Frascati
KY3*	Plasma boundary probes	Vertical probe drives for reciprocating Langmuir and surface collector probes	JET, UKAEA Culham
KZ3*	Laser injected trace elements	Particle transport, T_i, impurity behaviour	JET
Kγ1	Gamma rays	Fast ion distribution	JET

* Not compatible with Tritium

Table 4: Status of JET Diagnostic Systems, December 1994 - Existing Diagnostics

SYSTEM	DIAGNOSTIC	PURPOSE	ASSOCIATION	STATUS
KB3D	Bolometry of divertor region	Power balance of divertor plasma	JET	Operational.
KB4	In-vessel bolometer array	Time and space resolved radiated power	JET	Partially operational.
KC1D	Magnetic pickup coils	Plasma geometry in divertor region	JET	Operational.
KD1D	Calorimetry of Mark I divertor targets	Power balance of divertor plasma	JET	Operational.
KE4	Fast ion and alpha-particle diagnostic	Space and time resolved velocity distribution	JET	In installation.
KE9D	Lidar Thomson scattering	T_e and n_e profiles in divertor plasma	JET	Commissioning.
KG6D	Microwave interferometer	$\int n_e dl$ along many chords in divertor plasma	JET	Installed.
KG7D	Microwave comb reflectometer	Peak n_e along many chords in divertor plasma	JET	In installation.
KG8	E-mode reflectometer	Measurement of density fluctuations and profiles in edge and SOL	JET and CFN/IST Lisbon	In installation.
KJ3	Compact soft X-ray cameras	MHD instabilities, plasma shape	JET	Operational.
KJ4	Compact soft X-ray camera	Toroidal mode number determination	JET	Operational.
KK4D	Electron cyclotron absorption	n_eT_e profile along many chords in divertor plasma	JET	Commissioning.
KM2	14MeV neutron spectrometer	Neutron spectra in D-T discharges, ion temperatures and energy distributions	UKAEA Harwell	In installation.
KM5	14MeV time-of-flight neutron spectrometer		NFR Gothenberg	In installation.
KT1D	VUV spatial scan of divertor	Time and space resolved impurity densities	JET	Not operational, vacuum system fault.
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	JET	Waiting for installation.
KT6D	Poloidal view visible spectroscopy of divertor plasma using a periscope	Impurity influx, 2-D emissivity profile of lines	JET	Operational.
KT7D	VUV and XUV spectroscopy of divertor plasma	Impurity influx, ionization dynamics	JET	Operational.
KY4D	Langmuir probes in divertor target tiles and limiters	n_e and T_e in the divertor plasma and limiters	JET	Operational.
KY5D	Fast pressure gauges	Neutral flow in divertor region	JET	Operational.
KY6	50kV lithium atom beam	Parameters of the scrape-off-layer plasma	JET	Commissioning.
KY7D	Thermal helium beams	n_e and T_e in the divertor plasma	JET	Installed.
Kα1	Thin-foil charge collectors	Lost alpha-particle detection	JET	Waiting for installation.

Table 5: Status of JET Diagnostic Systems, December 1994 - New Diagnostics

operation in June 1983. The present status is that 48 systems are in existence. 35 of these systems were diagnostics which had existed from the 1992 campaign but had been modified or upgraded for operation in the new phase of JET or in the active D-T phase. Table 5 sets out the list of 23 additional diagnostics, which were specifically prepared for divertor operation, and shows their present status. Operational experience on the existing diagnostics has been good and most of the systems have operated automatically with minimal manual supervision. The resulting measurements have been of high quality in terms of accuracy and reliability, and have provided essential information on plasma behaviour in JET. Further details on specific diagnostics systems are given below.

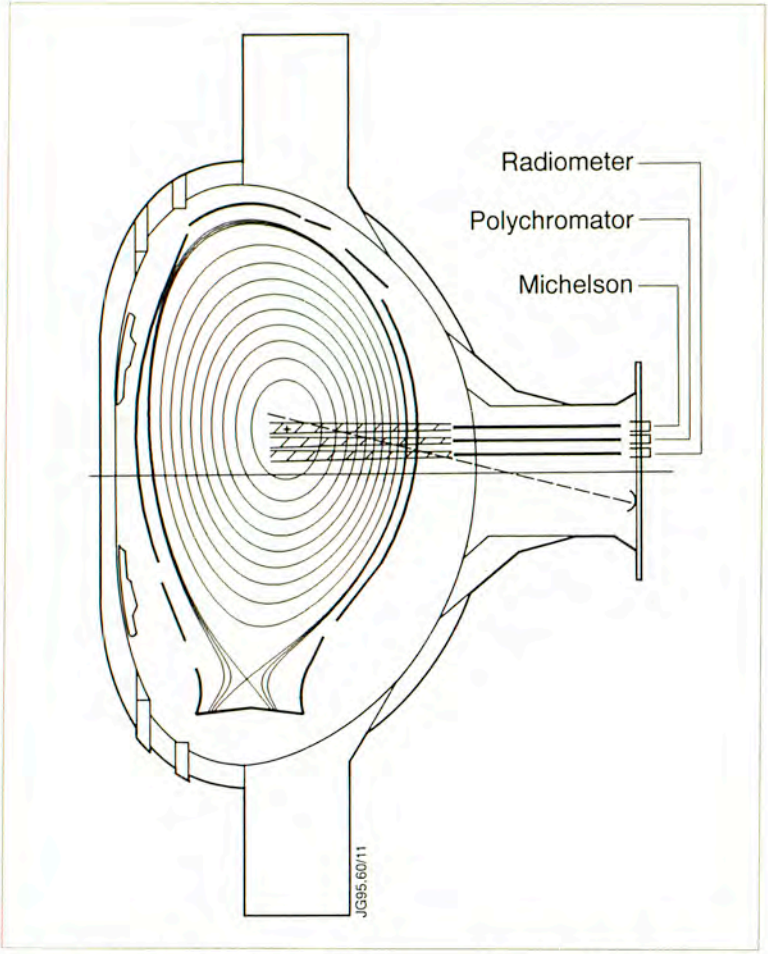


Fig.18: Poloidal cross-section of JET showing location of the horizontal ECE sightlines of the conventional antennae, and the near-horizontal sightline of the quasi-optical collection system. The sightlines employed by the different ECE measurement instruments are marked

Temperature and Density Measurements

Major modifications to the electron cyclotron emission (ECE) system were made during the shutdown to adapt the diagnostics to the new configuration and to further enhance the measurement capability. In 1994, the three different instruments in the ECE system (Michelson interferometers, the grating polychromator and the heterodyne radiometer) again provided electron temperature data on most pulses. At the interface with the machine, new antennae to adapt to the new plasma configuration and a quasi-optical collection system for the heterodyne radiometer were installed. All systems now view the plasma through double vacuum windows.

A new antenna array which provides three horizontal sightlines passing close to the new plasma centre (typically ~0.3m above the vessel midplane) replaces the old fan-shaped array which had sightlines

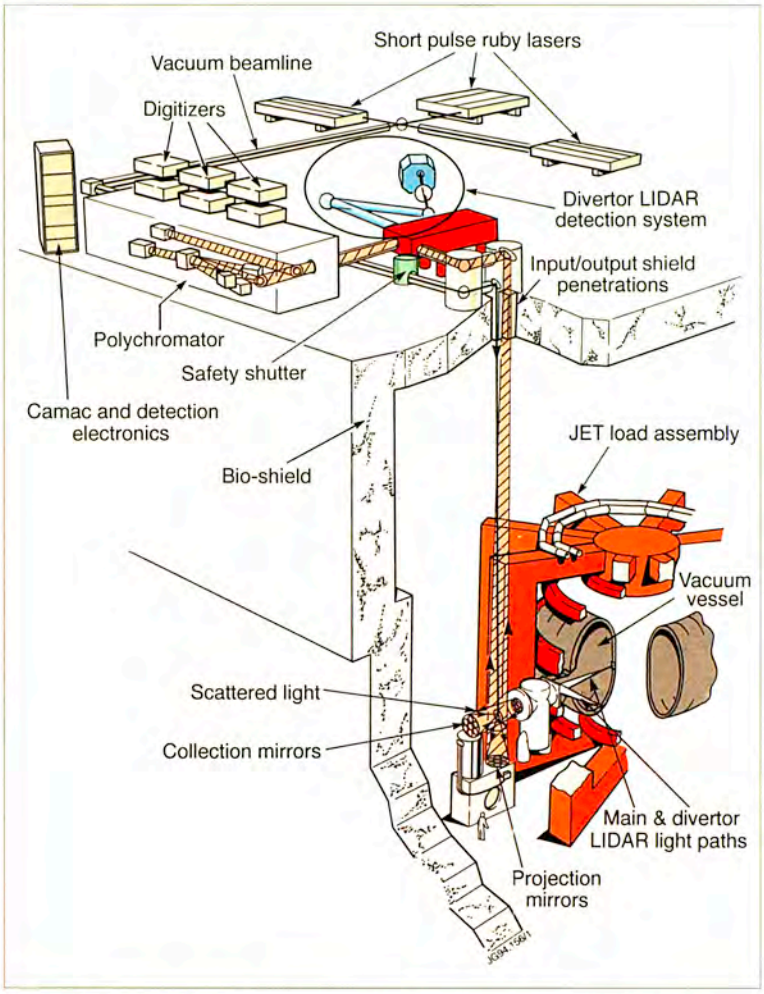


Fig.19: Cutaway section of the LIDAR Thomson Scattering System

spread across the whole poloidal cross-section. There is also one obliquely viewing antenna directed towards the plasma centre. The original array of four horizontally viewing antennae has been retained. To realise the high resolution capability of the heterodyne radiometer, a Gaussian beam (or quasi-optical) collection system, employing two mirrors inside the vacuum vessel and scalar feed horns outside, was also installed. This give a small antenna spot size in the plasma (~0.08m diameter) which minimises the effect of flux surface averaging of temperature measurements. Figure 18 shows the locations of the most important sightlines now installed.

Installation and testing of all the new components for the main LIDAR system were completed on time ready for integrated commissioning (see Fig.19). The first successful LIDAR Thomson scattering profile measurements at the new standard rate of 4Hz throughout a pulse were obtained in March. Figure 20 shows an

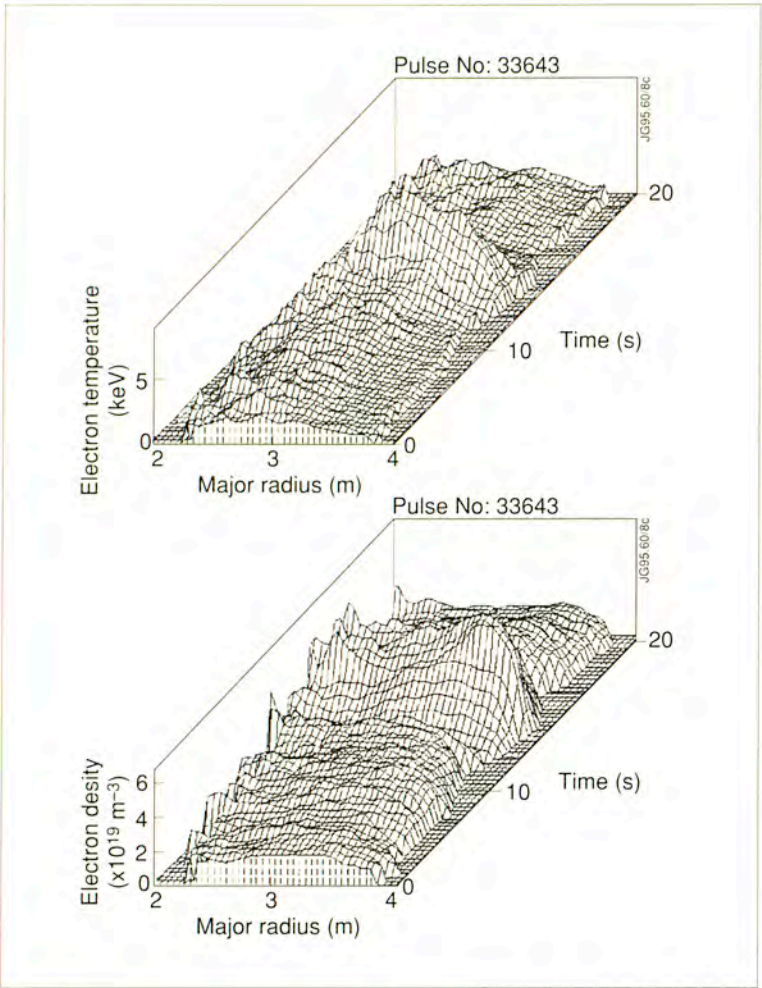


Fig.20: Temperature and density profiles obtained by the LIDAR System

example of LIDAR electron temperature and density profiles during a recent pulse.

During the year, the diagnostic has also demonstrated an 8Hz repetition rate capability of the system for a 7s burst during a pulse. To do this, the layout of the laser beam paths were reorganized in the Roof Laboratory. By using a dielectric polarizer as a beam combiner and a pulsed magneto-optic polarization rotator to correct the polarization, the beams of both the available 4Hz lasers were combined along the main LIDAR path. Better matching of the pulse energy between the two lasers is required to make this scheme a regular option.

A new LIDAR Thomson Scattering system was installed during the shutdown for density and temperature measurements in the divertor region. The system employs in-vessel mirrors to deflect the laser beam and the collected light from the divertor region out through the main horizontal port of Octant No.5. The silver coated mirrors failed during

the final thermal cycling tests and new aluminium coatings had to be applied to the substrates. The reduced reflectivity of aluminium relative to silver combined with the fitting of double windows on the vacuum vessel unfortunately meant halving of expected signal.

Alignment of the system has proved a problem. The in-vessel mirror system does not allow a direct view of the divertor structure outside the very narrow field of view of the actual collection area. When the laser strikes the divertor target, not only a large stray light pulse but an intense white light signal is observed. The combination of poor transmission, alignment difficulties and "wall pulse" has prevented plasma measurements, so far.

Three microwave diagnostics share a single set of waveguides and antennae for divertor plasma measurements. These are a dual frequency interferometer for line-integrated electron density measurements, a "comb" reflectometer for estimating the peak density along a sightline and an electron cyclotron absorption diagnostic for the determination of the local electron temperature-density product (the electron pressure).

An Electron Cyclotron Absorption (ECA) diagnostic is being developed, as it is not possible to determine the electron temperature from electron cyclotron emission measurements in the optically thin divertor plasma. The plasma is not sufficiently opaque to re-absorb the intense ECE from the core plasma, which radiates at the same frequencies. By measuring the absorption of radiation from an external source, the optical depth will be determined directly. From this measurement, the electron density-temperature product (the electron pressure) will be deduced. The diagnostic will measure the attenuation by the divertor plasma of radiation from swept frequency microwave sources. Commissioning the system has been delayed by problems with the supply of the Backward Wave Oscillator source, which is a key component of the system. It is expected that the diagnostic will be able to measure the spatial profile of the n_eT_e product in the range 2.5×10^{20} to $5\times10^{21}m^{-3}eV$ at most values of toroidal field, with an accuracy of ±20%, on a timescale of ~1ms.

Boundary Measurements

Langmuir probes are one of the oldest plasma diagnostics and still play a crucial role in diagnosing the plasma boundary. In JET, there are 39

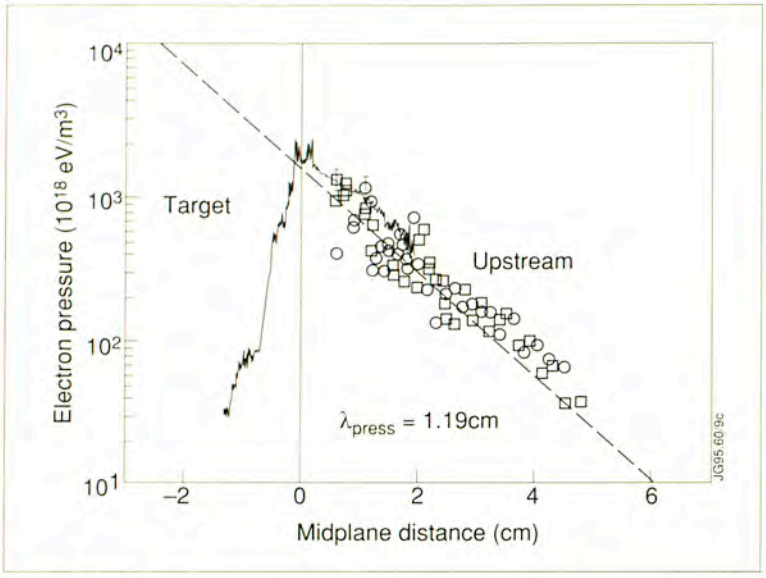


Fig.21: Comparison of electron pressure profile from the reciprocating probe at the machine top with electron pressure profile measured with a triple probe in the outer divertor leg, during a sweep of the strike point

triple and 32 single Langmuir Probes in the divertor and a further 22 single Langmuir probes in the limiters. This system is fully operational and due to its self testing and configuring, it is highly autonomous. The triple probe system can measure electron density and temperature at up to 5kHz, which is essential for the study of ELMs and discharges where the strike zones are swept across the divertor.

In the scrape-off layer (SOL) at the top of the machine, two reciprocating Langmuir probe systems provide measurements of the electron density, electron temperature, electron pressure, plasma flow velocity and floating potential. This data provides the best indication of separatrix parameters, which are crucial boundary conditions for modelling the SOL plasma. Figure 21 is an example of the plasma pressure recorded with the reciprocating probe. Also shown is the electron pressure profile in the divertor from a triple Langmuir probe, which had the outer strike zone swept across it. This has been mapped around the magnetic surfaces to the same coordinate system as the reciprocating probe. One of the reciprocating probe systems has an exchange chamber, which allows collector probes to be mounted. These have been extensively used to monitor beryllium evaporations and metallisation during glow discharge cleaning.

Since the new plasma configurations are further away from the original set of magnetic sensors, a new set of magnetic probes, saddle loops and full flux loops were installed inside the vessel to maintain the

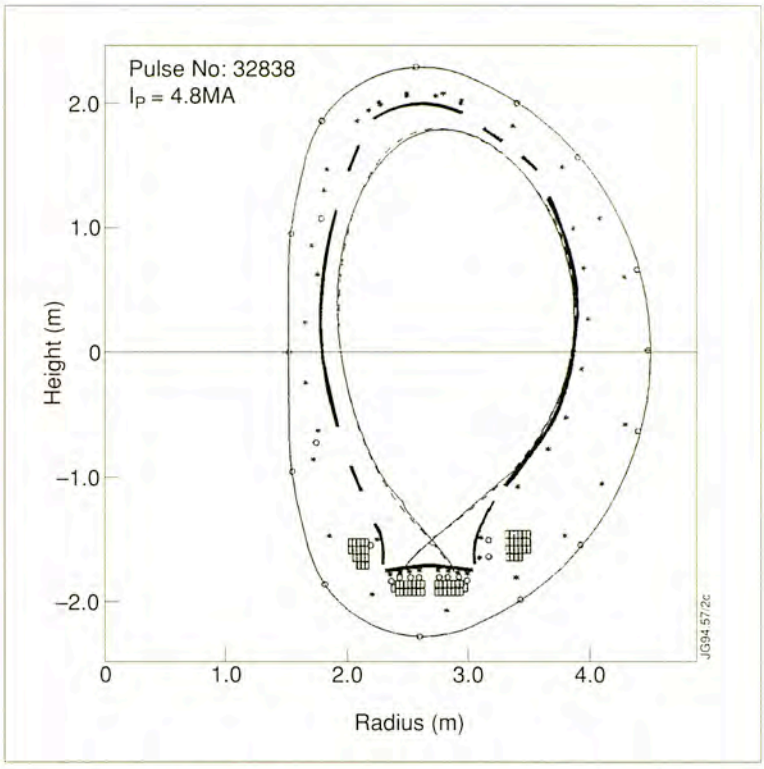


Fig.22: Real-time display of the plasma boundary of a 5MA discharge. Agreement with a full calculation by EFITJ is indicated by the dotted line

boundary reconstruction accuracy. The new sensor configuration was designed using simulated plasmas to assess the reconstruction accuracy. Operation has demonstrated boundary reconstruction in the new configuration, accurate to within a few centimetres using the EFIT and XLOC codes (see Fig.22). The main diagnostics used in verifying the calculated boundary position are the divertor Langmuir probes, reciprocating probe and plasma viewing cameras.

The XLOC code is also used in the real-time plasma position and current control system (PPCC) to control the current in the coils, which shape and position JET plasmas. Reliable magnetic data is essential for safe operation of JET. Therefore, data quality is monitored by post-pulse validation software, which warns of potential errors in the magnetics data. A minimal set of analogue signals is also connected to the PPCC system for safe termination of the pulse in the event of loss of digital communication through which the magnetic data is received.

Neutron Measurements

The most important diagnostic for the assessment of the fusion performance of the tokamak is the time-resolved neutron yield monitor comprising three pairs of fission chambers arrayed around the

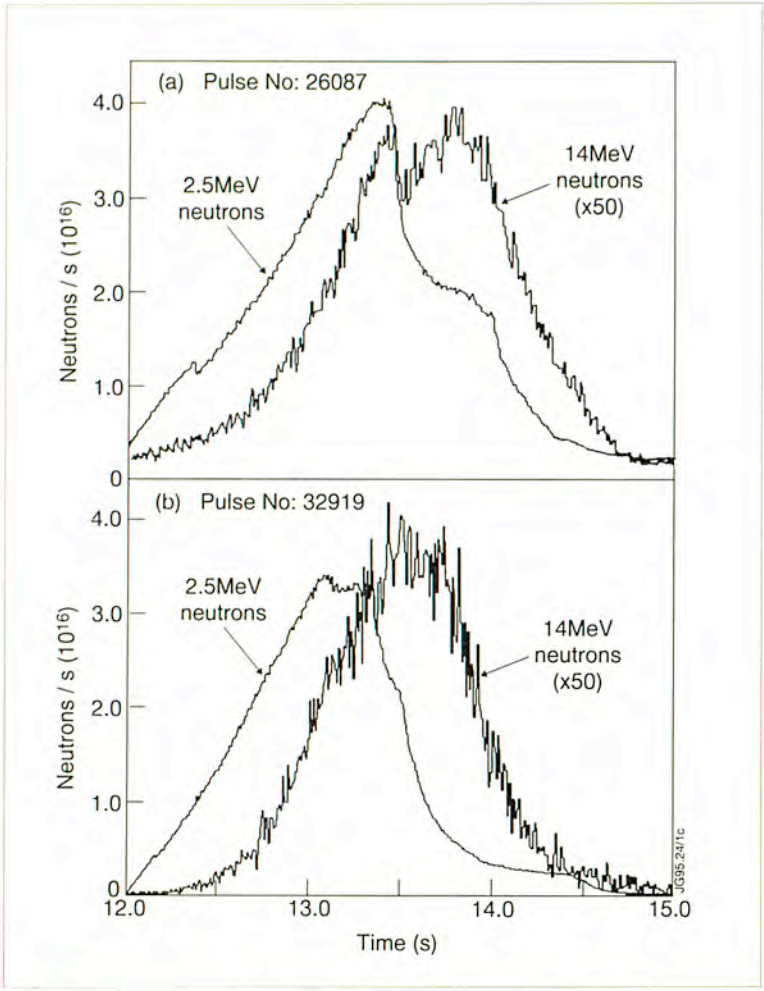


Fig.23: Comparison of 2.5MeV neutron emission with the resulting 14MeV emission from triton burn-up in D-D discharges. The 2.5MeV neutron time-traces are broadly similar during their ramp-up phases, although their terminations are accompanied by different phenomena, a carbon bloom for Pulse No:26087 and ELMs for Pulse No:32919. The effect of carbon influx is noticeable on the 14MeV neutron time-trace for Pulse No:26087. The 14MeV neutron signals have been multiplied by 50

tokamak. This is a highly reliable diagnostic for relative measurements but its absolute calibration must be determined anew when changes are made to hardware within and surround the machine. Experience of changes has been acquired over the years that new calibrations can now be predicted with some confidence. For the new divertor configuration, many changes were made. However, the divertor itself was not expected to affect the fission chamber calibrations. The predicted 2.5MeV neutron calibration for 1994 operations has been confirmed with measurements of neutron activation using the pneumatic transport system. Figure 23 shows a comparison between recent measurements and those of the previous campaign.

The neutron profile monitor provides detailed information on the fusion performance of JET plasmas that cannot easily be obtained by other means. The profile monitor is also furnished with alternate detector boxes, appropriate to the study of bremsstrahlung emission from fast electrons (FEB) accelerated by the lower hybrid current drive launcher. The FEB diagnostic is regarded as essential for LHCD studies. Due to this dual function for the profile monitor, the detector boxes have to be exchanged according to the needs of the programme. As a result, the neutron camera was not always in place when many interesting discharges were run. This upgraded profile monitor is designed to provide simultaneous acquisition of FEB and neutron data.

Considerable effort has been devoted to construction and fitting-out of the new profile monitor. It will use separate detectors for 2.5MeV neutrons, 14MeV neutrons and bremsstrahlung radiation and will possess remotely changeable collimation to provide coarse optimization of rate of detected events for differing plasma conditions. The existing profile monitor will be used until the new one is installed in early-1995.

Impurities

Three spectrometers were used to record the VUV, XUV and soft X-ray emission from the bulk plasma during 1994 operations; a crystal spectrometer and two grating instruments. These spectral regions are of particular value since they permit characteristic spectral lines to be monitored of all the intrinsic impurities found in JET plasmas. The Bragg crystal spectrometer consists of four scanning monochrometers. A balance must be made between the temporal resolution and the number of lines accessed. Initially, a time resolution of ~400ms was used and seven lines were routinely monitored. With increasing availability of spectroscopic data, a much improved temporal resolution of ~80ms was adopted with five lines being recorded. The SPRED spectrometer is a VUV, grating instrument providing coverage of the wavelength region from 100 to 1100Å with a spectral resolution of ~3Å (see Fig.24). The XUV, SOXMOS instrument is a grazing incidence spectrometer with two detectors. Each views a wavelength range of ~40Å and can be moved to record the spectrum at positions between 15 and 340Å.

A new VUV/XUV diagnostic has now been installed to observe line radiation of the main plasma impurities emitted from the divertor region. The instrument consists of three spectrometers, a double

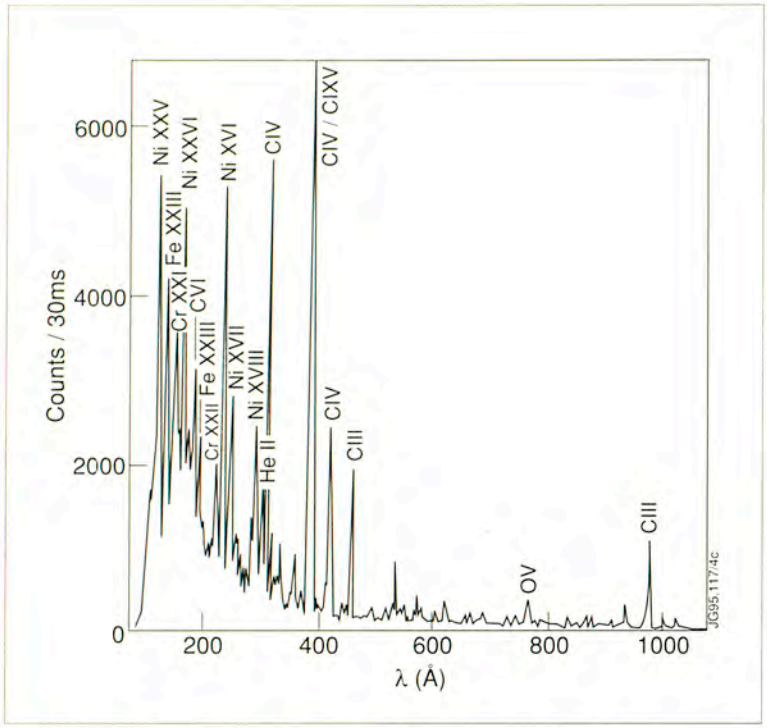


Fig.24: Typical VUV spectrum from the SPRED instrument

SPRED, which combines two VUV spectrometers in one instrument and a SOXMOS for the soft X-ray emission. All systems started to produce spectra on a regular basis at the end of 1994, although the SOXMOS only produces a weak signal. The reason for this is still under investigation.

Remote Handling

A comprehensive suite of remote handling equipment is being prepared to support operations during periods when man-access to the machine is not possible. During and immediately after periods of D-T operation, parts of the machine will become activated to an extent, where man-access is severely limited and, under these conditions, all maintenance and remedial work in the restricted areas will be performed by trained operators utilising remote handling equipment.

The remote maintenance philosophy is based on the principle that maintenance tasks must be undertaken by trained operators using remote handling machines. The operators position and deploy the equipment so that remote maintenance becomes a direct extension of the 'hands-on' maintenance operations. The inbuilt intelligence and adaptability of the human operator is retained and is aided and enhanced by the robotic devices, which provide force

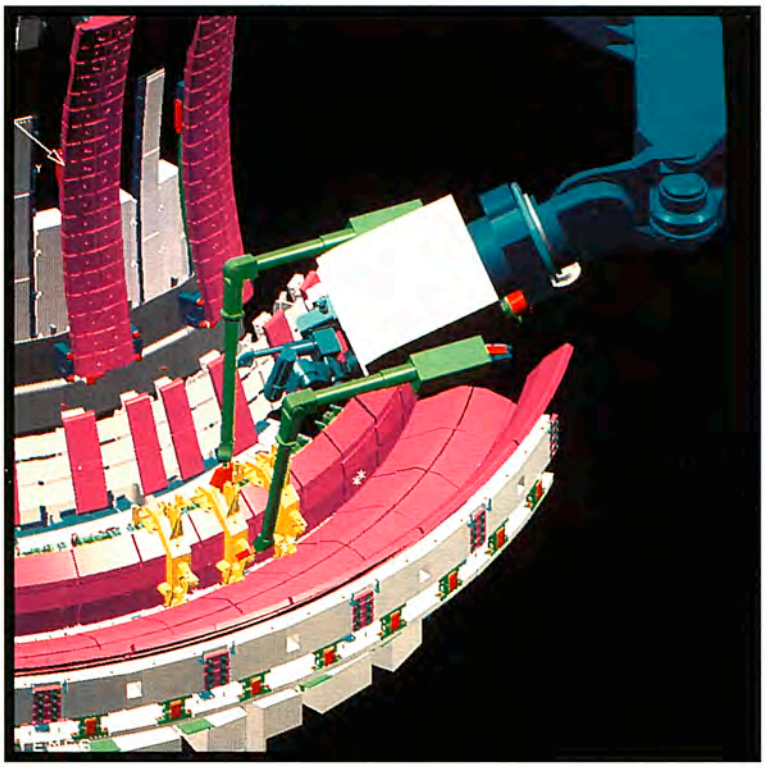


Fig.25: Computer generated image of Mascot servo-manipulator attached to the Articulated Boom performing in-vessel tasks

reflection from the work face and include functions, which enable the equipment to learn and repeat particular motions and procedures.

This approach has enabled the overall remote handling equipment development to be minimally affected by the continual modification and development of JET since 1983. During 1992/3, a complete reconfiguration and replacement of hardware took place within the torus. The only significant effect on the remote handling system equipment was in the area of specialised tooling where development of new cutting, welding and handling tools was required. All other major elements of the remote handling system remained unaffected.

During 1994, the Remote Handling Articulated Boom and special tooling were used to facilitate the final phase of Mark I Divertor installation, including installation of the Mark I divertor modules and the welding of the divertor module cooling pipework in-vessel.

In the rest of 1994, all effort within the Remote Handling Group was focused on preparing for the installation of the Mark IIA divertor system during 1995 and the fully remote exchange of Mark IIA tile carriers by the Mark IIB tile carriers planned immediately after DTE1. This remote tile exchange will be the first fully remote task undertaken on the machine and will also be the most significant fully remote handling task

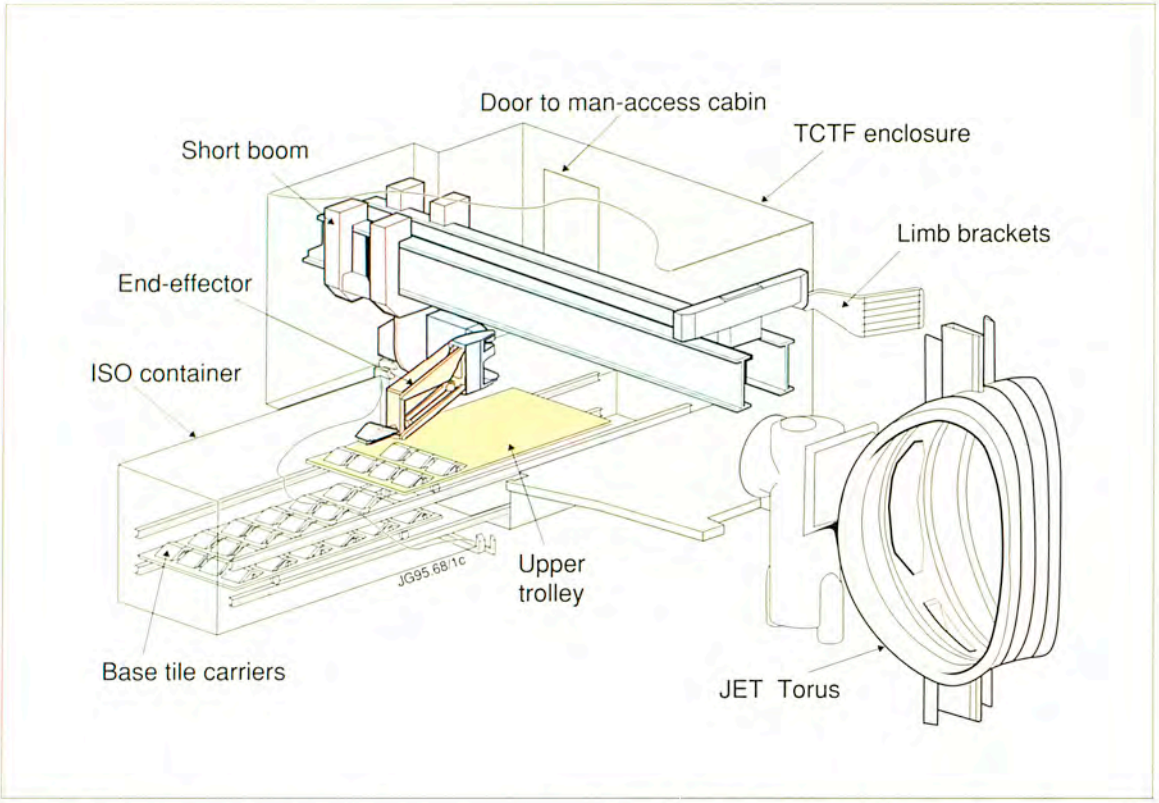


Fig.26: Octant No:1 Facility designed to transport tile carriers and tools fully remotely into and from the torus

undertaken on any fusion device in the world. Accordingly, extensive preparations are being made for the proving of the equipment function, performance and reliability together with the derivation and development of task techniques and operator training using full scale mock-ups. The operations will be performed in-vessel by use of the Mascot servo-manipulator mounted on the end of the Articulated Boom entering from Octant No:5 (see Fig.25). Special tools will be used to handle the tile carriers and all components will be transferred into and out of the torus through Octant No:1 main horizontal port.

The remote tile exchange concept requires the transfer of tile carriers and tools into and from the torus through the Octant No:1 main horizontal port. This transfer task and the placing of tile carriers in a transport container will be carried out fully remotely by means of a new Tile Carrier Transfer Facility which will be installed and operated at Octant No:1 (see Fig.26). The facility comprises a so-called short boom with an end effector suitable for carrying the various tile carriers and tools housed within a sealed enclosure for contamination control but with interface doors for a man-access module and tile carrier transfer containers. Concept design for the

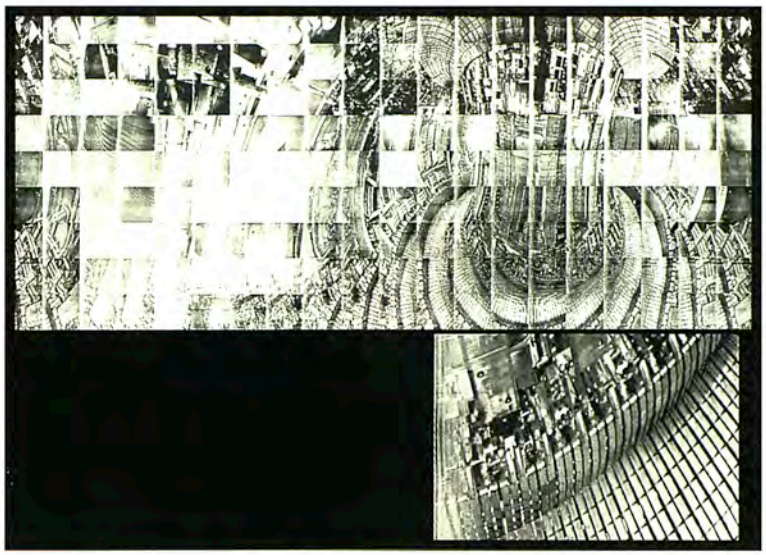


Fig.27: Picture mosaic of the inside of the JET vessel

whole facility has been completed and a contract for detail design and manufacture of the short boom has been placed.

The new In-Vessel Inspection System (IVIS), installed during the major shutdown, has been in operation since January 1994. It consists of four newly designed viewing probes and add tional lighting units and its upgraded enhancement. As a result, clearer pictures and a wider range of view, including the direct view of the divertor have been obtained. Consequently, the system has become an increasingly important instrument in machine operation, allowing early detection of any damage occurring during plasma operations and for calibrating several in-vessel diagnostics. IVIS campaigns, during which the entire vessel is scanned, are presently carried out on a weekly basis. Results are available in the form of picture mosaics (Fig.27), from which individual pictures can be easily chosen and enlarged for display on the monitor of any workstation connected to the internal network.

In 1994, an In-Vessel Training Facility was built, which is a realistic full-scale replica of Four Octants of the inside of the JET vessel and all of the important aspects of the in-vessel components (Fig.28). This facility has three main uses:

- a major concern for future shutdowns has been the provision of adequate in-vessel training. Due to beryllium contamination of the vessel and to remain within JET's self imposed radiation exposure limit of 10% of the legal limit, a large number of workers are required. The facility is used to ensure that in-vessel work can be done safely, efficiently and with a minimum of radiation exposure;

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 for detailed analysis. A summary of information from the JET pulses is held in the JET Survey Data Bank.

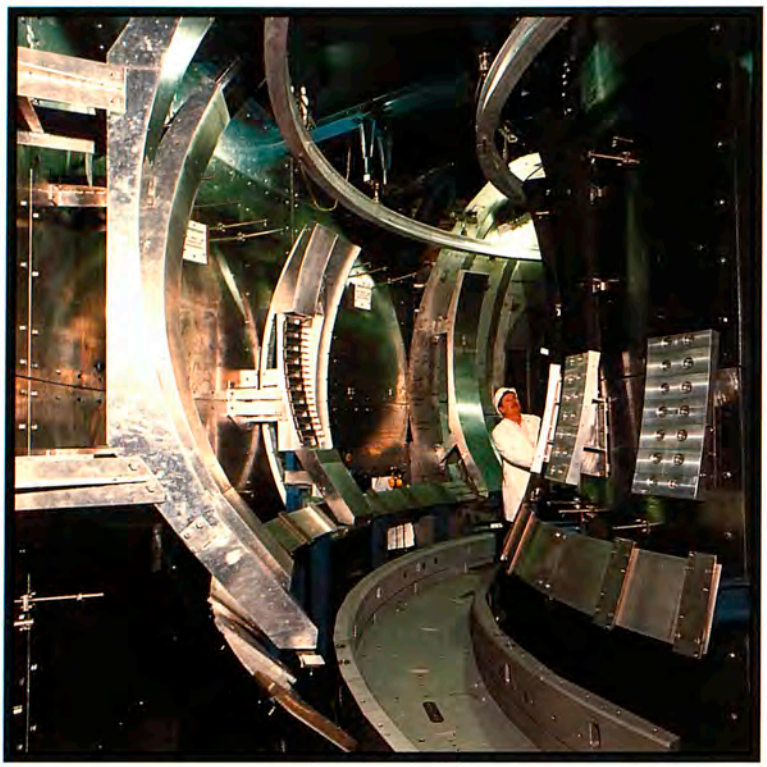


Fig.28: The In-vessel Training Facility

- the facility is used to check the details of new components, develop and test all new installation procedures, and interface problems to ensure that the Mark II divertor can be installed to the desired accuracy and with the prevailing boundary conditions;
- the facility will be used for remote handling operations trials. The preparation of equipment and operators for the fully remote exchange of the tile carriers will include the full-scale testing and proving of each individual task both under normal operating conditions and under failure case conditions. The forthcoming mock-up task trials will include extensive testing and proving of the capability to recover from worst case remote handling equipment failures and also from worst case failure of in-vessel components such as seizure of bolts or fractured tiles.

Waste Management

The Waste Management Group is responsible for provision of facilities in support of interventions and shutdowns, respiratory protection equipment and disposal of radioactive and beryllium wastes. This involves operation of controlled areas, including the Torus Access Cabin (TAC), the two Beryllium Handling facilities and Waste Handling.

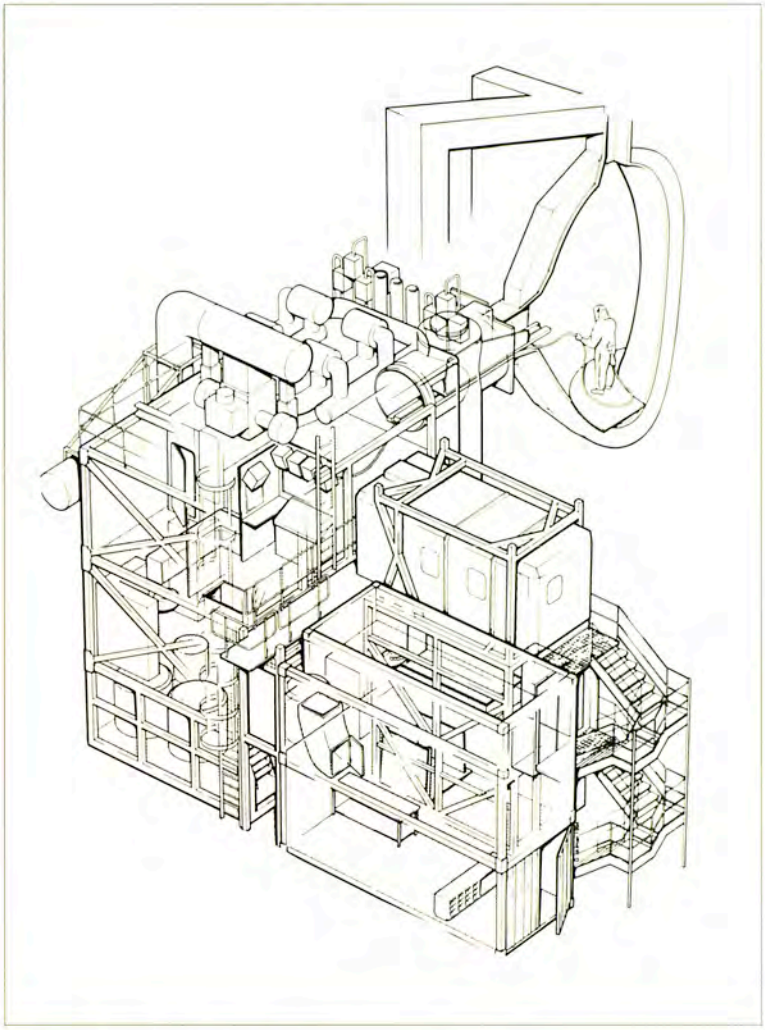


Fig.29: Torus Access Cabin

The Torus Access Cabin (TAC) enables personnel to access the JET vessel in a manner which prevents the spread of radioactive of beryllium contamination beyond controlled areas and includes decontamination, workshop and waste transfer facilities. The latter includes the ability to transfer both drummed waste and waste in ISO containers. These can be docked onto the TAC by means of a special double PVC membrane. It is designed in such a way as to present a clean, closed face on both controlled area openings to the Torus Hall environment. The containers, presenting a clean outside surface, can then be transferred through uncontrolled areas and docked onto the J30 Waste Handling Facility. The arrangements, which are shown in Fig.29, have reduced the transfer time from one shift to 45 minutes.

Although the TAC attached to the vacuum vessel was a semi-permanent arrangement operating continuously for 22 months of

the divertor shutdown, improvements have been made to permit rapid docking to the machine to permit access for emergency interventions. The TAC is normally parked in the Assembly Hall and is craned into the Torus Hall when required. Three interventions were made during 1994 and the in-line time involved in connection of the TAC (including opening of the pumping chamber door, establishment of services and preparation for entry in pressurised suits) has been reduced to less than one shift.

The Beryllium Handling Facilities have been operated continuously during 1994. These provide containment and contamination facilities to enable the carrying out of cleaning and maintenance on beryllium-contaminated components.

Control and Data Management

The JET Control and Data Acquisition System (CODAS) is based on a network of minicomputers. It is the only interface to operate JET and it provides centralised control, monitoring and data acquisition. The various components of JET are logically grouped into subsystems like 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 distributed front-end instrumentation. Signal conditioning and some data conversions are made using EUROCARD modules. The rest of the instrumentation is based on CAMAC and VME standards. Embedded front-end intelligence is implemented through microprocessors that are also used for real-time applications. The actions of the various computers are coordinated by supervisory software running in the Machine Console computer. This supervisory function includes the countdown sequence for the control and data acquisition during each plasma discharge.

During 1994, the main efforts were focused on completion of the move of CODAS to a UNIX environment, the support of the restart of operation in the new configuration and the implementation of a significant number of VME-based front-end applications. This required full integration of the new control systems into the operation and deployment of new front-end intelligence based on the VME standard and connected by a dedicated network.

The mainframe computing service is based on an IBM 3090/300J with 160 GigaBytes (GB) of disc storage and a further 2000 GB of

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.

automated cartridge tape storage. The service has operated since June 1987 and 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 384 MegaBytes (MB) of memory (128MB central and 256MB expanded), almost doubling the processing capacity.

In February 1992, a Memorex-Telex automated cartridge tape library (ATL) with a capacity of 1000GB was installed. The ATL not only provides storage for all raw JET data (JPFs) and archived processed JET data (PPFs), but also provides storage for backup and dump tapes, previously handled using manually operated cartridge tape drives. This together with the introduction of automated operations via the product AutoMate/MVS has completely eliminated the requirement for operator cover. In December 1993, the eight tape drives of the ATL were upgraded doubling the data recording density, and this doubled the capacity of the ATL. This disc system hardware was enhanced in two stages during 1994, with the addition of a further 53GB of a more recent generation of IBM discs.

The JET IBM Computer Centre was originally established at the UKAEA Harwell Laboratory but was relocated to Building J2 at JET in July 1992. This service has run successfully from the JET location, providing the expected improvement in communications, integration with the UNIX and PC systems, and a reduction in staff. The integration was further improved early in 1994 with the introduction of TCP/IP services on the mainframe using the Interlink SNS/TCP access software to provide more accessible means of data transfer between the mainframe and the UNIX and PC or Macintosh systems. It also improved the terminal emulation access from the PCs and UNIX system.

Summary of Machine Operations

During 1994, operations were made up of two distinct periods.

(a) First Period (Weeks W4 to W17)

The Divertor Shutdown ended in January and the machine integrated commissioning phase started. Commissioning activities were undertaken by a "Restart" Task Force (Task Force R), whose main objectives were the progressive and complete integration of systems for:

- producing the first plasma;
- establishing a 2MA divertor plasma with up to 10 MW of heating.

The first plasma pulse was followed by a period of initial operation for plasma commissioning with the Plasma Position and

Current Control system (PPCC) using the magnetic signals from the magnetic diagnostic (KC1). Development and commissioning of a new improved magnetic diagnostic (KC1D) was carried out in parallel and progressively integrated. The Task Force R work continued until the end of April 1994 and obtained reliable 3MA plasma discharges with additional heating. 65% of pulses were successful. Of these, 60% were commissioning pulses and 40% were plasma pulses.

(b) Second Period (Weeks W18 to W51)

Operation time within the experimental programme was shared between the three main Task Forces: Divertor Assessment (D); High Performance (H); and Tokamak Concept Improvements (T). Some time was also devoted to plasma commissioning (C) and ICRF commissioning (within Task Force RF).

The experimental programme was carried out in double-shift operation, six days a week. The number of shifts were distributed as follows:

TASK FORCE	NO: OF SHIFTS	% OF TOTAL
TASK FORCE D	104	32.30%
TASK FORCE H	84	26.10%
TASK FORCE T	76	23.60%
TASK FORCE C	48	14.90%
TASK FORCE RF	10	3.10%

The overall ratio of successful pulses either for commissioning or plasma increased to 77% (compared with 65% during the period devoted to Task Force Restart).

Interruptions to Operation

The time available for operation was interrupted by three in-vessel interventions:

- the first interruption occurred during Weeks W12 and W13. Some large nickel influxes, sparks and sometimes a long afterglow were visible in most of the divertor or limiter plasmas. Detailed in-vessel inspection and completion of some diagnostic systems were possible over this non-operational period of seven days;
- the next intervention, during Weeks W38 and W39, was spread over nine operational days. It was then possible to carry out repair work

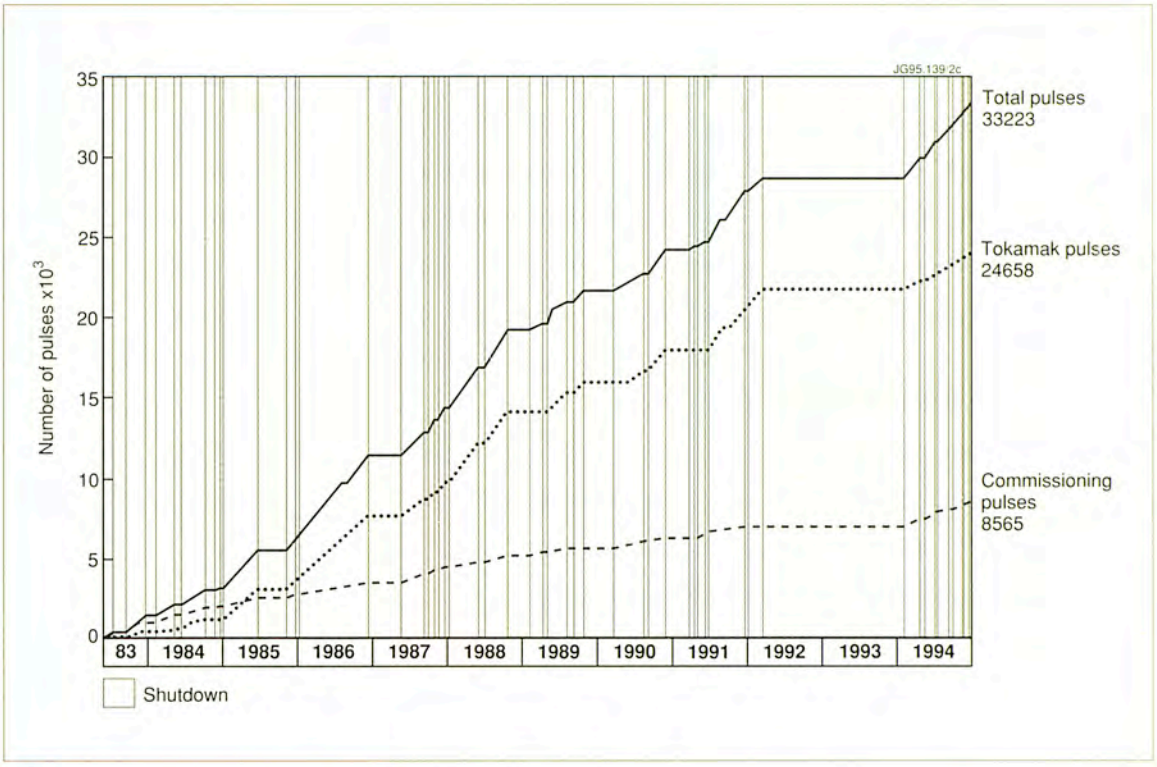


Fig.30: Cumulative totals of JET Pulses: 1983-1994

on the upper saddle coils, following some severe distortions on the cross-over busbar sections. At the same time several other activities were successfully completed;

- the third interruption occurred during Weeks W44 and W45, when nine days were used for replacement inside the vessel of the three beryllium evaporator heads located at Octants Nos.1, 5 and 7. These units were fitted with new extension pieces whilst the evaporator at Octant No.3 was removed.

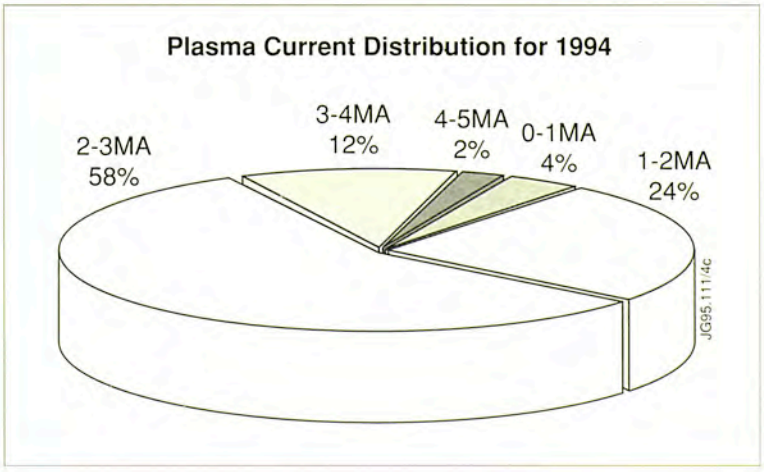


Fig.31: Plasma Current Distribution for 1994

Overall Summary

The total number of pulses carried out in 1994 was 4869 with an overall distribution over 1983-1994 as shown in Fig.30. An analysis of the distribution of the plasma currents showed that currents of 2-3MA were routinely used during 1994. These represented 82% of the total plasma pulses (Fig.31). A total of 43 pulses in the range 4-5MA were also obtained.

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 shutdown periods. Reference should be made to the section on the Future Programme of JET to relate these technical developments to the overall JET Programme.

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. Work is underway on two systems: a high-speed pellet launcher and a pellet centrifuge for shallow deposition.

The high-speed launcher is designed to accelerate sabot supported 6mm deuterium pellets to speeds up to $4kms^{-1}$ by using two-stage light gas gun driver technology for deep central plasma fuelling purposes. Sabots are small plastic pistons, set behind the deuterium pellets to protect them from the hot driving gas. These sabots, needed for pellet velocities in excess of about $3kms^{-1}$ and being accelerated with the pellet, are made from two halves of equal mass, which split after leaving the barrel due to aerodynamic forces and can be eliminated from the pellet path by a shear cone.

This system was installed on the machine and commissioned with plasma towards the end of 1994. First pellet launches into the plasma have been performed. The two-stage gun performed with few problems and no further modifications were required. Some limited experiments have been performed with pellet injection using

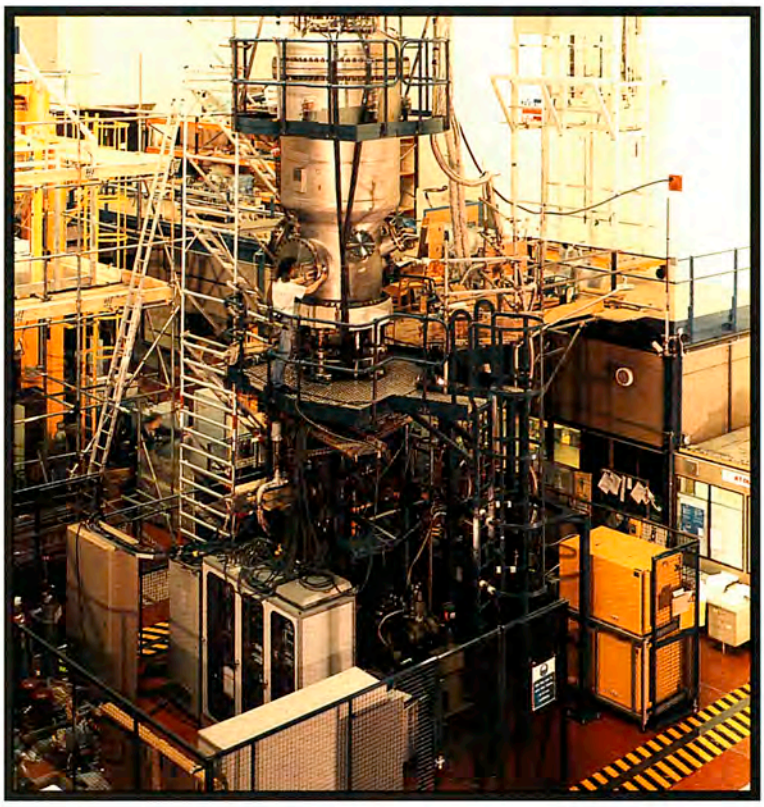


Fig.32: The Pellet Centrifuge

polypropylene (impurity injection experiment) and deuterium. The high-speed pellet activity was terminated at the end of 1994 and the equipment was mothballed.

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. The injection parameters chosen are pellet sizes 1.5-3mm with repetition frequencies up to $40s^{-1}$ at speeds of $50\text{-}60ms^{-1}$ for long pulses approaching one minute. Fig.32 shows the pellet centrifuge unit on its stand in the Assembly Hall. It accelerates pellets mechanically from the hub to the tip of a rotor arm, where these leave at about 1.4 times the rotor tip speed. Each size of deuterium ice pellet will be launched from one of up to four possible individual extruder units into the central part of the centrifuge rotor hub, into which reaches the stationary stop cylinder featuring a hole to ensure the proper starting conditions for the pellet on the rotor arm. The design of centrifuge rotor and stop cylinder follows very closely that of the centrifuge developed for ASDEX Upgrade by IPP, Garching, Germany, who also advise JET under contract. The extruder is of a new design by JET to provide a much larger number of pellets per tokamak pulse.

The first extruder for 3mm pellets has now been assembled, and it has undergone warm pre-testing of its piston action. The extruder will be installed in the main vessel unit in early 1995. It will then be commissioned and made operational from initial cool-down to ice extrusion and pellet formation phases, with the aim of preliminary application of pellet injection into the torus.

Design of Mark II Divertor

The Mark I divertor currently installed in JET has been successful from several points of view. In particular, its power exhaust capability in the sweeping mode has been very good, and the cryopump works effectively. The fact that Mark I is relatively wide has made it possible to investigate a large variety of magnetic equilibria with strike points on both the horizontal and vertical sections of the target tiles.

However, the openness of the divertor also has drawbacks. Such an open divertor is not as effective at restraining neutrals in the divertor region as a more closed design, particularly at high powers and low to moderate main chamber densities. Under these conditions, the divertor plasma itself is not opaque enough to re-ionize the recycling neutrals. Thus, if was decided to design and construct a series of more closed divertors, beginning with Mark IIA, to be installed in 1995, and followed by Mark IIGB (Gas-Box) in 1997. The purpose of these closed divertors is to permit a high neutral density in the divertor volume while maintaining a low neutral density in the main chamber. The first condition is required to exhaust power to a large fraction of the total divertor wall area by the volumetric processes of radiation and charge exchange (i.e. to operate in the highly radiating, partially or fully detached regime). The second requirement, that of low neutral density in the main chamber, is necessary for highest main plasma performance. Mark IIA is a moderately closed semi-V divertor which flares out at the upper region of the side plates in order to accommodate a wide variety of equilibria. These include several low X-point equilibria of varying degrees of flux expansion, which have strike zones on the "outward facing" dome targets, as well as higher X-point equilibria which contact the "inward facing" vertical targets. The target tiles are 40 cm long in the poloidal direction, providing a high toroidal wetted length, relative to Mark I. This feature, along with the inclination of the targets, gives a wetted area sufficiently large that the need for sweeping is

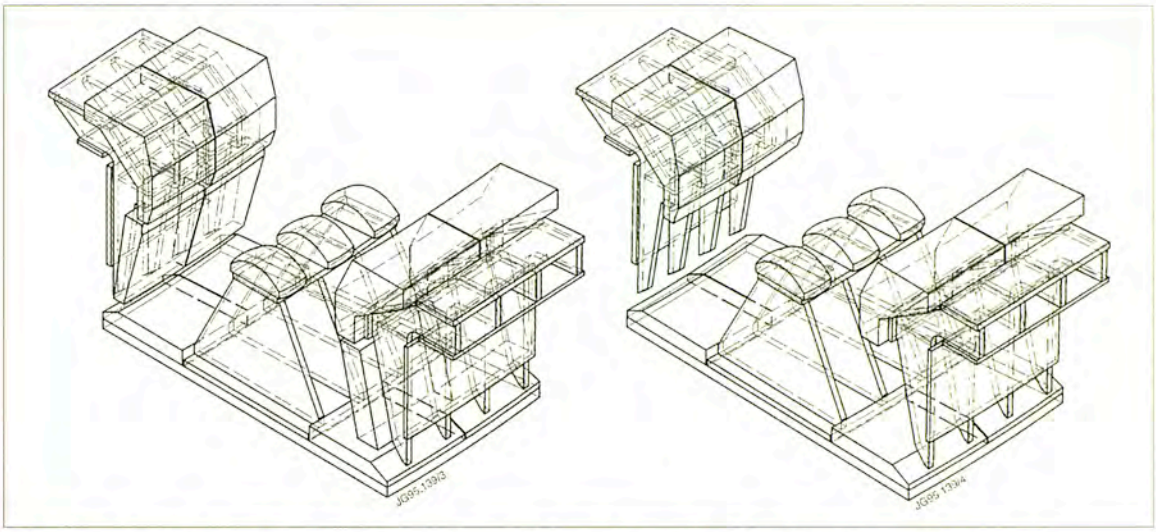


Fig.33: Mark II Gas-Box divertor: (a) with lower side-wall tiles, and (b) without lower side-wall tiles

eliminated. Simulations indicate that the pumping performance should be somewhat better than that of Mark I. Mark IIA is intended to satisfy the dual and sometimes conflicting needs of performing well in both the high performance campaigns (low to moderate density, magnetic axis not too far above the midplane) and ITER-related high density radiating divertor studies. The design of Mark IIA was finalised in 1992 and procurement and fabrication went ahead throughout 1993/94. It is scheduled to be installed starting in June 1995.

The term "Gas-Box" is used to denote a divertor characterized by a close fitting baffle near the X-point, below which there is a relatively large volume in which the recycling neutrals (both hydrogenic and any seeded recycling impurity which may be used) can circulate freely in order to-re-enter the divertor plasma along its entire surface. This promotes the volume loss of energy and momentum required to prevent excessive energy deposition on the target "strike zones". The JET Gas-Box divertor uses the same coils, pump, and support structure as the Mark IIA divertor, but has new target tiles and tile carriers to change the geometry from moderate-V to Gas-Box configuration. The system has been designed so that the changeover can be carried out via remote handling, following the D-T campaign scheduled for late 1996.

To provide a specific test of the Gas-Box concept, the range of equilibria must be restricted to allow for a baffle which is long enough, and close fitting enough, to prevent the back-flow of neutrals into the main chamber. In practice, this means that only high X-point equilibria can be investigated.

Two versions of the Gas-Box are being designed and are shown in Figs.33 (a) and (b). These use a common baffle, base target, and septum assembly. The "vertical target" version shown in Fig.33(a) has a nearly vertical tile placed just below the baffle on both the inside and outside legs, and operates with the strike points on these vertical tiles. In this case, the recycling of neutrals takes place through the private flux region only. Such vertical target divertors have certain advantages, and current ITER thinking favours such a design. Among the advantages are a higher wetted surface for handling power during non-detached phases of the discharge, such as start-up or in ELMs. In the second Gas-Box configuration (Fig.33(b)), the vertical target tiles are removed and the strike points are placed on the bottom plates. The wetted surface is now smaller, but the pumping is somewhat stronger.

In contrast to the design of Mark IIA, the tile carriers for Mark IIGB will probably be fabricated from CFC plates. The anticipated radiative loads on the carriers, including the septum, are sufficiently high that bare inconel could not be used, and would need to be clad. Thus, the adoption of CFC as the structural member should lead to a simpler engineering solution, and tests are in progress to validate this approach. The target plate and the baffle material is CFC. In principle, beryllium could be used for a later version, if justified by continuing interest from ITER and favourable results from the Mark I beryllium tile test in 1995.

At present, it is planned to test both versions of the Gas-Box divertor. However, the scheduling is very tight, and it may be necessary to omit one of them. The vertical target version is the one currently favoured by JET (as well as by ITER), but the design is sufficiently flexible to permit either version to be operated first.

Tritium Handling

The purpose of the Active Gas Handling System is to pump the torus, to collect gases from various systems (torus, neutral beam injection, pellet injection and various diagnostics), to purify and isotopically separate these gas mixtures (consisting of the six hydrogen molecules, helium and impurities such as hydrocarbons, oxygen, nitrogen, etc) and to re-inject pure tritium and deuterium gas into the torus.

The system is situated in a separate building and can be separated into sub-systems as shown in the block diagram (Fig.34). During the year, the inactive commissioning phase of the sub-systems continued in

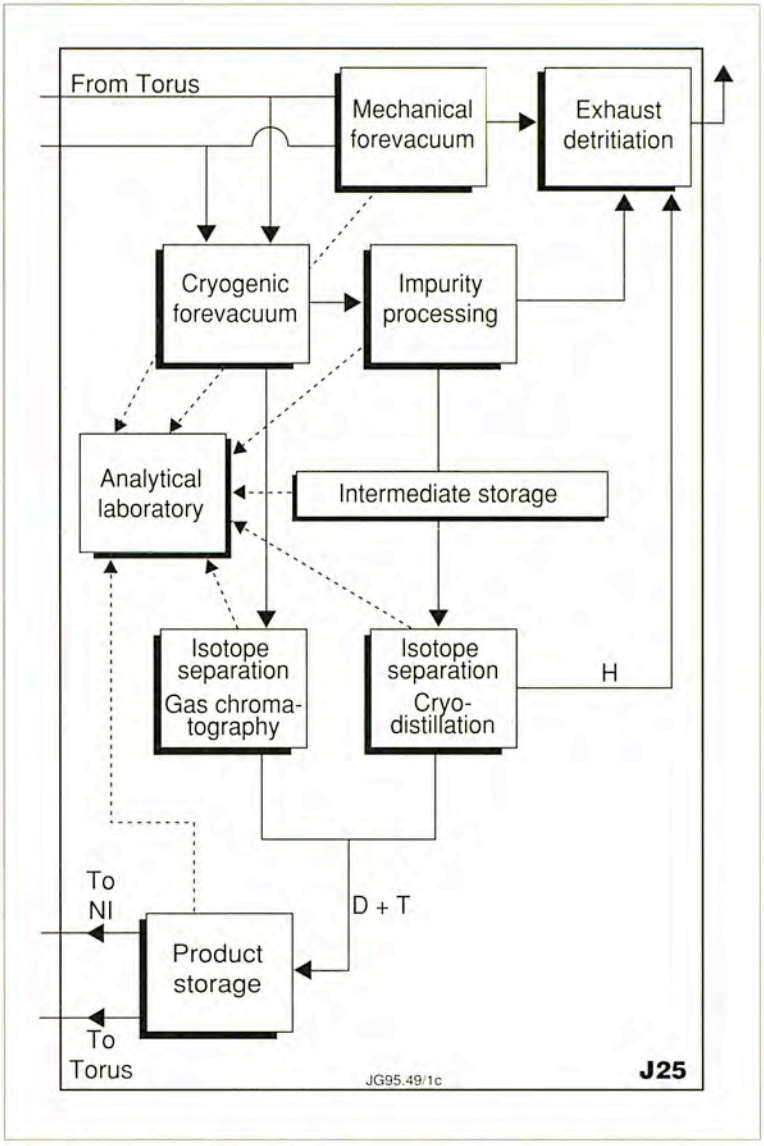


Fig.34: Block diagram of Active Gas Handling System (AGHS)

accordance with the JET programme for D-T operations in 1996. Some sub-systems are almost ready for active commissioning with tritium trace amounts, which is planned for mid-1995.

With the commissioning of the individual sub-systems being substantially complete, the emphasis has been on operation of the system in an integrated way under conditions relevant to the actual tritium processing cycle. For example, the gas chromatography isotope separation system has been operated in conjunction with the intermediate and product storage using uranium beds in these sub-systems for hydrogen isotope pumping.

The completion of Safety Related Commissioning Procedures is a pre-requisite for obtaining approval for the start of commissioning with

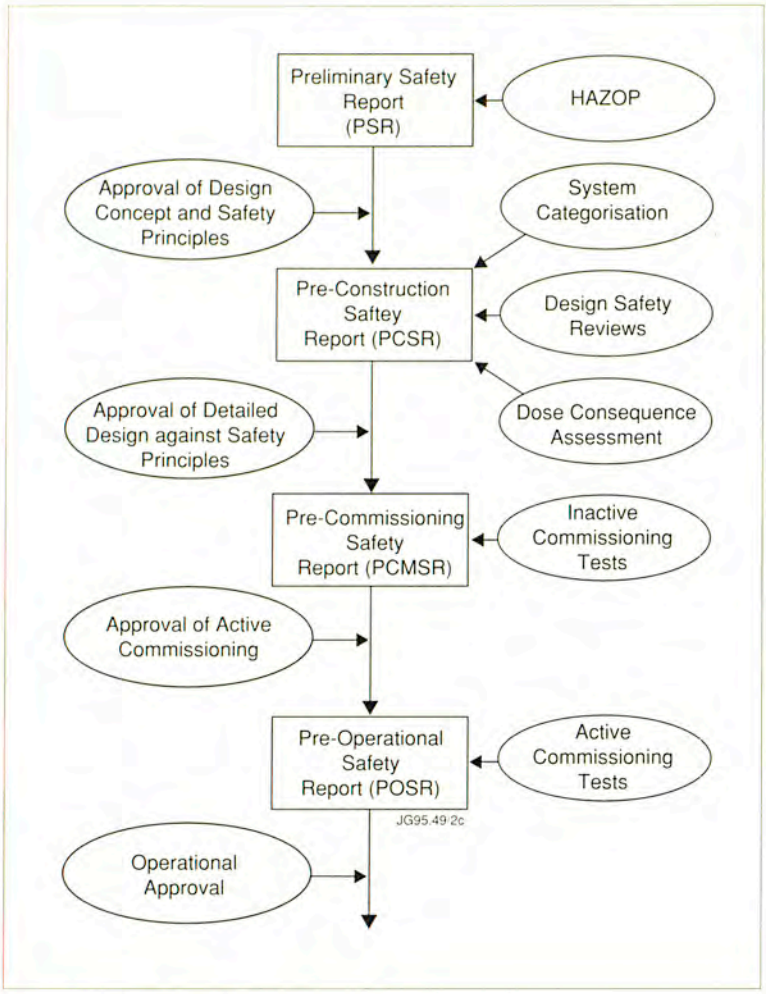


Fig.35: JET Safety Submissions

tritium. These cover specific conditions such as leak rates, ability to transfer gas to special tanks for processing, and verification of the correct operation of all valves from the control system. A block diagram of the safety submissions required are set out in Fig.35.

The formal safety documents which comprise the safety case for the AGHS have been endorsed by the Fusion Safety Committee and the UKAEA's Safety and Reliability Directorate (SRD). In particular, the Pre-Commissioning Safety Report (PCMSR) was endorsed during 1994 and a number of procedural aspects requiring resolution were identified. Once these are cleared, approval for tritium commissioning of the AGHS is expected from UKAEA, which should enable it to proceed in mid-1995.

The UKAEA's Safety and Reliability Directorate (SRD) endorsed the Preliminary Safety Report (PSR) for the torus which provides part of the required documentation to use tritium. Work then began on the

next phase of documentation, the Pre-Construction Safety Report (PCSR). This report will provide a comprehensive description of the torus and its auxiliary systems, as well as safety analyses of postulated accidents and probabilistic risk assessments.

An analysis was undertaken to assess the torus and its auxiliary systems for their intrinsic radiological hazard. This work was completed and conclusions accepted by the UKAEA and the Fusion Safety Committee. Much of the supporting work for the PCSR analyses was completed in 1994 and a few remaining jobs initiated. This work is expected to be completed in early 1995.

Technical Preparations for D-T Experiments

A DTE1 Technical Preparations Group was set up to identify the technical work which needed to be completed before the DTE1 experiments could take place in 1996. In particular, it was important to identify all the work which needed to be included in the 1995 shutdown, bearing in mind that this would be the last major shutdown before DTE1. The group also took note of additional work which would be required for DTE2 in 1999, but the main emphasis was on DTE1. For this purpose, DTE1 was taken to be a programme producing $\approx 2 \times 10^{20}$ D-T neutrons in a three-month period at the end of 1996, with provision for a remote handling divertor configuration change immediately after the experiment.

The technical work required to prepare for DTE1 was identified and in-vessel work to be accomplished in 1995 was included in the in-vessel planning. The ex-vessel work was included in the ex-vessel planning, and a large amount of ex-vessel work could be accomplished within the planned shutdown duration.

It was shown that there were no technical or approval obstacles which could not be overcome in time for DTE1 to take place at the end of 1996 and experience with the PTE meant that there was confidence that the programme could be executed in a safe and professional manner. However, it was clear than improvement in the reliability of JET operation was required before DTE1 could take place efficiently.

When the present operating campaign comes to an end, the process of specifying the content of the DTE1 programme in more detail should be started. This would be difficult in 1995 when there

was no experience of Mark II divertor operation. Nevertheless, the preparation of specific experimental proposals should begin. The proposals should address the main areas already identified:

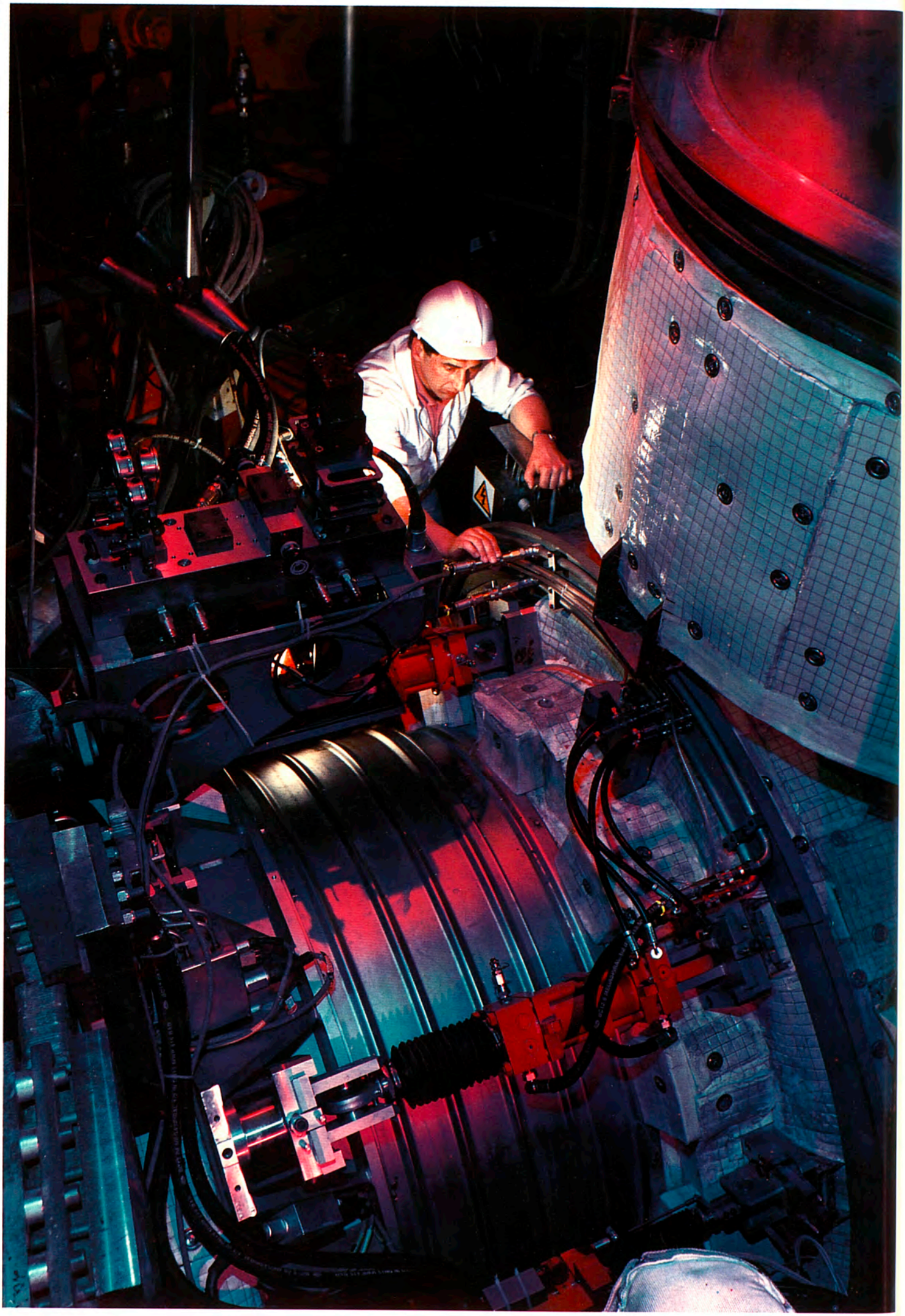
(a) refine the objectives of each experiment;

(b) estimate the experimental time required;

(c) estimate the required neutron production;

(d) define the D-D comparison discharges; and

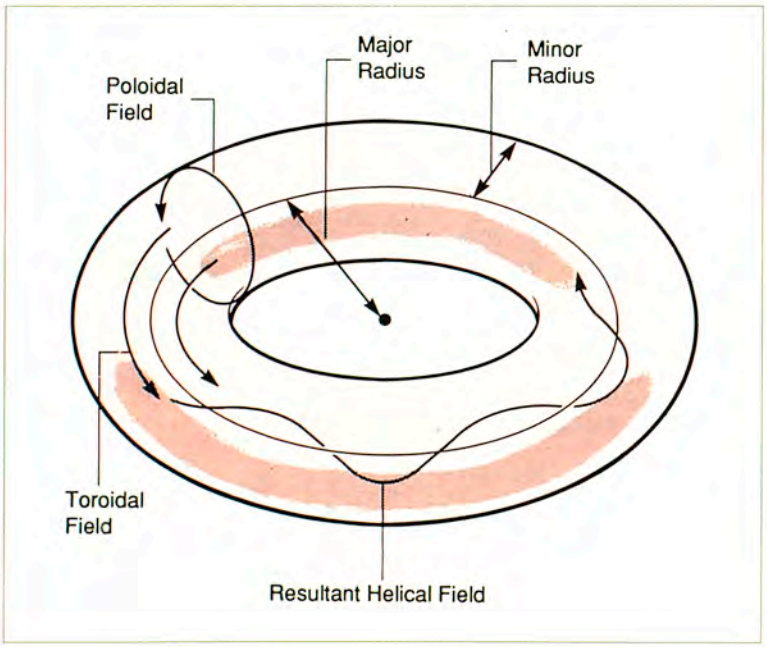
(e) specify the plasma performance to be demonstrated in D-D before the experiment could proceed.



Scientific Advances During 1994

Introduction

The overall objective of the Project is to study plasma in conditions and with dimensions close to those needed in a fusion reactor. The central values of temperature, density and energy confinement time required for a reactor operating with deuterium and tritium are such that the fusion triple product, $(n_i \tau_E T_i)$, must exceed the value of $5 \times 10^{21} m^{-3} keV$. Typical values for these parameters which must be attained simultaneously in a reactor, are given in Table 6. With chmic heating alone in JET, temperatures of 3keV and 4keV for the ions and electrons, respectively, densities of $4 \times 10^{19} m^{-3}$ and energy confinement times of 1s are the limits that have been achieved. These parameters were obtained simultaneously during one discharge and resulted in a fusion product of $1.2 \times 10^{20} m^{-3} keVs$.



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.

TABLE 6: REACTOR PARAMETERS

CENTRAL ION DENSITY, n_i	$2.5 \times 10^{20} m^{-3}$
GLOBAL ENERGY CONFINEMENT TIME, τ_E	1-2s
CENTRAL ION TEMPERATURE, T_i	10-20keV

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.

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 methods. Even so, these substantial increases in temperature were associated with a reduction in energy confinement time as the heating power was increased. Thus, gains in plasma temperature have been partly offset by degradation in energy confinement time. The fusion product values 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 so-called magnetic limiter (X-point) configuration. During the 1991/92 campaign, values of $9\text{-}10 \times 10^{20} m^{-3}$keV were obtained using up to 16MW of additional heating.

Ignition

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

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 temperature up to 30keV, energy confinement times up to 1.8s and central densities up to $4 \times 10^{20} m^{-3}$.

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 acceptable 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. The Ohmic Heating, Phase 1, 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 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 was completed in early 1992. In 1991, JET's lifetime was prolonged by four years until the end of 1996. The extension was approved to allow JET to implement the new Pumped Divertor Phase of operation, the objective of which is to establish 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.

The 1991/92 experimental period concentrated on: optimisation of plasma performance; advancing understanding in certain key areas of tokamak physics, (such as: physics of the H-mode; energy transport and confinement; and transport of particles and impurities); and establishing the basis for pumped divertor and Next Step physics (including a Preliminary Tritium Experiment).

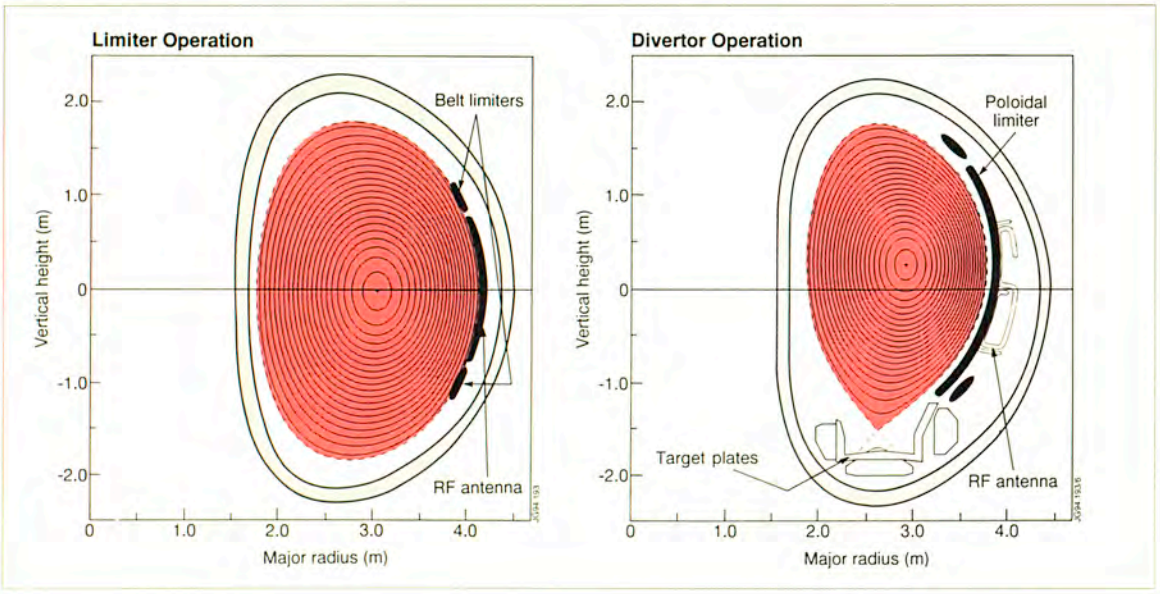
Operating Modes

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.



The machine entered a major shutdown in early 1992, which lasted throughout 1992 and 1993, to prepare the machine for the divertor phase of operation. At the start of 1994, JET was nearing the end of the longest and most extensive modifications since initial assembly of the device. During the long shutdown, the interior of the vacuum vessel was essentially replaced. Following completion of this work, JET was in a position to begin its planned programme of operations to demonstrate effective methods of power exhaust and impurity control in conditions close to those envisaged for ITER.

Main Scientific Results

The main objectives of the 1994 campaign period were: to assess divertor performance on the vertical and horizontal targets of the Mark I divertor; to demonstrate the high performance capability of the JET pumped divertor; and to study those physics areas in which JET could make important contributions to ITER and DEMO. The main scientific results obtained during the 1994 campaign are set out below.

Power Handling and Particle Control with the Mark I Pumped Divertor

Power Handling

The study of divertor performance under steady-state conditions at high power has been a central goal of the programme. Particular care was taken in the design and installation of the power handling surfaces of the Mark I divertor target to optimise power handling, in particular, by avoiding exposure of tile edges. The strike points could be swept over the target plates at 4Hz with an amplitude of ~10cm to increase the area of the target effectively "wetted" by the plasma. The Mark I divertor showed excellent power handling capability, with no "carbon blooms" limiting the performance, as seen in the past. The performance of the target was investigated over a wide range of plasma conditions, at powers up to 26MW and in steady-state H-modes lasting up to 20s. In no case, including hot-ion H-modes where performance was previously limited by impurity influxes, has a bloom of carbon impurity been observed. The combined effects of sweeping and giant ELMs, which are now a feature of H-modes, are such that the surface temperature of the target has not exceeded 1200°C, even in cases where over 100MJ of energy has been delivered to the horizontal target.

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.

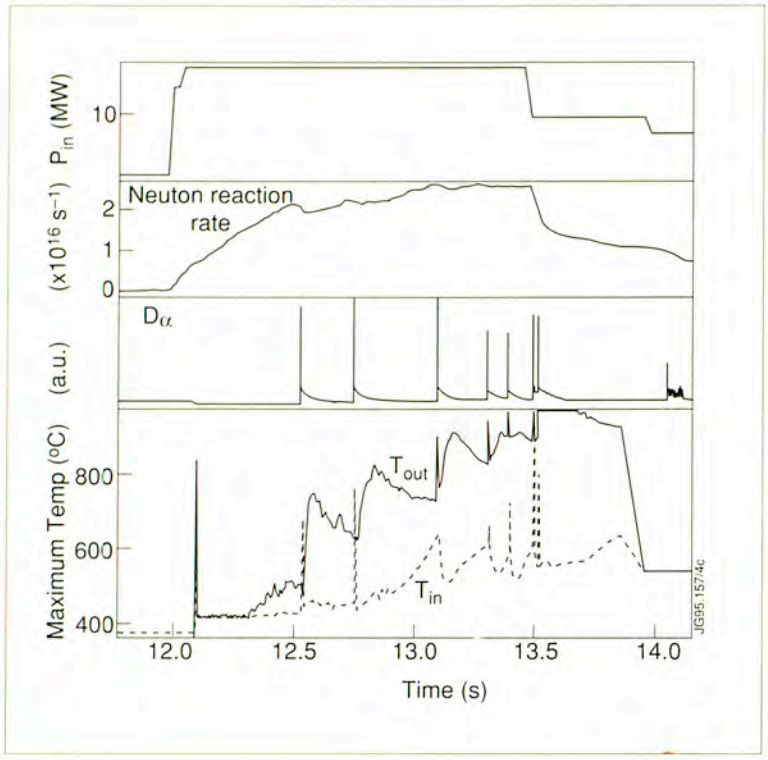


Fig.36: Time evolution of tile temperature during swept and unswept cases

From the power handling point of view, the appearance of edge instabilities (ELMs) had two effects. Firstly, they deposit power on surfaces other than the strike zones, and secondly, since they create a spike in the radiated power, the power conducted to the strike zones decreased immediately after the ELM event. In this sense, ELMs are extremely effective in alleviating the heat load on the strike zones, but the penalty is the uncontrolled deposition of power in zones which may have a poor power handling capability. In Fig.36, the time evolution of the target temperature in the two strike regions is shown for a discharge in the hot-ion H-mode regime (I_p=5MA, B_T=3.4T, P_{in}=20MW). Between giant ELMs, the power imbalance changed between inner and outer strike zones.

Sweeping of the strike points over the target plates spreads the power load and improved the power handling capability of the tiles. Fig.37 shows the power injected during two pulses, one with and one without sweeping: the current in the divertor coil D2 is an indicator of the sweep, and the time evolution of the maximum temperature of the tiles is shown for the two cases. In the swept case, the peak temperature decreased when the sweep began and stabilised at ~550°C, in contrast to the unswept case where the peak temperature

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.

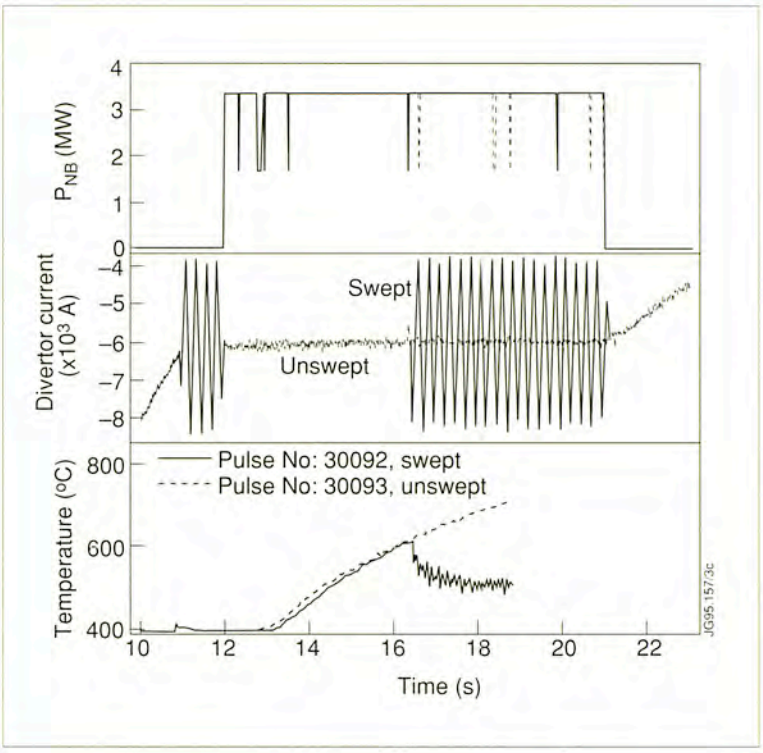


Fig.37: Time evolution of the surface temperature in the two strike regions for a discharge in the hot-ion mode

continued to increase steadily. With sweeping, the effective power seen by the target was ~66% of the power in the unswept case.

An example of the power handling capabi ities is shown in Fig.38, in which the discharge is subjected to 25MW of combined neutral beam and ICRF heating. 72MJ of energy was conducted to the tiles and their surface temperature increased ~650°C. In other discharges, almost 140MJ of neutral beam energy was injected during ELMy H-mode plasmas, raising the surface temperature of the tiles to ~550°C. In the X-point configuration of 1991/92, an input energy of about 15MJ was sufficient to raise the tile temperature above 1500°C and provoke a severe impurity influx (carbon "bloom") which quenched the discharge.

The vertical side plates, intended primarily as protection for the divertor coils, have also been used as the targets for ohmic and ELM-free H-modes with neutral beam heating. In line with model predictions, the density scrape-off layer is narrower and the temperature profile is inverted with operation on the vertical target. This result gives confidence that the basic modelling assumptions are correct and with further validation and refinement will provide the tools to help design a reliable ITER divertor.

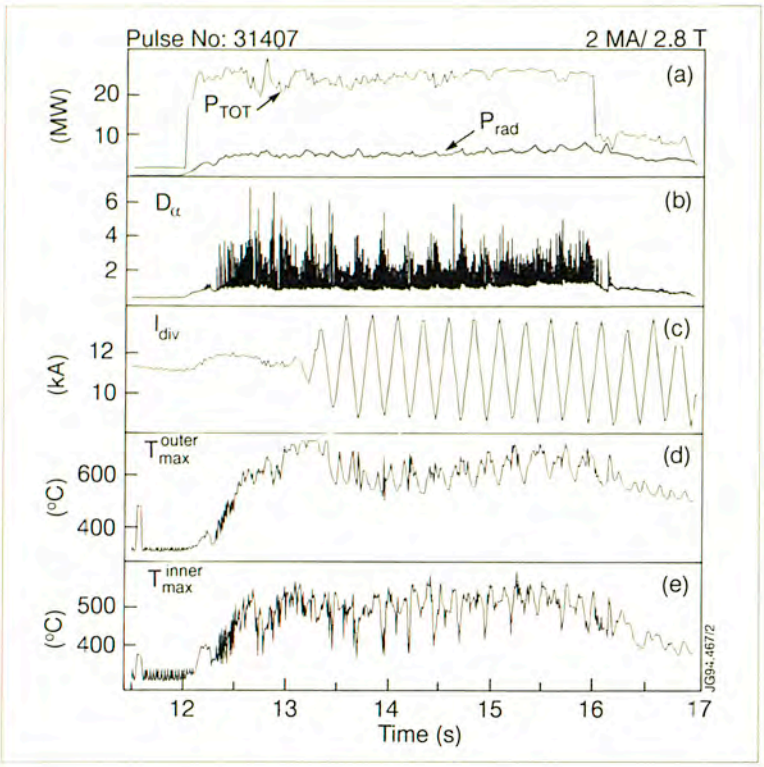


Fig.38: Time evolution of a 2MA ELMy H-mode with 26MW of combined heating, illustrating the low tile surface temperatures achieved with the Mark I divertor ((d) and (e)) when strike point sweeping ((c)) is applied

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.

Particle Control

Active particle control, including efficient helium exhaust, is required for steady-state operation in a reactor. The study of the effects of fuelling and active pumping on plasma properties has been one of the main topics of the divertor campaign. In particular, an upgraded gas introduction system, characterised by toroidally and poloidally distributed gas sources in the vacuum vessel, and an entirely new internal cryopump capable of providing active pumping during plasma pulses have been extensively used.

The torus cryopump has proven to be effective for plasma purity and density control, removing neutral deuterium for a wide range of strike point positions on both the horizontal and vertical target plates. The particle removal rates for the different strike point positions are shown in Fig.39 for typical ohmic plasmas. The variation in the removal rate is useful in facilitating density control in steady-state ELMy H-mode plasmas as shown in Fig.40 for three different magnetic configurations. The particle removal rate is several times the maximum neutral beam fuelling and this helps deplete gas reservoirs in plasma facing surfaces and reduce recycling.

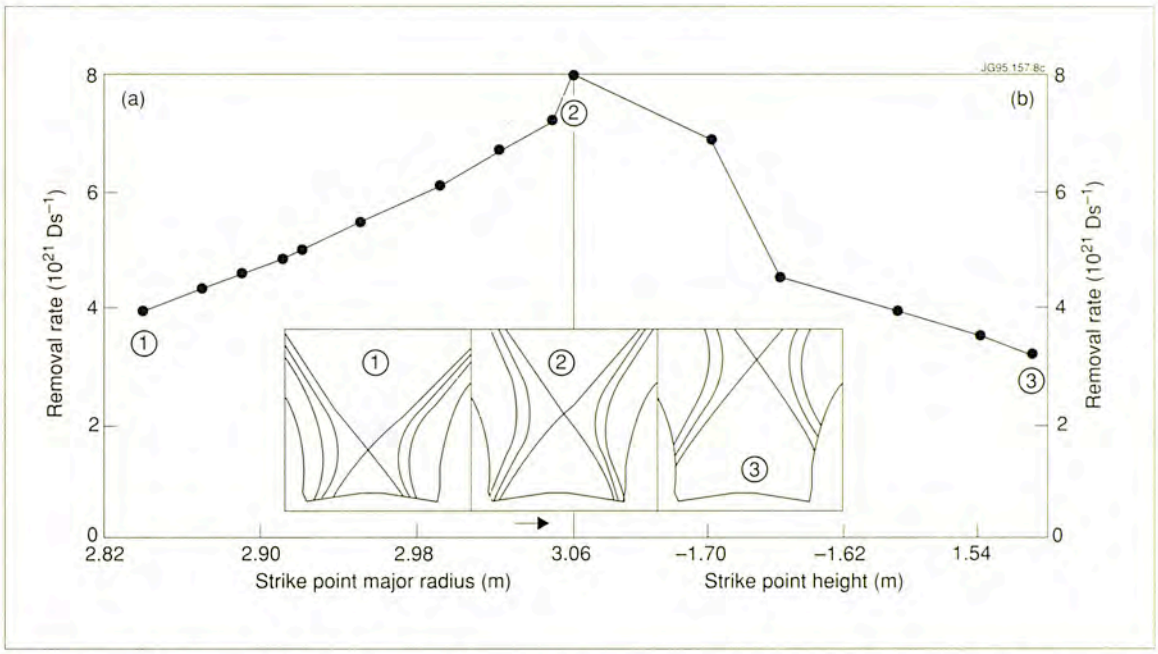


Fig.39: Particle removal rate as a function of separatrix position on the horizontal plates (a) and on the vertical plates (b). The full sweep of separatrix position is carried out in ~10s

The efficiency of the cryopump in reducing recycling has been exploited in high performance hot-ion H-modes to the point where the neutral beam particle source dominates fuelling from recycling. Hence, peaked density profiles, favoured for improved neutral beam penetration in this regime, are maintained and lead to improved fusion performance.

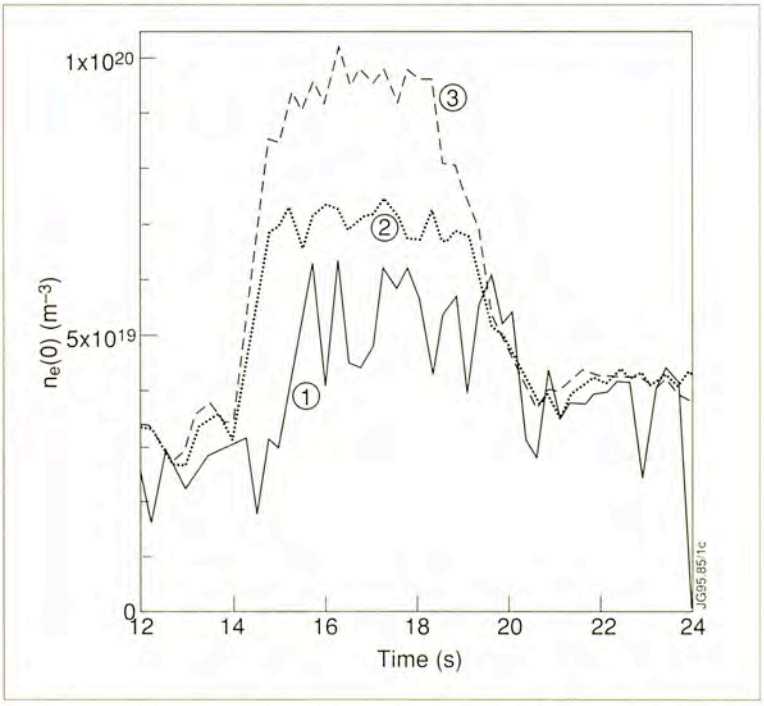


Fig.40: Density control in ELMy H-modes with 15MW NB heating showing variation of strike point relative to divertor cryopump: (1) Strike point near pump, no added gas; (2) Strike point near centre of horizontal target, no added gas; (3) As (2) but with added gas flow of ~2x10^{22}D$^{\circ}$s^{-1}

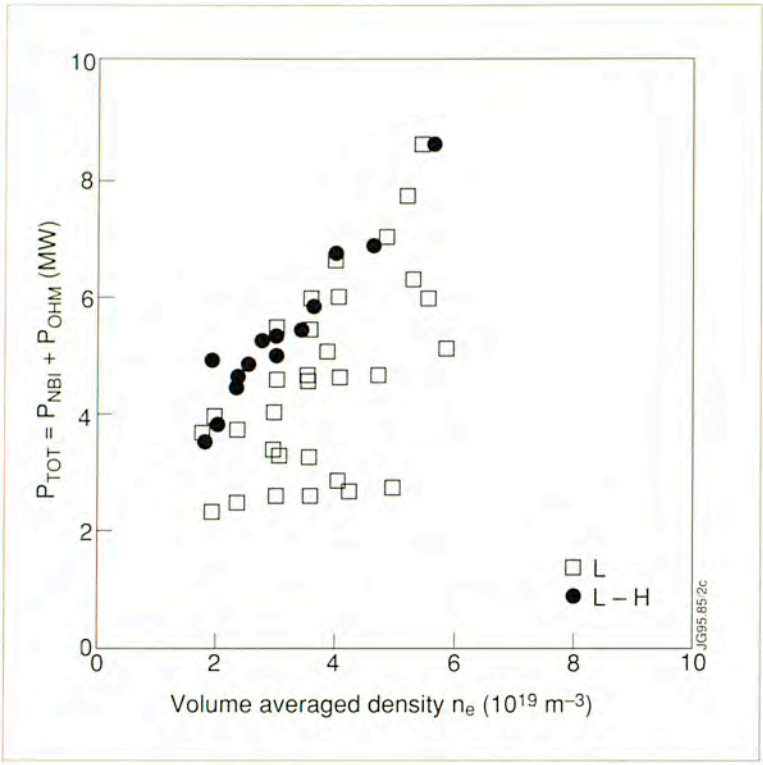


Fig.41: Power threshold for H-modes at 2MA as a function of density

H-Mode Threshold and Confinement

Transition from low confinement (L-mode) to high confinement (H-mode) regimes have been achieved with similar input powers as those required during previous campaigns. During 1994, a more thorough study was carried out and demonstrated a clear scaling of H-mode threshold power with plasma density and with toroidal magnetic field. Figure 41 shows the power threshold as a function of density carried out at constant toroidal field. When combined with results from other tokamaks operated with the same geometry (for example, ASDEX-Upgrade and COMPASS in Europe), a more accurate size scaling of threshold power will be determined and will be used in predictions for ITER.

In previous operation, edge instabilities (ELMs), were rarely encountered in the H-mode and were only observed either just above the L-H transition threshold; with extremely strong gas fuelling; or at high beta after a period free of ELM instabilities. The plasma in the new configuration now develops regular ELMing as standard behaviour. Long ELM-free periods (greater than 1s) can only be obtained with special preparation. These edge instabilities have the characteristics of 'giant' ELMs identified on other machines and have an ELM frequency which increases with input power and reduces with current.

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)).

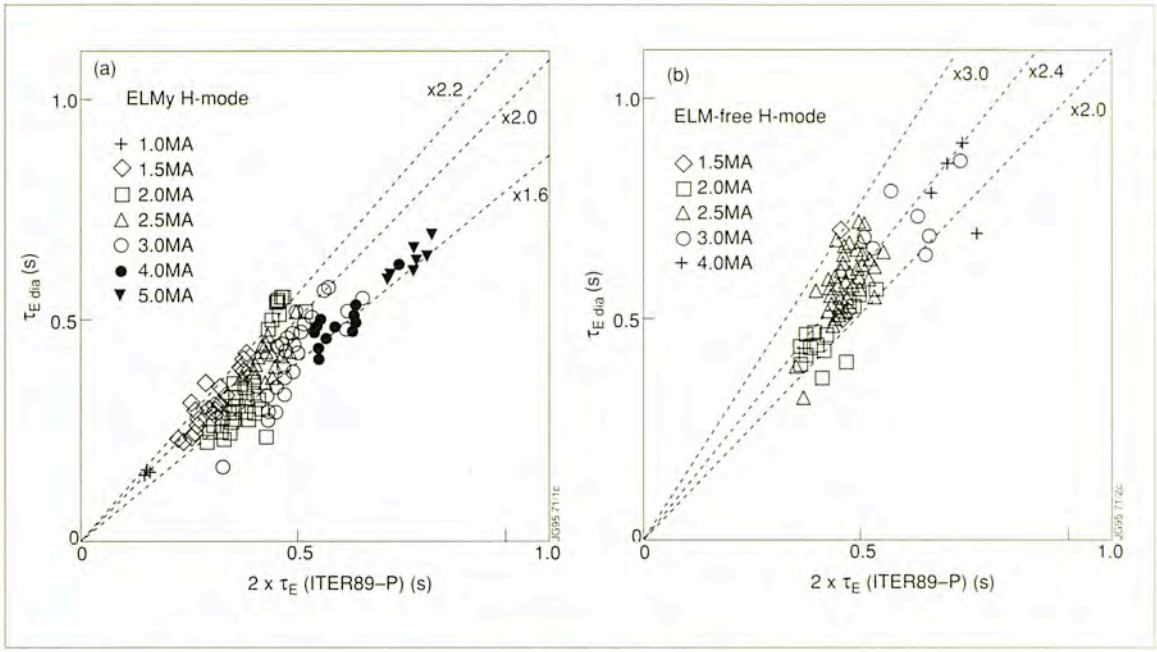


Fig.42: (a) Measured confinement time (τ_{EDIA}) for 1994 JET ELMy H-mode dataset versus twice the prediction from ITER89-P scaling law ($\tau_{EITER89P}$) (Data are quasi-steady-state); (b) for 1994 ELM-free H-modes with $\tau_{ELMFREE}$>0.5s

As a consequence, two approaches to high performance can be pursued: long pulse, ELMy H-modes, which maintain steady plasma conditions for many energy confinement times (considered the most credible mode of operation for ITER); and ELM-free, hot-ion H-modes in which the highest performance is achieved transiently.

ELMy H-modes show a typical enhancement factorin confinement time, H≈1.8, but can rise to H≈2.5 in certain circumstances, as shown in Fig.42(a). However, confinement in the ELM-free phase is typically a factor, H≈2.2-2.4 higher than low confinement (L-mode) scalings (see Fig.42(b)). This value is not significantly different from past campaigns, although the absolute level of confinement is, as expected, about 15% lower, since the plasma volume with the divertor is now smaller.

Long Pulse Steady-state H-modes

The excellent power handling capabilities of the divertor target, together with the in-vessel cryopump and the facility for sweeping the strike zone position, have been exploited to produce long pulse steady-state H-modes. The H-modes produced are characterised by frequent regular giant ELM behaviour, a regime which is now the most commonly observed at JET.

An example of the temporal evolution of a discharge of this type is shown in Fig.43. This shows the longest duration H-mode achieved,

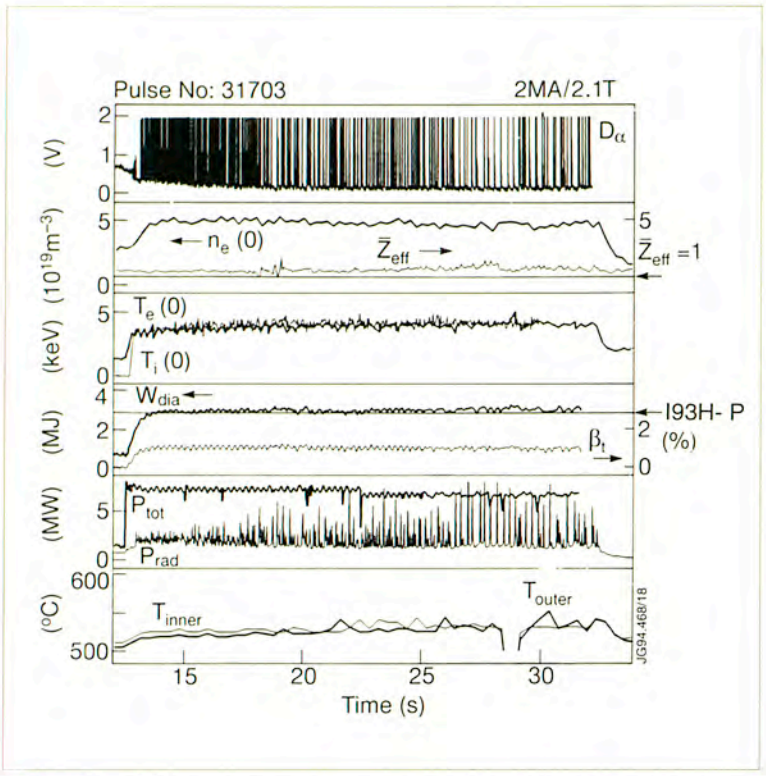


Fig.43: Time evolution of a long pulse steady-state ELMy H-mode at 2MA

with a pulse length of 20s, which was limited only by machine engineering constraints. The H-mode phase in this 2MA discharge lasts for ~50 energy confinement times (~50τ_E). The regime has been extended up to plasma currents of 3MA for at least 20 energy confinement times. The plasma density, the effective charge (Z_{eff}), stored energy, recycling conditions, plasma temperature, plasma beta and radiated power all remained constant for 20s. The radiated power (at 20-40% of input power) and Z_{eff} (~1.6) were typically low in the ELMy phase of these plasmas. An important result is that the divertor target temperature remained far below the design limit during the pulse.

During the H-mode phase of this discharge, plasma fuelling was by neutral beam injection alone, and the divertor cryopump controlled the density, with no saturation of the pumping. This contrasted with previous operation in 1991, when wall pumping saturated after about 10s and no true steady-state could be achieved. The energy confinement time was enhanced by a factor, H≈1.8, above L-mode confinement. The demonstration of these steady-state H-modes with edge safety factors, q_{95}, as low as 2.9 is a relevant ignition scenario and is a significant step along the road to the successful operation of ITER.

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.

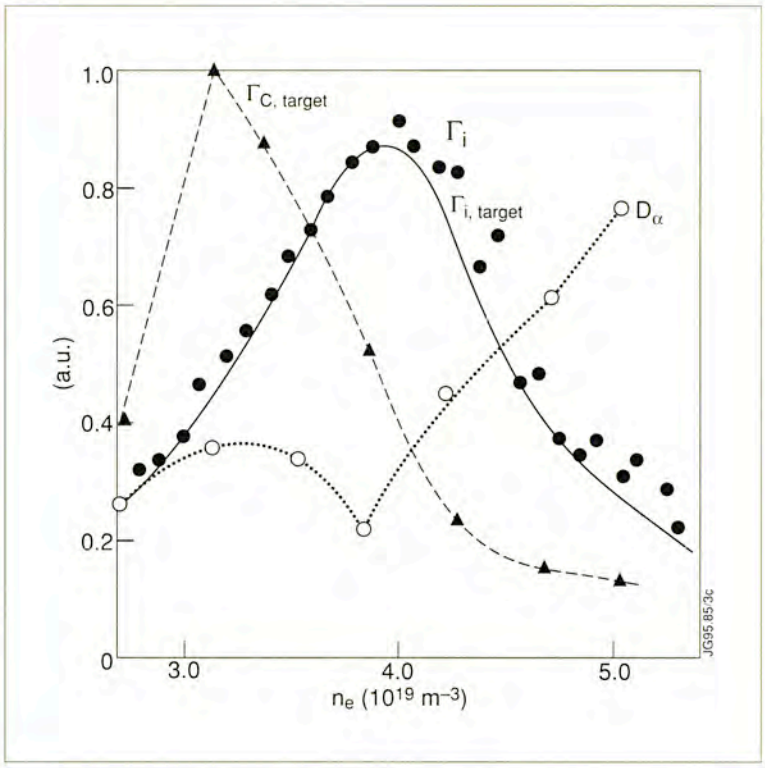


Fig.44: Time evolution of deuterium flux to target (Γ_i), recycling light (D_α) and carbon influx (Γ_c) showing development towards detachment at high density

Gas Target/Radiative Divertor Plasmas

While poloidal divertors have been employed successfully in many tokamaks to reduce impurity contamination of the plasma core, an operating scenario which scales to reactor plasmas has not yet been demonstrated. The exhaust of particles, momentum, and power from a burning plasma is such that, the divertor target plates would not survive unless a significant fraction of the exhaust is redirected across magnetic fields lines by volume loss processes. In addition, it has been shown that radiation alone cannot be used to solve the power loading problem in a so-called high recycling divertor where there is no loss of momentum along the scrape-off-layer (SOL) field lines. A pressure drop of a factor of ten is consistent with a radiative fraction sufficient to satisfy constraints on ITER target loads (~5MWm^{-2}).

When the core plasma density is increased by some fuelling method, the plasma density in the divertor typically increases, the dependence being as high as the third power. This staggering rate of increase is self-limiting, because at the same time the temperature in the divertor and the SOL is rapidly decreasing and losses, such as charge exchange and radiation, begin to dominate. At this stage, the density at

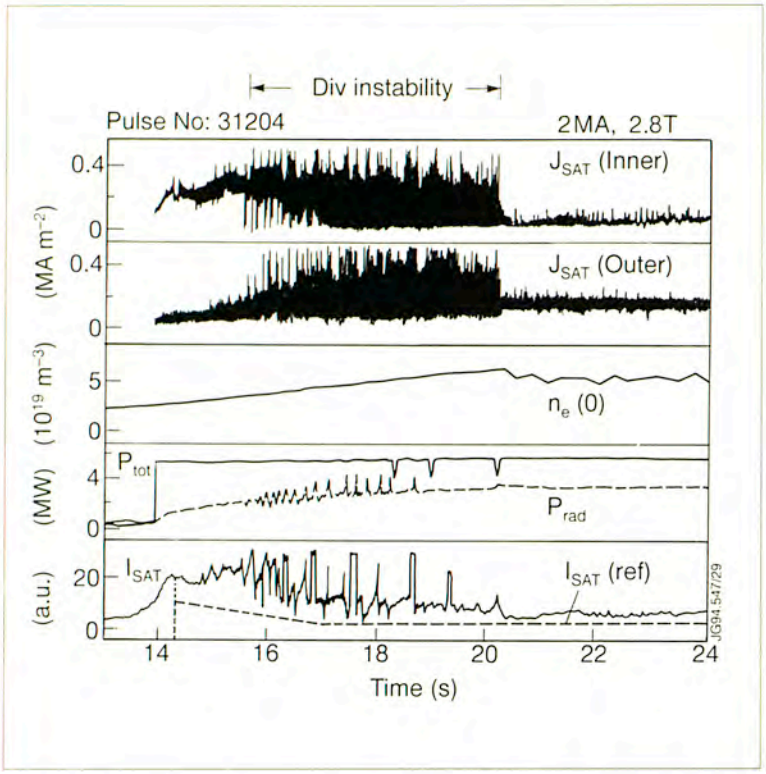


Fig.45: Time development to detachment in a 2MA L-mode discharge

the divertor plates decreases as the core density, is increased even further. This regime, where not only the density but also the flows of particles, momentum, and energy are decreasing functions of increasing core density, is the so-called detached plasma regime and is the favoured candidate for future burning plasma tokamaks.

However, the onset of volume losses is gradual as the SOL/divertor temperature drops and presents a problem of defining exactly when a plasma is detached. Complete detachment and the concomitant reduction of heat flux to the divertor target plates was first seen during the 1991/92 campaign. With the Mark I divertor, this regime has been extended to cover a broad range of plasma conditions and magnetic configurations at high density in ohmic, L-mode and H-mode plasmas.

Figure 44 shows results from an ohmic detached discharge where the density was increased from $3\times10^{19}m^{-3}$ to $6\times10^{19}m^{-3}$ over several seconds. As the density increased, the particle flux density (Γ_i) to the target first increased (the high recycling regime), and then decreased dramatically with the onset of detachment. The recycling light (D_α) also increased as the number of excitations per ionisation increased rapidly at very low temperature. The carbon influx (Γ_c) from the divertor plasma temperature fell below that required for significant

Divertor

JET was originally configured as a "limiter Tokamak" where the edge of the plasma (the "last closed flux surface" - LCFS) was defined by contact with a material boundary called a limiter, which absorbs the exhaust power of the plasma. Since the edge of the plasma is quite hot, material is eroded by sputtering and the sputtered impurities enter the plasma relatively easily. This enhances radiative losses and dilutes the plasma, which lowers the fusion reaction rate.

The JET vacuum vessel and magnetic field system have been modified to operate in a "divertor" mode. The field configuration includes an "X-point" so that the LCFS (in this case designated the separatrix) bounding the main plasma does not intersect the wall. The power crossing the separatrix is transmitted in a thin layer called the scrape-off layer (SOL) to the divertor at the bottom of the vessel and is absorbed by the divertor "target plates". Divertor operation reduces the impurity content of the main plasma through a combination of effects. The divertor plasma is generally much cooler than the main plasma edge, so that sputtering and erosion are reduced. Moreover, impurities which are produced at the target plate tend to be retained in the divertor area by friction with the plasma streaming towards the divertor plates. In addition to controlling impurity content, divertor operation tends to allow higher SOL temperatures, thus facilitating access to high confinement regimes.

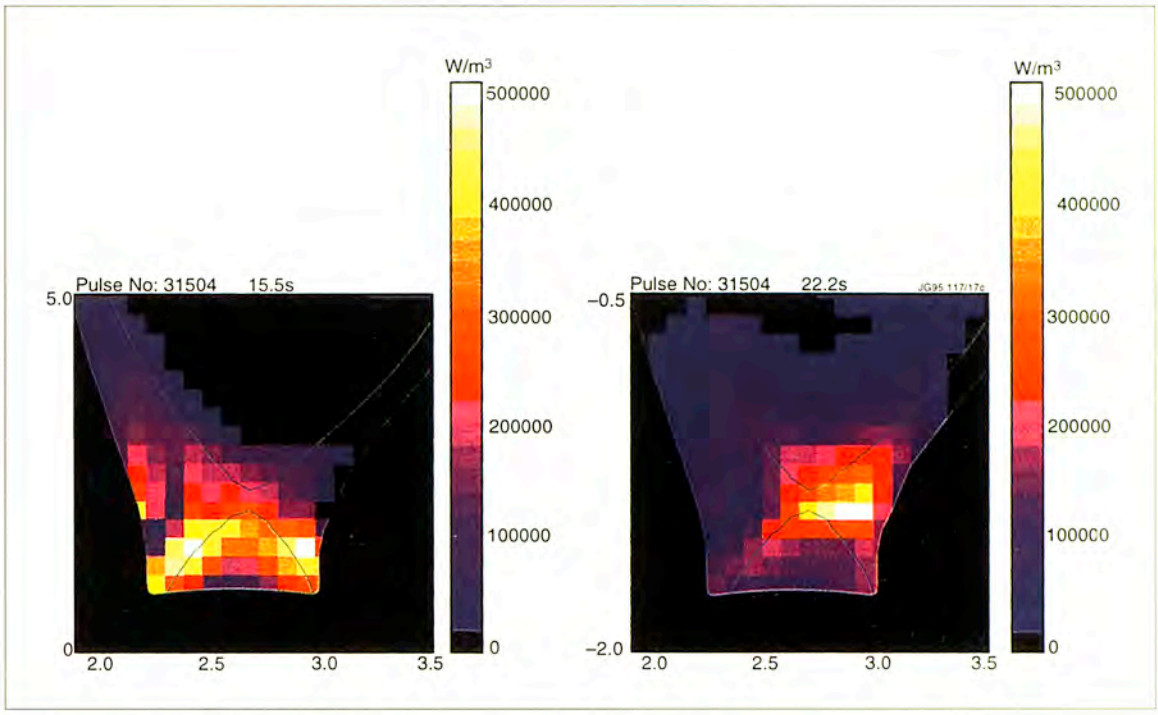


Fig.46: Radiation distribution in (a) attached state and (b) detached state

sputtering, and then decreased further with decreasing temperature and particle flux during the detached phase.

Figure 45 shows an L-mode discharge in which detachment was achieved at a power level of 5MW. The gas input was controlled by feedback from the ion saturation current of one of the Langmuir probes close to the inner target. There was a gradual fall in the ion saturation current between 16s and 20s, while the density was increasing. The pressure at the probe fell while the total radiated power was about 50%. Plasma radiation was localised near the target plates during the attached phase of such discharges, and near the X-point during the (partially) detached phase (Fig.46). Studies are in progress to control the location of the radiating region. Attempts to achieve complete detachment in H-mode plasmas, by puffing deuterium into steady state H-modes, precipitate an increase in the ELM frequency, and eventually the plasmas return to L-mode before complete detachment can occur.

A step forward was achieved by puffing a mixture of nitrogen and deuterium into the plasma. With this mixture, it was possible to maintain quasi-steady-state ELM behaviour which had not been possible, with a mixture of neon and deuterium. The quasi-steady-state regime could be established with ~80% radiation (Fig.47) and initial analysis showed that about 70% of this was emitted from outside the separatrix. The degree

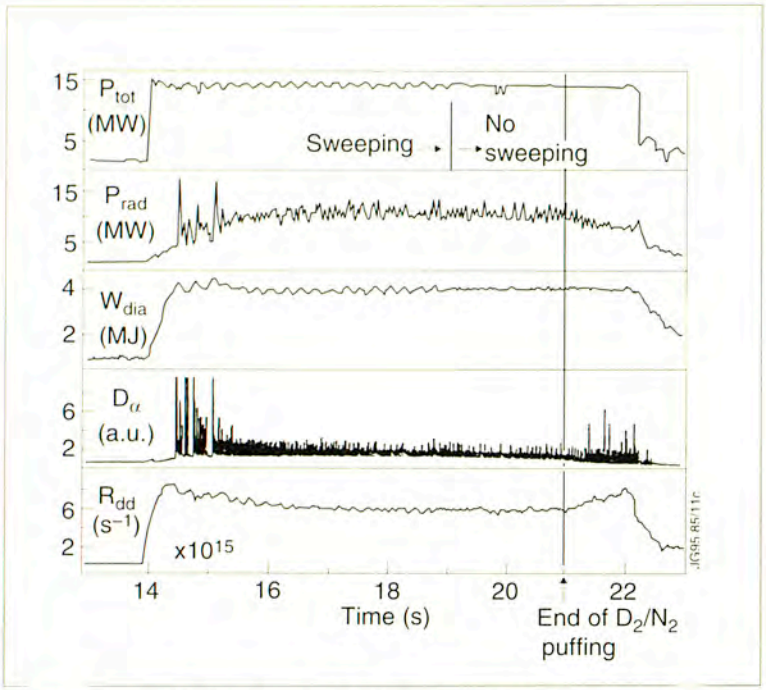


Fig.47: Time evolution of an ELMy H-mode with N_2/D_2 gas puffing showing achievement of quasi-steady-state ELM behaviour with ~80% radiated power

of detachment in these discharges is still being investigated, but it seems that there is complete detachment between ELMs, though each ELM results in a "burn through" of the divertor plasma and a spike in the ion saturation current. The confinement enhancement in these discharges was relatively modest (H≈1.4). The H factor increased, however, when ICRF power was coupled successfully to these discharges, indicating, perhaps, a dependence on heating profile. So far, combined heating powers of up to 24MW have been coupled into these plasmas with the divertor target tile temperature remaining constant at 350°C, without sweeping, for the 2.5s duration of the heating pulse. Further studies at higher combined heating power should provide the most convincing demonstration of the relevance of this regime to ITER.

High Performance

The major objective was to develop plasmas with high core plasma performance in the new configuration. This included the optimisation of parameters such as temperature, density, stored energy and fusion yield, with a view to defining regimes of relevance to future D-T operation in JET and ITER. The pumped divertor was designed for operation of diverted configurations with plasma currents up to 6MA with safety factors of, typically q_{95}~3 (and down to 2.2), in line with mainstream ITER

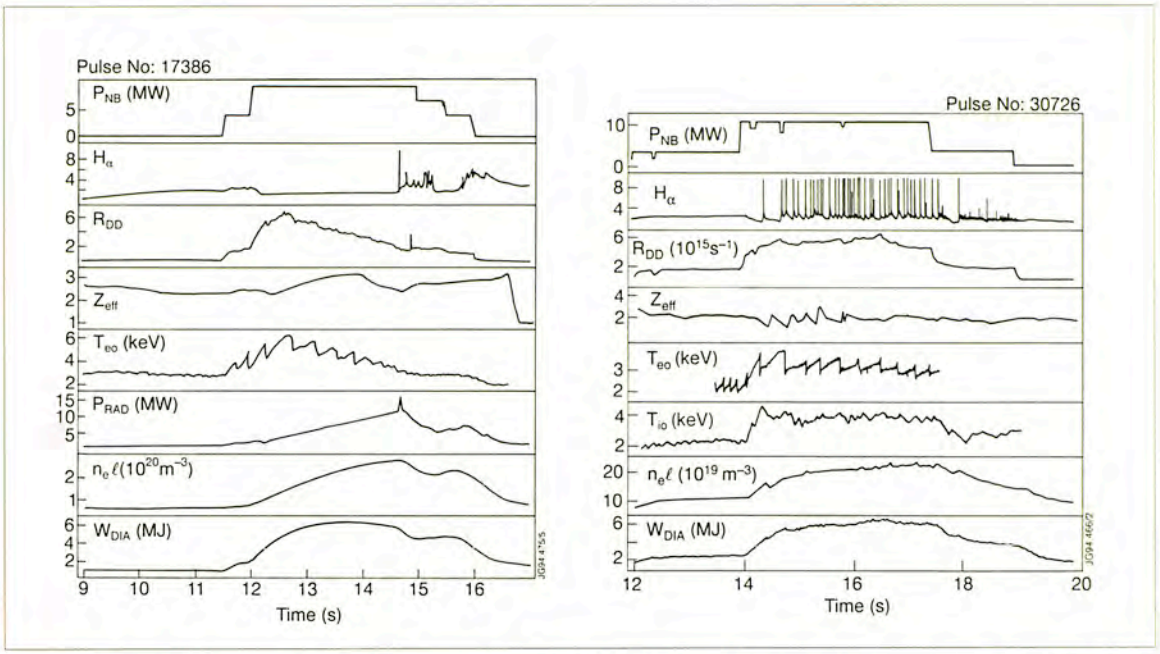


Fig.48: (a) Typical H-mode behaviour in the old JET for 10MW of beams into a 3MA plasma (Note the long ELM-free period): (b) Typical H-mode behaviour for standard FAT configuration in the new JET for 10MW of beams and plasma current of 3MA

design. The improved current capability in diverted configurations opened up the domain of H-mode operation and raised the question of scaling of confinement with plasma current at such low safety factor.

In the previous configuration, ELM-free H-modes of several seconds were the natural regime for any high power heated diverted configuration, as shown in Fig.48(a). Above a clearly defined threshold power, the plasma would enter and remain in this regime until, typically, increasing radiated power reduced the exhaust power below threshold. In this case, optimisation of performance required low recycling, density control, optimum beam deposition but were limited in plasma current. The hot-ion H-mode regime in the old JET was transient limited by carbon blooms, earlier appearance of giant ELMs and, in some cases, limited by sawteeth and other internal MHD.

In the new configuration, the natural H-mode regime is ELMy as shown in Fig.48(b). In this case, the initial ELM-free period is relatively short and is followed by repetitive large ELMs. These larger ELMs lead to steady-state conditions, which can be maintained for many energy replacement times. This is directly relevant to ITER, and, as such, has been exploited effectively. This difference in behaviour has provided a challenge and a rethink of strategy for performance optimisation has had to be implemented.

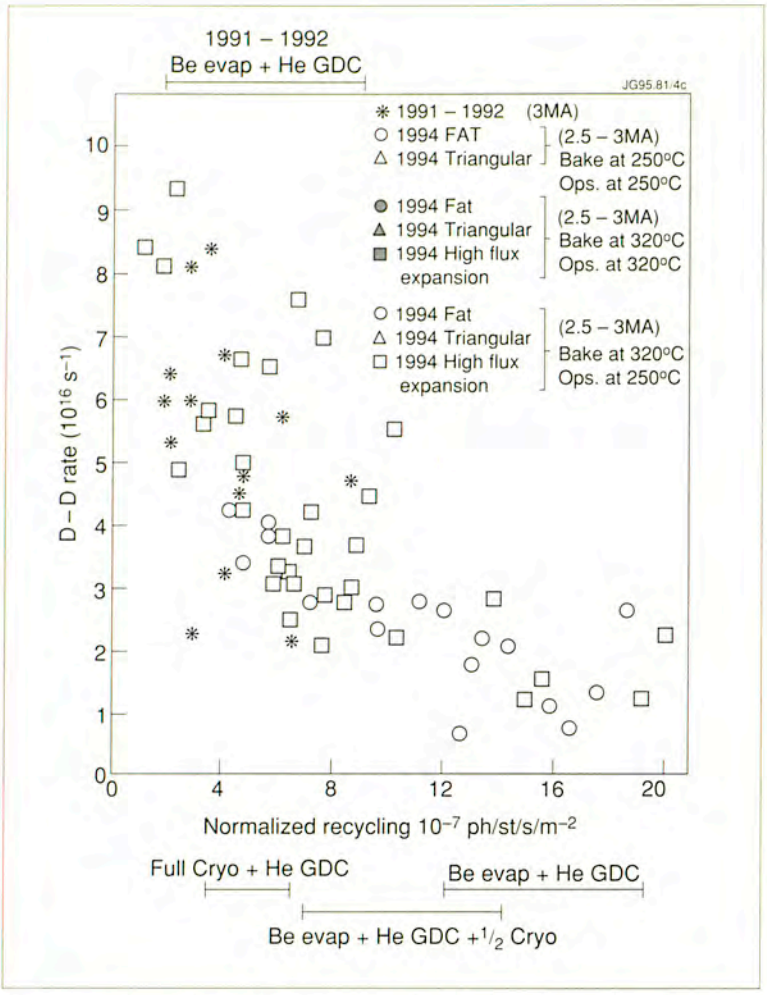


Fig.49: The dependence of D-D fusion reaction rate on normalised main chamber recycling flux. Data is for 2.5-3MA hot-ion H-mode plasmas from 1994 with 15-18MW neutral beam heating. Comparison is made with 3MA hot-ion H-modes from the 1991/92 campaign

A picture is emerging whereby the optimisation of core conditions is constrained by a complex web of interrelated physics. Progress has been made in understanding some of the key links, such as that between configuration and ELM-free period; between recycling and ELM-free period; and between recycling, refuelling and core density profiles.

The high heat load capability of the Mark I divertor target means that carbon "blooms", which previously terminated high performance, have been eliminated. This has shown, as previously suspected, that MHD modes, which precede fast collapses (ELMs) and slow "roll-overs" of storec energy and neutron yield, limit high performance. The achievement of high performance ELM-free H-modes is strongly affected by vessel conditioning, gas recycling from in-vessel components and the magnetic configuration. Recycling in the new configuration has been higher than previously and this has impacted on performance.

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.

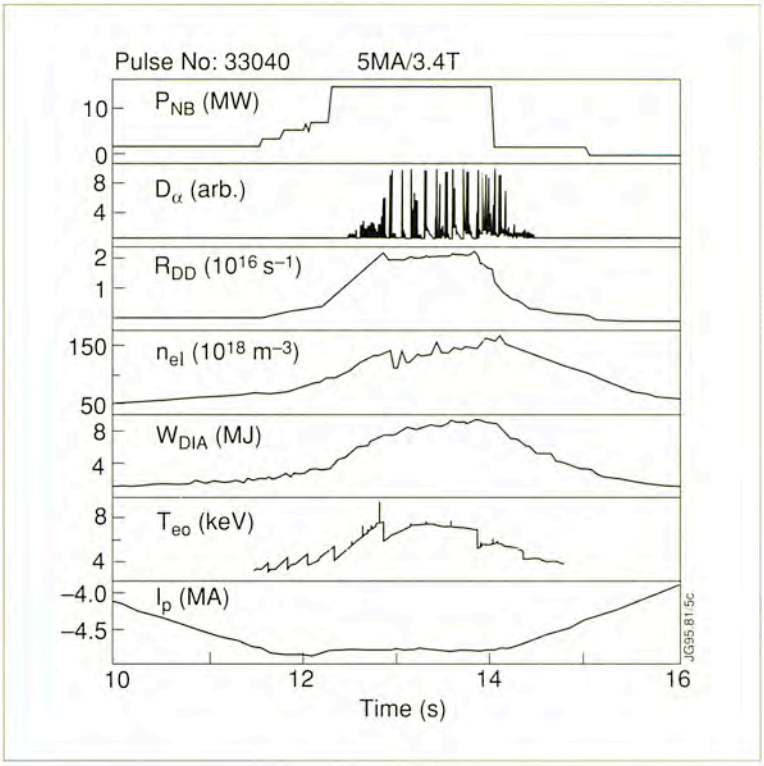


Fig.50: Time evolution of a 5MA ELMy H-mode with 15MW of neutral beam injection

Special attention has therefore been paid to identifying and reducing the source of higher recycling and thereby extending the ELM-free period and improving fusion yield.

In particular, the divertor targets and the vacuum vessel have been baked to high temperatures (200°C and 320°C, respectively) and the high pumping capacity of the cryopump has been used to deplete neutral gas reservoirs in the tiles and walls. Specific magnetic configurations ("high flux expansion" in the divertor) have been developed to reduce leakage of neutrals into the main plasma from the divertor. The importance of low recycling in achieving a high fusion yield is shown in Fig.49 where recycling levels in present hot-ion discharges are compared with those in past campaigns.

Specific studies have been undertaken to establish the role of the magnetic configuration in improving edge stability and reducing the occurrence of ELMs. Studies at constant input power, constant current and constant toroidal field showed that: high flux expansion at high edge shear increased the ELM-free period, while increasing edge shear and triangularity at high flux expansion also increased the ELM-free period.

In practice, the increased ELM-free period could not always be exploited for increased stored energy and neutron yield, since a slow

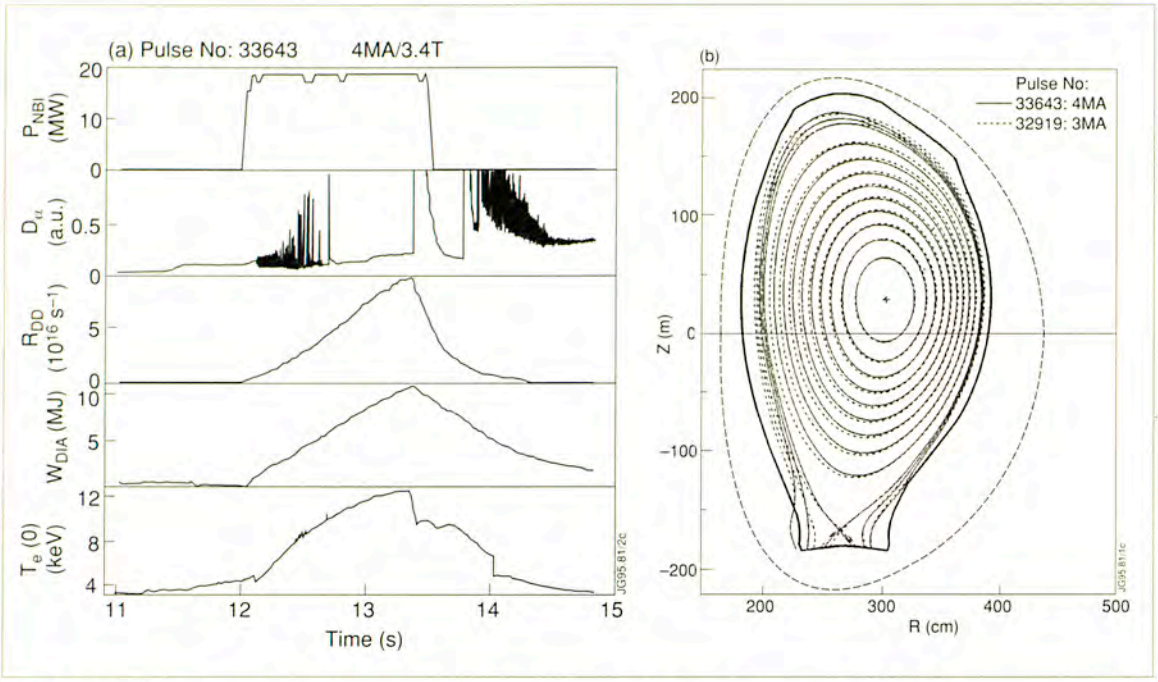


Fig.51: (a) Time evolution of discharge with highest fusion yield in deuterium; and (b) High flux expansion equilibrium of discharge with highest fusion yield in deuterium (compared with best 1994 discharge at 3MA)

"roll-over" often occured in these high performance plasmas. This phenomenon is accompanied by a variety of MHD activity. In these high performance discharges, it is this phenomenon (already noted in some of the best hot-ion H-modes of 1991/92) that limits plasma pressure to a normalised $\beta_N \approx 2$. In hot-ion H-modes at higher plasma current (3.5-4MA), the phenomenon was much weaker, or was entirely absent. The limitation then appeared to be the giant ELMs or sawteeth.

High Plasma Current, Stored Energy and Fusion Yield

Experiments in 1994 achieved plasma currents up to 5MA. At 5MA, ELMy H-modes were obtained with up to 15MW of neutral beam injection (see Fig.50). At 4MA current, steady-state ELMy H-modes lasting more than four energy confinement times, were obtained with more than 18MW of additional heating. The maximum stored energy was ~8MJ and the fusion triple product ($n_D T \tau_E$) reached $2.6 \times 10^{20} m^{-3} keV s$. The highest stored energy achieved was 11.3MJ at the end of the ELM-free period of a 4MA discharge with 19MW of additional heating.

The best fusion performance achieved in 1994 was comparable to the best of that in the past, even though the new plasmas were ~15% smaller in volume. The record neutron rate in 1991/92 deuterium plasmas was exceeded. In the ELM-free phase of a 3.8MA hot-ion

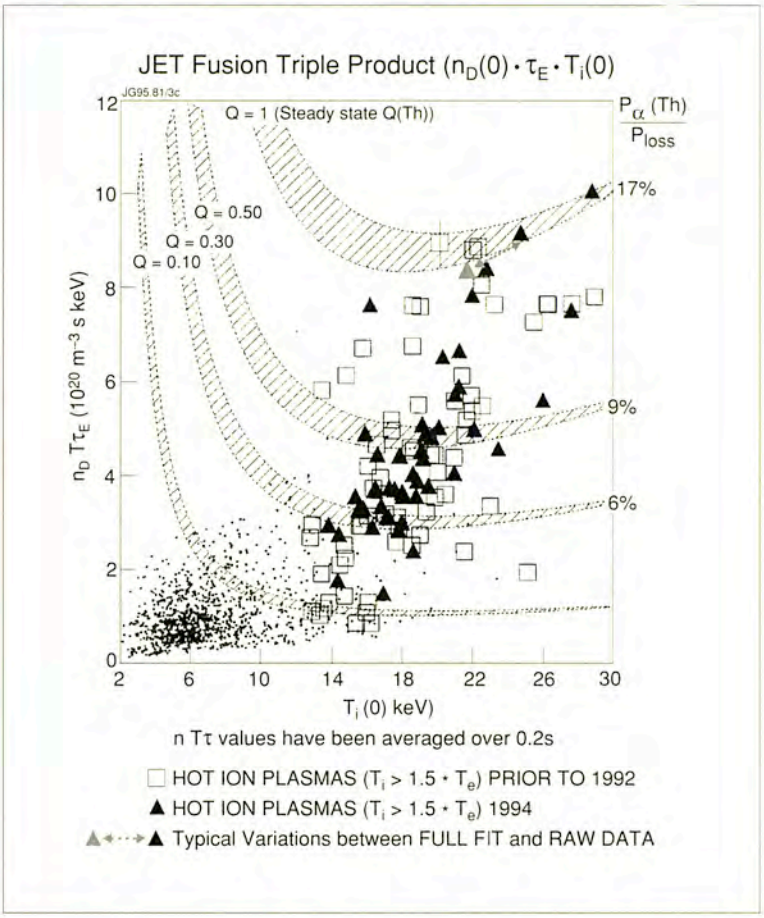


Fig.52: Fusion triple product for 1994 compared with previous campaigns

H-mode, the neutron rate reached $4.7 \times 10^{16} s^{-1}$, more than 10% higher than the previous best with deuterium in 1991/92. The highest fusion triple product ($n_D T_i \tau_E$) exceeded $8 \times 10^{20} m^{-3}$keVs, within ~10% of the previous best. The time evolution of this discharge is shown in Fig.51(a), together with its magnetic equilibrium (Fig.51(b)).

The fusion triple product achieved in a representative selection of 1994/95 discharges is shown in Fig.52, compared with a similar selection from earlier campaigns.

Advanced Tokamak Studies

Studies have been undertaken to investigate coherent steady-state tokamak concepts (the so-called Advanced Tokamak Scenarios) for ITER and DEMO. Advanced Tokamak scenarios for steady-state operation require a high bootstrap current fraction (>70%), high beta and good H-mode energy confinement. The broad current density profiles typical for high bootstrap current operation are prone to MHD instabilities. Active profile control is therefore required. Such a plasma would also need to exceed the Troyon β limit, if very high toroidal fields

Plasma Beta

The economic efficiency of a tokamak reactor is determined, partly, by the maximum plasma pressure which can be contained by the magnetic fields in the device. In particular, the important parameter is the plasma beta β_T, defined as the ratio of plasma pressure to the pressure of the confining magnetic field (β_T is proportional to nT/B_T^2, where n is the plasma density, T the plasma temperature and B_T the toroidal magnetic field). This limit expected theoretically, is the so-called Troyon limit $\beta_T(\%)=2.8 I_p(MA)/B_T(T)a(m)$, where I_p is the plasma current and a is the minor radius.

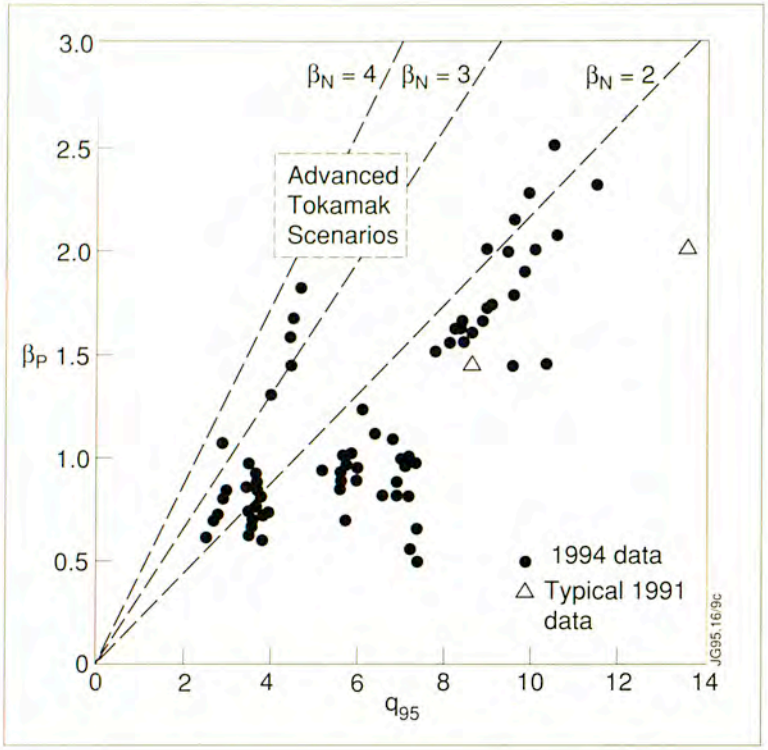


Fig.53: Achieved values of β_p plotted against the safety factor q_{95}

were to be avoided. Investigation of the stability and confinement characteristics of plasmas in this domain have been undertaken to assess the prospects for steady-state reactor concepts.

In previous experiments, a bootstrap current fraction of 0.7 was obtained in plasmas with $\beta_p \approx 2$ and high confinement compared with the usual L-mode and H-mode scalings. However, these discharges were obtained at $q_{95}>10$, which would be uneconomic and technically difficult to achieve in a reactor. In such plasmas, high values of β_p can be achieved without reaching the Troyon β limit. Nevertheless, these discharges were not stable and collapsed with a large ELM after ≈2s. The cause of the collapse has not yet been unambiguously identified.

The new pumped divertor configuration has several features which differ from the old configuration. The plasma volume is smaller and, consequently, for similar plasmas current and toroidal field, the safety factor, q, is lower. On the other hand, the power handling of the divertor has improved. Long ELM-free periods and very high confinement have not yet been obtained in high β_p experiments with the new configuration. While the cause of this is still under investigation, the benefits of ELMy H-mode plasmas have been exploited to meet the aim of achieving quasi-steady-state plasmas at least with respect to density control.

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.

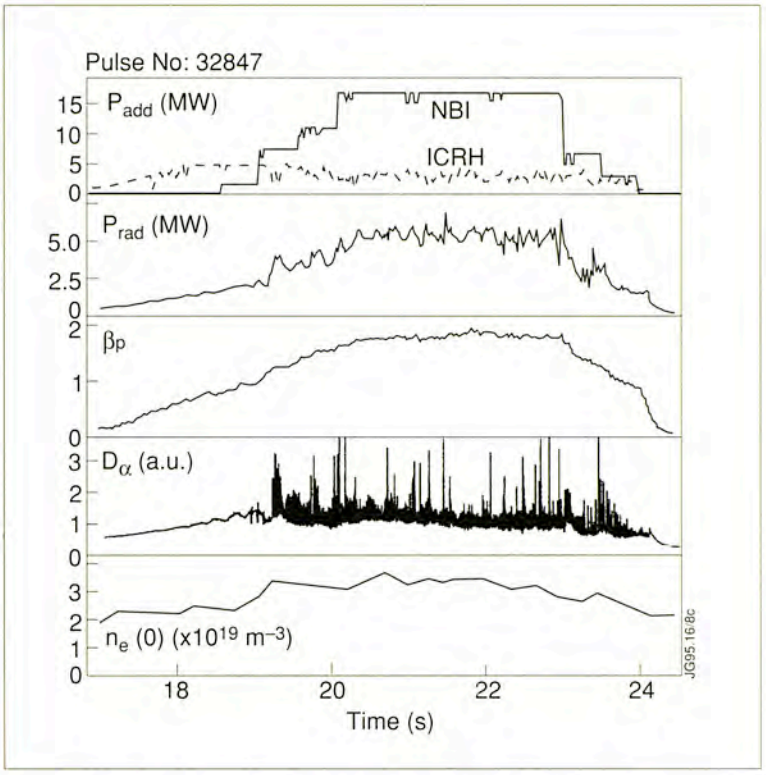


Fig.54: Time history of heating and plasma parameters for a high β_p pulse

High β_p plasmas have been investigated in a much wider range of plasma parameters during 1994 than in previous JET campaigns. Figure 53 shows the achieved values of β_p plotted against the safety factor q_{95}, and typical data from previous experiments are also shown for comparison. The 1994 data are significantly closer to the 'Advanced Tokamak' domain; whereas previous JET data were obtained at modest values of β_N, values up to 3.7 have now been achieved.

The time evolution of a typical high β_p plasma with a plasma current of 1MA and toroidal field of 2.8T is shown in Fig.54. The combined heating power of the neutral beam ICRF heating systems was 20MW, which resulted in β_p value of 2.0. However, in previous experiments, this value had been achieved with <10MW of heating power in an ELM-free very high confinement (VH) mode plasma. The plasma has reached quasi-steady conditions with respect to stored energy, radiated power and density, whereas the previous VH-mode conditions were achieved only transiently.

The bootstrap current fraction calculated for these pulses is about 0.6, somewhat lower than previously. This is largely due to the higher non-thermal fraction of the plasma stored energy (not included in the bootstrap current calculation), which results from the increased

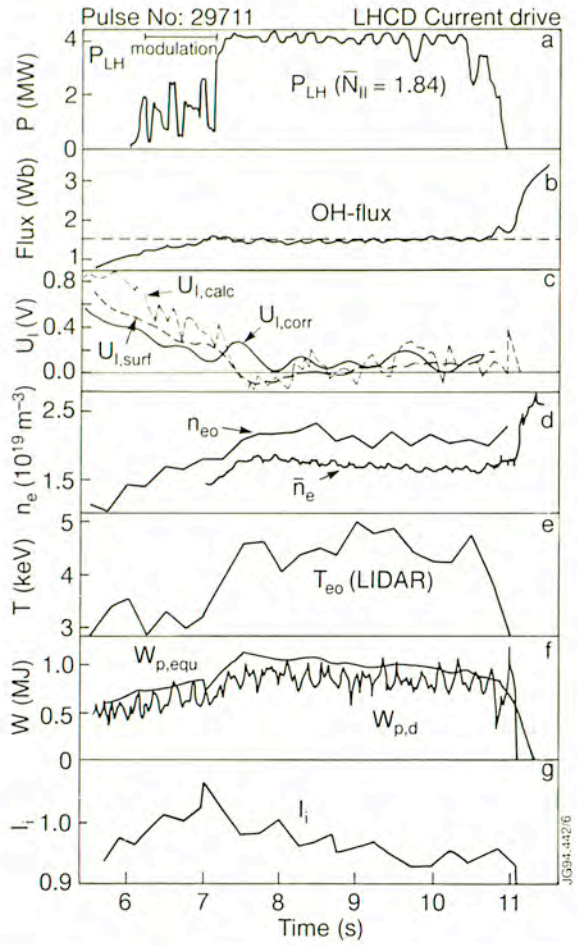


Fig.55: LHCD discharge at I_p=2MA,B_t=2.8T: (a) LH power, (b) OH flux consumption, (c) measured, corrected and calculated loop voltage, (d) central and line averaged electron density, (e) central electron temperature, (f) energy content from diamagnetic and equilibrium field measurements and (g) internal inductance

additional heating power. In some cases, around 30% of the plasma stored energy is estimated as non-thermal. The calculated beam driven current can account for about 20% of the plasma current.

Three high power noninductive current drive systems are available now on JET: lower hybrid current drive (LHCD), fast wave current drive (FWCD), and high energy neutral beam current drive (NBCD). Experiments have been performed with the full LHCD system in the plasma current range of 0.7-3MA. Maximum LH power of 6.5MW and a energy of 36MJ have been coupled to the plasma. A representative 2MA discharge at a line averaged density of $\langle n_e \rangle = 1.8 \times 10^{19} m^{-3}$ is shown in Fig.55. The LH-current drive with 4MW coupled power replaces the ohmic flux consumption. The measured surface loop voltage $U_{l,surf}$ drops to zero. The electron temperature rises from 3 to 5keV and the total

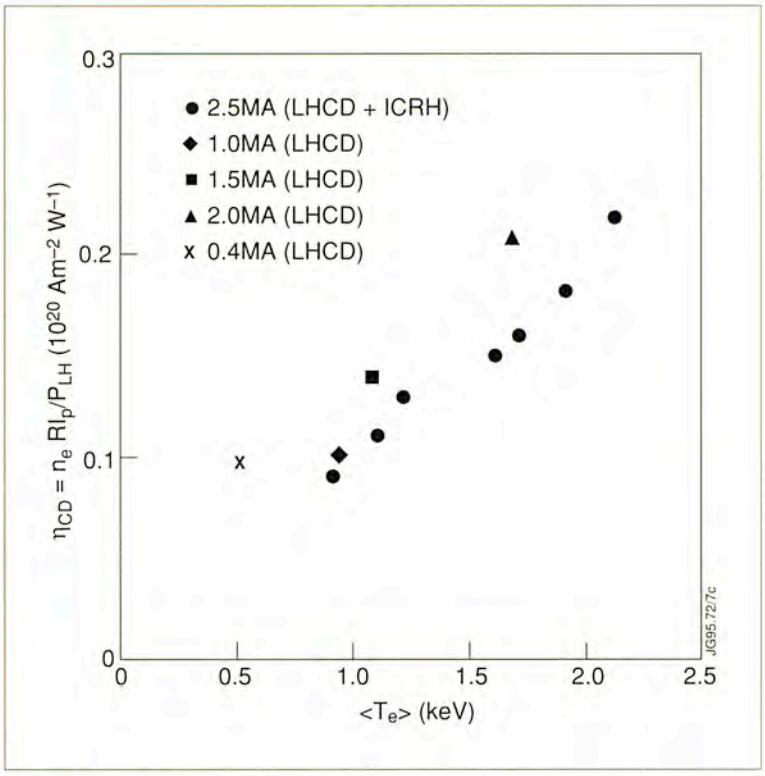


Fig.56: Scaling of LH-current drive efficiency with volume averaged electron temperature

energy content doubles. A decrease of the internal inductance, ℓ_i, indicates a broadening of the current profile. Full current drive conditions have been achieved at plasma currents up to 2.5MA.

The current drive efficiency, η_{CD}, increases with electron temperature, for LHCD alone and in combined heating/current drive scenarios. The scaling of η_{CD} with the volume averaged temperature is shown in Fig.56 for full current drive discharges with LH alone and for partial and full current drive in combined operation with LHCD and ICRF heating over a wide range of powers.

Current Profile Control

A main goal of lower hybrid current drive application on JET is the active control of the plasma current profile in order to improve MHD stability and energy confinement and to explore scenarios for tokamak performance improvement in combination with other heating and current drive methods. In a first stage, the extent of deposition profile variations and the resulting modifications of the current profile were assessed.

The current profile is broadened in most cases with LHCD, and a strong dependence of the LH deposition profile was found with

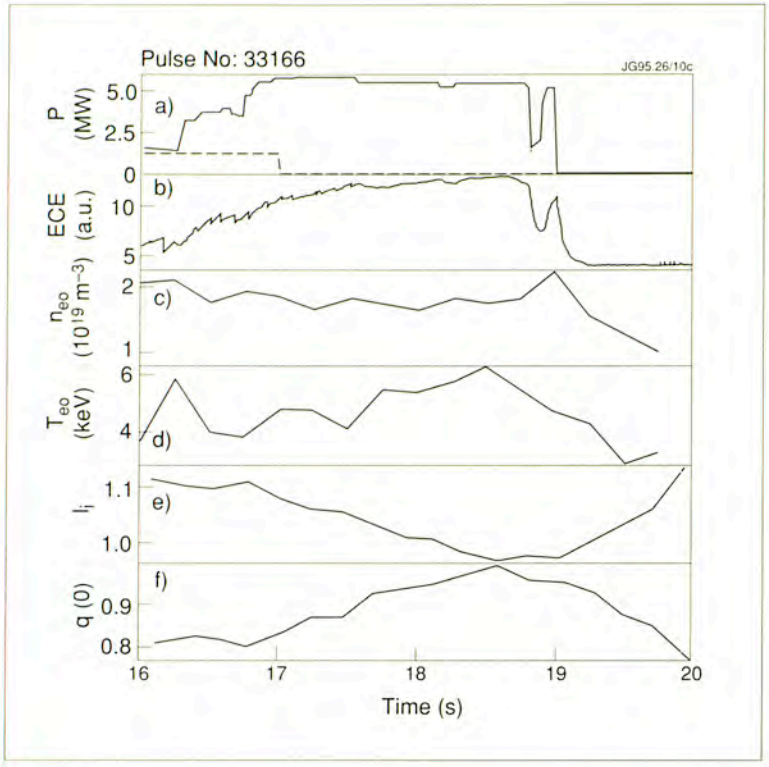


Fig.57: Temporal evolution of LH power, central ECE emission, $n_e(0)$, $T_e(0)$, internal inductance and q(0) during sawtooth stabilisation with LHCD

plasma current. Stabilisation of sawtooth instabilities was achieved with nearly full LH current drive. The waveforms of a 3MA discharge at a density $n_e(0) \approx 2 \times 10^{19} m^{-3}$ are shown in Fig.57. The LH power was ramped up within 1s to 5.5MW. The last sawtooth was seen on the ECE signal 1.5s after start of the LH wave. m=1 oscillations continued for about 0.5s and then disappeared, but only transiently. The central electron temperature rose, with an increase from 3.8 to 5.4keV.

Toroidicity-induced Alfven Eigenmodes

Toroidicity-induced Alfven Eigenmodes (TAEs) can be driven unstable by fast particles originating from neutral beam injection, ICRF heating or fusion reactions. Destabilisation of such modes in reactors is a subject of concern as they may lead to anomalouly rapid losses of energetic alpha-particles, reducing alpha heating efficiency and potentially damaging the first wall. These modes are characterised by high frequency (typically 100-200kHz in JET), which depends on the Alfven speed.

First experiments have been carried out, in which Alfven eigenmodes have been externally excited using the saddle coils driven by a high frequency amplifier. In ohmic plasmas, the Alfven nature of the

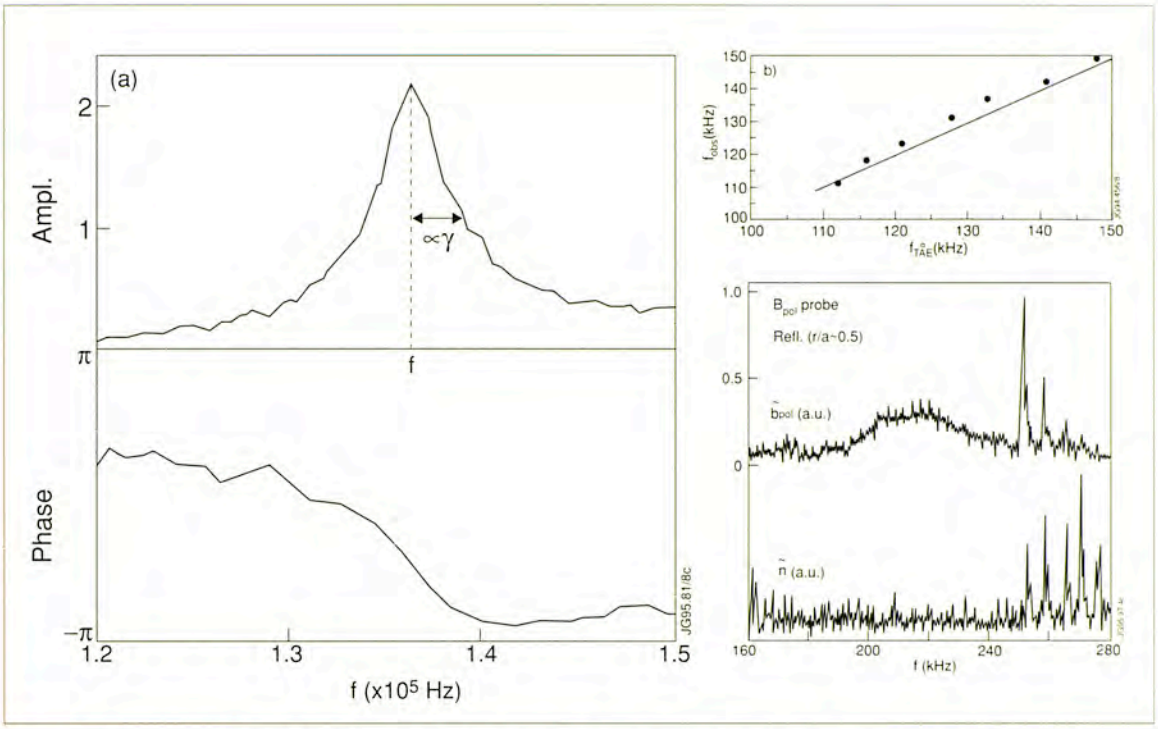


Fig.58: (a) Resonance detected at the Alfven frequency when plasma is excited using the Saddle Coils; (b) resonant frequencies observed in toroidal field scan compared to calculated Toroidal Alfven Eigenmode frequencies; and (c) multipeak structure of resonant frequencies on magnetic and reflectometer data above the Alfven frequency

observed resonances has been verified by scanning the toroidal magnetic field and comparing the measured resonance frequency with that calculated (Fig.58(a and b)). The measured damping of the modes (γ/ω in the range 0.1 to 0.01) are being compared with predictions. With 6MW of ICRF and 2.5MW of LH powers ($T_e/T_i>2$ and $\beta\approx0.3\%$), a strong multi-peak structure has been seen on magnetic and reflectometer data (Fig.58(c)) at frequencies above the continuum damping and with very low damping. This has been identified as the first observation of kinetic toroidicity-induced Alfven eigenmodes. Experiments will continue in 1995 to observe their effect on the plasma.

Next Step Issues

Certain work has been carried out on issues of particular relevance to Next Step devices, especially for ITER. The main highlights are set out below.

ITER H-mode Database

Energy confinement predictions for H-mode operation in ITER require a scaling law based on tokamaks of different dimensions. JET has

continued to add new data to the H-mode database for global confinement scaling, throughout 1994, at the request of the ITER Project. The work is performed as a combined effort from JET and from other tokamaks (DIII-D (General Atomics, USA), ASDEX (IPP Garching, FRG), JFT2M (JAERI, Japan) PBXM and PDX (PPPL, USA)). The database now contains measurements from a variety of heated H-modes (electron cyclotron resonance, ICRF and neutral beams). The scaling of thermal H-mode confinement using two-term power law models has been studied. A linear scaling has been assumed for the thermal energy content, W_{th}, of the form $W_{th}=W_o+\tau_{inc}P$, with W_o, τ_{inc} and P being the offset incremental confinement time and loss power, respectively. Using the standard dataset of the H-mode database, the following ELM-free and ELMy power laws for W_o and τ_{inc} have been obtained from non-linear least squares regression:

ELM-free:

$$W_o=0.0117\ I^{1.0}B^{0.5}n^{0.75}M^{-0.25}R^{3.2}\kappa^{0.05}\ (a/R)^0$$

$$\tau_{inc}=0.0493\ I^{1.0}B^{-0.3}n^{-1.0}M^{2.0}R^{-1.0}\kappa^{2.3}(a/R)^0$$

ELMy:

$$W_o=0.0055\ I^{0.67}B^{0.2}n^{0.8}M^{-0.1}R^{4.0}\kappa^{1.5}(a/R)^{0.3}$$

$$\tau_{inc}=0.0851\ I^{1.05}B^{-0.1}n^{-0.45}M^{1.5}R^{-1.1}\kappa^{0.35}(a/R)^{0.3}$$

The units are [W_o(MJ), t_{inc}(s), I(MA), B(T), n, M, R(m), k].

ITER L-mode Database

The L-mode database has been expanded by adding more information about the experimental configurations, conditioning, heating and radiative power, and fast ion energy content. The database has also been extended by new data from various devices. These changes have made the L-mode database more complete and compatible with the ITER H-mode database. The updated L-mode database consists of data from eleven different tokamaks (ASDEX, DIII, DIII-D, FTU, JET, JFT-2M, JT-60, PBX-M, PDX, TORE-SUPRA and TEXTOR).

The scaling analyses of total and thermal confinement times have not yet been completed.

ITER H-mode Threshold Database

The H-mode threshold database has been extended and improved by the addition of new data. The database now consists of data from

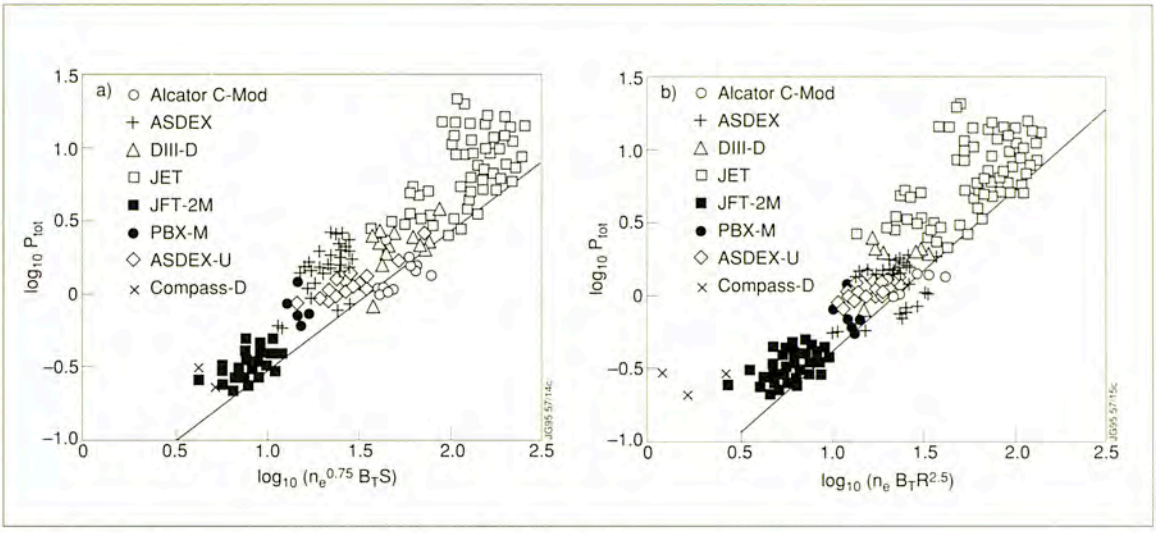


Fig.59: Tests of the variations (a)$P \sim n^{0.75}BS$, and (b) $P \sim nBR^{2.5}$

ALCATOR C-MOD, ASDEX, ASDEX Upgrade, COMPASS-D, DIII-D, JET, JFT-2M and PBX-M. It has been suggested that the threshold power increases linearly with the product of density (n) and toroidal field (B) for each device and that the plasma surface area, S, could be used to unify the multi-machine database. However, the form P~nBS is not dimensionally correct, but it can be made to depend only on dimensionless parameters. Two main forms have been investigated:

$$P \sim n^{0.75}BS \qquad (1)$$

$$P \sim nBR^{2.5} \qquad (2)$$

Figure 59(a) and (b) shows tests of Eqs (1) and (2). The straight lines are approximate lower bounds to the threshold power, and are given by the expressions $P=0.035n^{0.75}BS$ (Fig.59(a)) and $0.4nBR^{2.5}$ (Fig.59(b)) in units of $10^{20}m^{-3}$, T, m^2, m. Assuming $n=5x10^{19}m^{-3}$ for ITER, the first scaling yields a threshold power for the H-mode of ≈100 MW, while the second gives ≈200MW.

Toroidal Field Ripple Experiments

In 1991, JET carried out an experiment in which only one of two sets of toroidal field coils was energised. This increased the toroidal field (TF) ripple at the plasma edge from about 1% to 10%. The effects of high toroidal field ripple on plasma behaviour in general, and on fast particle losses, in particular, were studied. Results from these experiments were sufficiently encouraging to prepare for a second ripple experiment. In this case, the two sets of coils are operated at different currents. This modification will make possible the exploration of the intermediate ripple regimes, with ripples in the

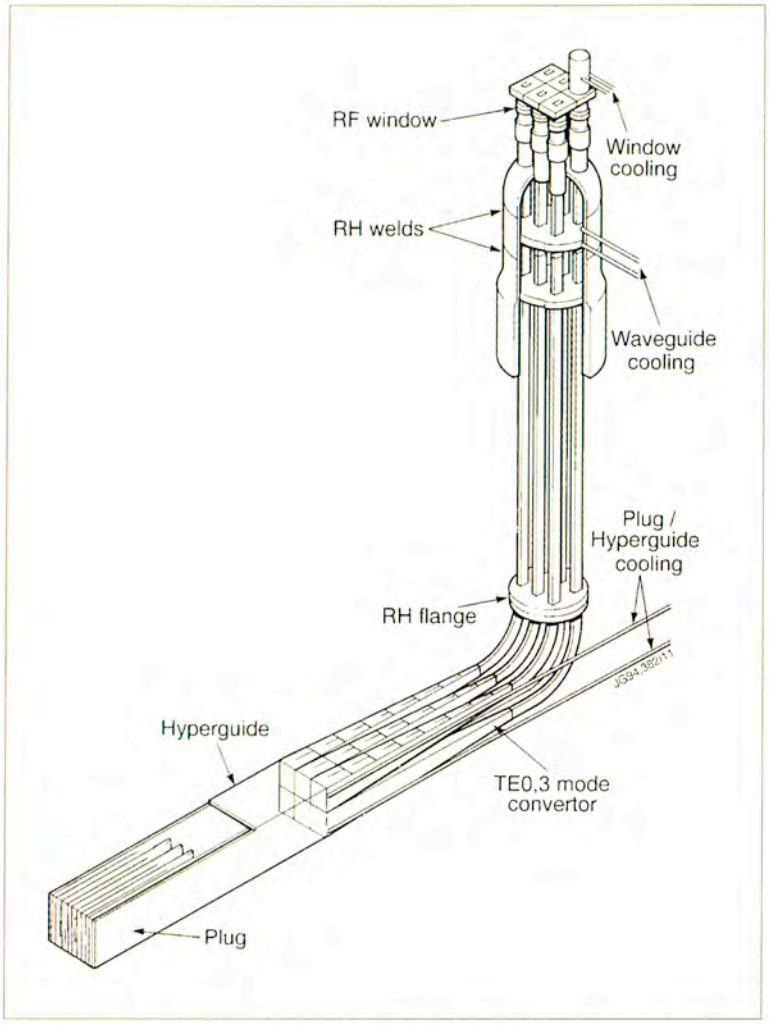


Fig.60: Schematic view of a module of a LHCD launcher design for ITER

range of 0.5% to 3%, in the pumped divertor configuration. The first ripple experiment showed that at the fixed high ripple level of 10% the plasma was strongly degraded. The second ripple experiment should allow quantification and investigation of threshold ripple values and other ripple effects. These experiments have strong relevance to Next Step devices, in general, as there will always be engineering pressure to minimise the number of toroidal field coils. The constraints imposed by the physics on this minimisation need to be established.

Technical preparation for this experiment is now well advanced. A study was made of the trajectories of lost particles and local heatloads on in-vessel components due to the ripple. As a result, maximum input powers and energies for the heating systems have been formulated.

LHCD Launcher Design for ITER

A conceptual design for a Lower Hybrid (LH) Heating and Current Drive launcher for ITER was made in cooperation with NET and CEA, Cadarache, France. Lower hybrid current drive has the feature to retain a good current drive efficiency in plasma regions of low temperature, or in low β plasmas. These aspects are important for the 'Advanced Tokamak' scenarios. The conceptual launcher operates at 5GHz and couples typically 20MW through one main horizontal port. It uses a combination active/passive waveguide grill with hyperguide and mode convertors feeds to give a structure which combines a robust grill with simple supply networks and good neutron attenuation at the first wall (see Fig.60).

Summary of Scientific Programme in 1994

The first plasma in the pumped divertor configuration was produced in mid-February and by mid-March successful 2MA divertor plasmas had been established. Subsequent achievements have been:

- the plasma current was increased to 5MA, the total heating power to 26MW, the stored energy to 11.3MJ and the neutron rate to 4.7×10^{16} neutrons per second;
- 1994 saw significant progress in optimising peak fusion performance (the best neutron rates of past campaigns in deuterium have now been exceeded) and extending operation to the reactor relevant steady-state ELMy H-mode, which has now been obtained under a variety of conditions (plasma currents up to 4MA, power levels up to 26MW, in the high β_p regime, in discharges with shear reversal, and at high β_N). The high β_p regime was also extended to steady-state and to the reactor relevant domain;
- the high power handling capability of the Mark I divertor target was demonstrated and the severe impurity influxes (carbon "blooms"), which previously terminated high performance plasmas, have been eliminated. About 140MJ of neutral beam energy was injected during ELMy H-mode plasmas, raising the surface temperature of the target tiles to less than 550°C;
- the cryopump improved vacuum conditions, reduced recycling, eliminated the effects of wall saturation (observed in previous long pulse operation), allowed effective particle control, and generally permitted higher performance;

- the two neutral beam injectors routinely injected up to 19MW of power into plasmas. 133MJ of power in a 20s ELMy H-mode represented a JET record;
- up to 13MW of ICRF power was coupled to JET plasmas. This power limit is set by a combination of unsatisfactory control electronics, unequal coupling of the straps of the antennae array and low power transfer to the plasma under some phase conditions. New control electronics systems are being installed and will be tested with plasmas in early 1995. Further antenna modifications are scheduled for the beryllium tile exchange and during the Mark II divertor shutdowns which are planned for mid-1995;
- up to 7.3MW, and up to 46MJ, of lower hybrid power has been coupled to plasmas using position control feedback. Up to 2.5MA has been driven non-inductively;
- the saddle coils have been used for initial experiments on TAE modes and the disruption feedback stabilisation system is in final stages of commissioning. Only the lower saddle coils are now available for experiments, since the upper saddle coils were diabled after being damaged in September 1994;
- the prototype high speed pellet launcher has been brought into operation and used to deliver its first deuterium pellets into plasma in commissioning pulses.

Progress towards a Reactor

An assessment of the fusion performance in the new pumped divertor configuration showed that, by the end of 1994, a D-D reactivity of $7.75 \times 10^{16} s^{-1}$ had been achieved with an input power of 18MW in a plasma with 3MA current and toroidal field of 3.4T. This gives a fusion efficiency, Q_{DD}, of 2.6×10^{-3}, which would correspond to an equivalent thermonuclear fusion efficiency (if deuterium and tritium mixture had been used) of Q_{DT} ~0.6.

The triple fusion product ($n\tau_E T_i$) obtained was $7.4 \times 10^{20} m^{-3} s$ keV, which was about 75% of the previous best value obtained during the 1991-92 campaign. It is expected that further progress will be made in increasing this and the thermonuclear fusion efficiency in the continuing campaign during 1995.

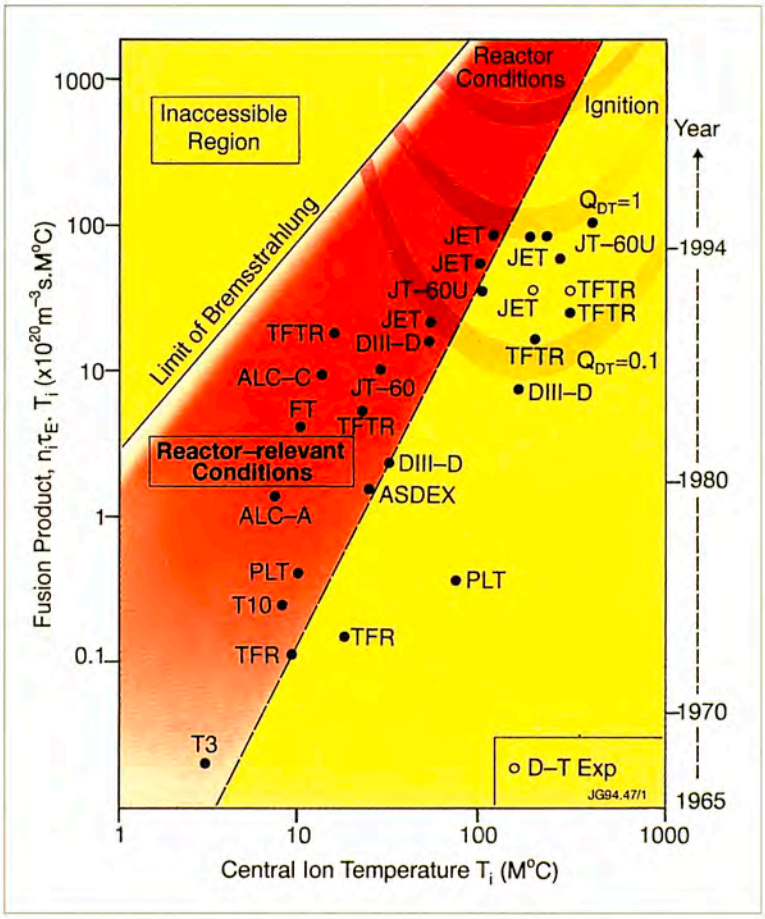


Fig.61: Triple fusion product as a funtion of ion temperature, T_i, for a number of tokamaks worldwide

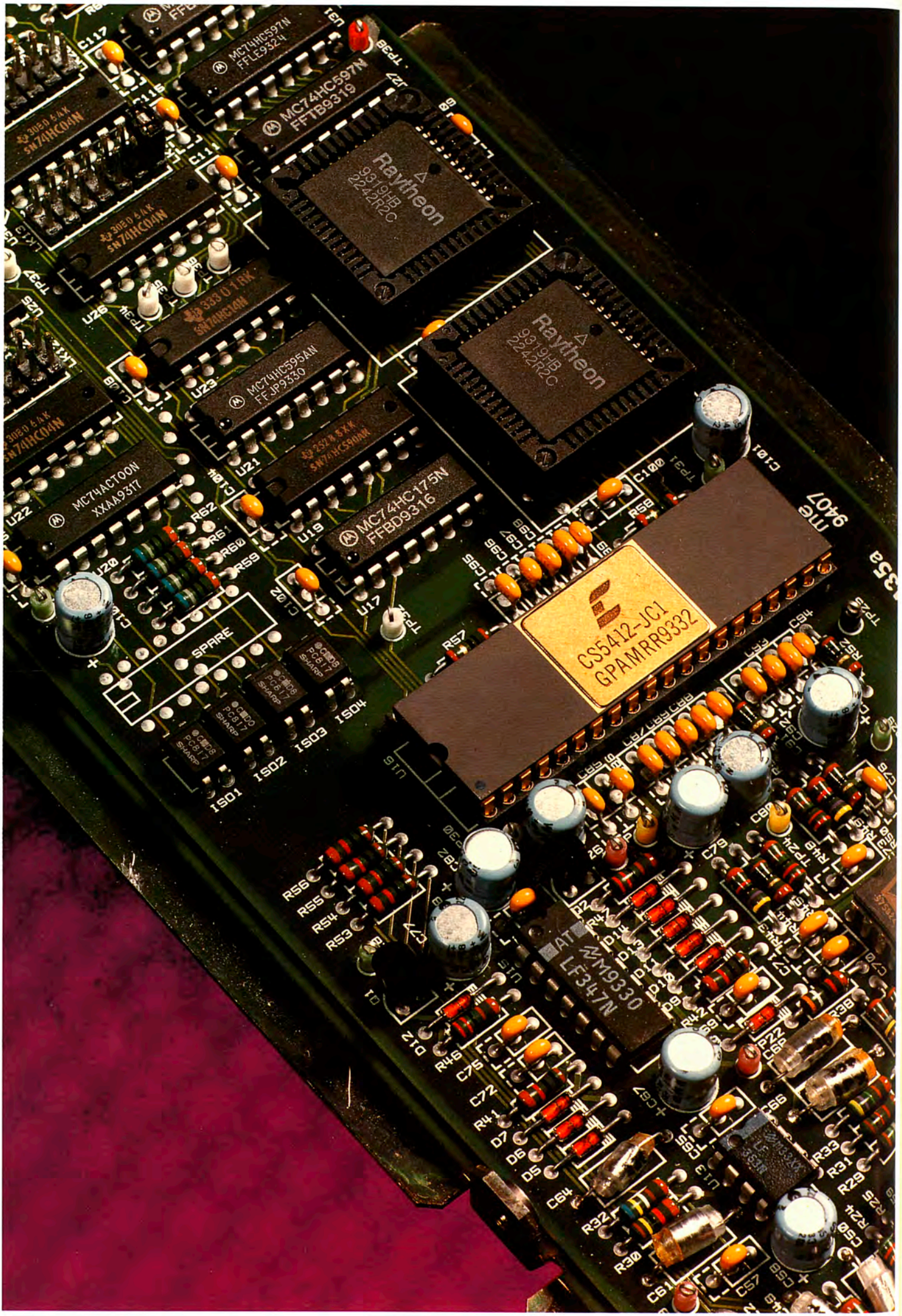
The carbon bloom, which was a restriction on JET performance in the old divertor configuration, has been eliminated by the new divertor. However, at high values of the stored energy (W~12MJ), the fusion performance is limited now by a variety of MHD phenomena (slow rollover, giant ELMs, etc). A comprehensive experimental and theoretical study of these events is being undertaken in an effort to alleviate the problem.

During 1994, considerable progress was made on other experiments world-wide. In the TFTR machine at PPPL, USA, experiments have continued using ~50:50 mixtures of deuterium and tritium. A fusion power exceeding 10MW has now been reached transiently, with an input beam power of 39MW into a "supershot" plasma. The improved yield has been obtained by increasing the toroidal magnetic field to 5.5T and the plasma current to 2.7MA.

In the large Japanese tokamak, JT-60U, the highest fusion product ($n\tau_E T_i$) of $12.3 \times 10^{20} m^{-3}$keV has been achieved at an ion temperature of

41keV, with an equivalent Q_{DT} value of ~0.6. In addition, a quasi steady-state ELMy H-mode was produced in deuterium with a high fusion product of $4.5 \times 10^{20} m^{-3} keV$ and an equivalent Q_{D-T} of 0.25-0.36.

The fusion triple product values of the high performance pulses in both impure deuterium and in the D-T pulses for JET and TFTR are compared in Fig.61 with the latest results from other machines world-wide to illustrate the progress that has been made over the last 30 years.


MC74HC597N
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MC74HC595AN
FFJP9330
MC74ACT00N
XXAA9317
MC74HC175N
FFBD9316
CS5412-JC1
GPAMRR9332
SPARE
ISO1
ISO2
ISO3
ISO4
PC817
SHARP
M9330
LF347N

Future Programme

Introduction

In 1978, the original objectives of JET were set out in the JET Design Proposal, EUR-JET-R5, as follows:

'The essential objective of JET is to obtain and study a plasma in conditions and dimensions approaching those needed in a thermonuclear 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 alpha-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 1994, JET neared the end of the longest and most extensive modifications since initial assembly of the device. During the

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.*

long shutdown, the interior of the vacuum vessel was essentially replaced. Following completion, JET was ready to begin its planned programme to demonstrate effective methods of power exhaust and impurity control in conditions close to those envisaged for ITER.

Since the beginning of its experimental campaign, extensive studies had been made in the first and third areas of work of JET's objectives: reactor relevant temperatures (up to 30keV), densities (up to $4x10^{20}m^{-3}$) and energy confinement times (up to 1.7s) 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 ionic discharge 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.

During 1991, the JET Council had 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, after having obtained all necessary regulatory approvals, JET successfully carried out a preliminary tritium experiment (PTE-1) in November 1991. A release of fusion energy in the megawatt range in a controlled fusion device had been achieved for the first time in the world.

In the 1991/92 campaign, JET achieved plasma parameters approaching 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 was due to the poisoning of the plasma by impurities (the 'bloom'). This further emphasised the need to provide a scheme of impurity control suitable for a Next Step device.

In late 1991, the Council of Ministers approved 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.

During 1992/93, a large effort was devoted to shutdown work for the pumped divertor phase. The first stage of the shutdown in 1992 had involved replacement of faulty toroidal magnetic field coils. The second stage in 1992/93 involved assembly of the four divertor coils and casings inside the vacuum vessel. The third stage began in mid-1993, with the final positioning of the coils. The shutdown was successfully completed in January 1994. The first plasma in the Pumped Divertor Characterisation Phase was produced in mid-February and by mid-March 2MA divertor plasmas had been established. During 1994, the plasma current was increased to 5MA, the heating power to 26MW, the stored energy to 11.3MJ and the neutron rate to 4×10^{16} neutrons/s.

1994 saw significant progress in optimising peak fusion performance and extending operation to the reactor relevant steady-state ELMy H-mode, which has now been obtained under a variety of conditions (plasma currents up to 4MA, power levels up to 26MW, in the high β_p regime, in discharges with negative central magnetic shear, and at high β_N). The high β_p regime has also been extended to steady-state and to the reactor relevant domain.

The high power handling capability of the Mark I divertor target was demonstrated and the severe impurity influxes (carbon "blooms"), which previously terminated high performance plasmas, have been eliminated. The cryopump reduces recycling, eliminates the effects of wall saturation (observed in previous long pulse operation), allows effective particle control, and generally allows higher performance.

JET Strategy

These achievements show that the main objectives of JET are being actively addressed. The overall aim 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.

JET aims to build up 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 using 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 high energy deuterium 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 feedback controlled internal saddle coils;

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

- trying to lengthen 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, ICRF, and 20MW, NB)

c) Increasing the Energy Confinement time τ_E by:

- increasing to 6MA the plasma current in 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 impurities;

- controlling impurities and edge material using the pumped divertor configuration.

In parallel, preparations for the D-T phase of operations have continued. JET has completed installation of the main components of the active gas handling system and pre-tritium commissioning has continued. During 1994, the inactive commissioning phase of subsystems continued in accordance with the programme for D-T operations in 1996.

JET is now continuing its programme to demonstrate effective methods of power exhaust and impurity control in operational conditions close to those envisaged for ITER. ITER relevant studies will provide stimulation to JET and JET's results will make an important contribution to development of the ITER design.

Future Plans

The JET Programme was divided into phases governed by the availability of new equipment and fitting within the accepted life time of the Project. 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) ended in February 1992. The scientific aims of Phase III were to obtain maximum performance in limiter configuration (currents up to 7MA) and to optimize X-Point Operation (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 new phase of the Project (Phase IV: Pumped Divertor Configuration and Next Step Oriented Studies). This phase is subdivided into a Divertor Characterization Plasma and an ITER Support Phase, and extends the lifetime of the Project to the end of 1996.

The present campaign on divertor characterization (to end in May 1995) has still to address: high power combined heating; further exploration of the detached divertor regime and its compatibility with ELMy H-mode operation; helium pumping and transport using argon frost on the cryopump; current profile control; toroidal magnetic field ripple studies; use of the saddle coils with the disruption feedback controller and for generating error fields; and the extension of X-point operation towards 6MA. In March 1995, the present CFC divertor target

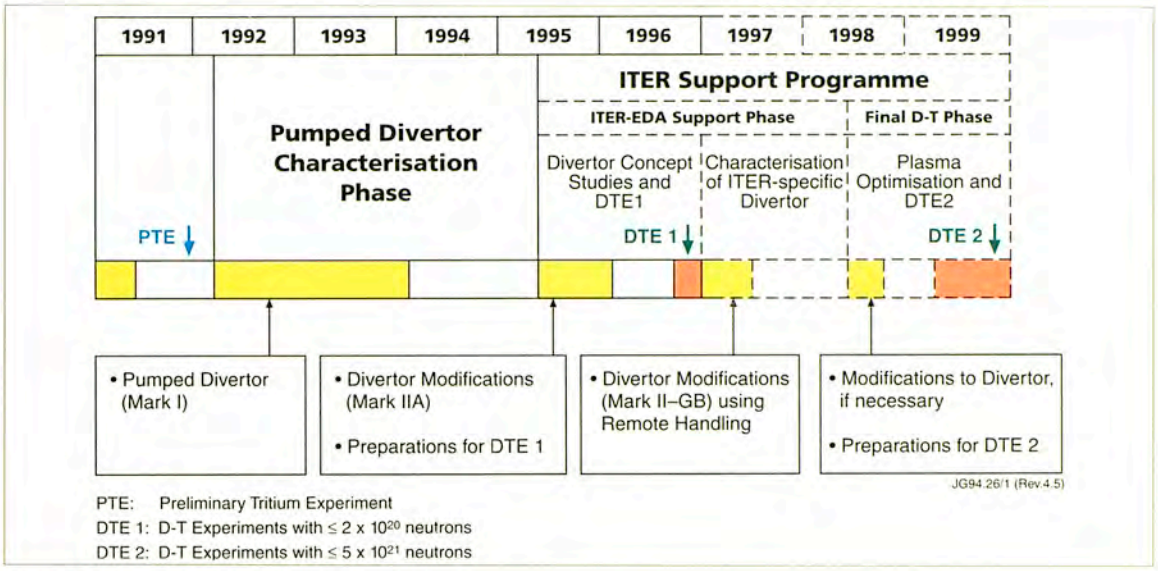


Fig.62: The JET Programme Schedule to 1999

tiles will be exchanged for beryllium target tiles and a comparison will be made for a few specific plasma configurations.

The next major milestone is then to demonstrate compatibility between the high confinement reactor-relevant ELMy H-mode regime and detached divertor operation. This might require the more closed Mark II divertor structure, which will be installed in 1995 and tested in 1996 with particular emphasis on the effect of geometry on gas target/radiative divertor plasmas, which forms the physics basis for the divertor concept favoured by the ITER Joint Central Team.

Extension of the Programme to end of 1999

An extension of the JET Programme to the end of 1999 is currently being proposed. The purpose of the extension would be to provide further data of direct relevance to ITER, especially for the ITER-EDA, before entering into a final phase of D-T operation. In particular, the extension would:

(i) make essential contributions to the development and demonstration of a viable divertor concept for ITER, and

(ii) carry out experiments using deuterium-tritium plasmas in an ITER-like configuration, which will provide a firm basis for the D-T operation of ITER;

while allowing key ITER-relevant technology activites, such as demonstration of remote handling and tritium handling, to be carried out.

Extending JET by three years to the end of 1999 would allow a three year ITER-EDA Support Phase to be inserted into the JET

Programme before the final D-T phase begins. The proposed programme to the end of 1999 is shown in Fig.62.

More information on the future phases are indicated below.

Pumped Divertor Characterisation Phase (1992 to mid-May 1995)

The shutdown begun in February 1992 was completed in January 1994. Experiments during the 1994/5 are focused on:

- establishing reliable operation in the new configuration;
- studying control of impurities, plasma density and exhaust;
- assessing power handling using the full range of ancillary equipment;
- extending performance to high power, long pulse operation;
- studying specific physics and ITER related issues.

The relatively open geometry of the Mark I Divertor accepts a wide range of plasma equilibria, and divertor studies are possible for a range of configurations within the designed maximum plasma current capability of 6MA. Divertor pumping and exhaust issues, particularly helium exhaust, are being addressed with the torus cryopump, which has already shown clear benefits for plasma purity and density control.

Carbon fibre composite divertor target plate tiles were installed for initial operations with the Mark I Divertor and will be used for most of the campaign. During a short intervention in March/April 1995, these will be replaced by beryllium tiles. A comparison will then be made between CFC and beryllium target tiles for a few specific plasma configurations.

ITER-EDA Support Phase (mid-1995 to mid-1998)

The next major shutdown is scheduled to start in May 1995 and to be completed in 9 months. As agreed by the JET Council, the Mark IIA Divertor will be installed. This will have a considerably higher unswept power handling capability than the Mark I divertor, particularly on the vertical side plates. The new divertor configuration will use close-fitting, precisely-aligned large target tiles which result in a much larger plasma "footprint" on the targets. The Mark IIA divertor target will also be more "closed", facilitating the production of a low temperature, high density, high recycling, radiative divertor plasma in which atomic processes (such as radiation and charge-exchanged neutral losses)

reduce the conducted power to the targets. During the shutdown, work will also be undertaken to bring all JET systems and sub-systems to a level of tritium compatibility adequate for the D-T operations planned for 1996. This will include a number of modifications to strengthen or remove components in order to ensure the reliability of the machine.

During 1996, the programme will develop toward long pulse, high performance operation. The Mark IIA divertor target structure can accommodate a wide range of plasma configurations and operating conditions, making possible high power, high current operation on both the horizontal and vertical target plates. The programme will extend many of the issues now being addressed initially with the Mark I divertor and give further emphasis to studies of the effect of geometry on gas target/detached plasmas which form the physics basis for the divertor concept favoured by the ITER Joint Central Team.

A period of D-T operation (DTE-1) is scheduled for the last four months of 1996. It will assess whether the more favourable confinement found during D-T operation in TFTR extends also to the ITER-relevant divertor and operating conditions in JET. These experiments will allow more accurate scalings for the size and heating requirements of ITER to be determined. In addition, it will demonstrate long pulse fusion power production (fusion amplification factor, $Q \approx 1$ with more than 10MW of fusion power). As a result, DTE-1 could last up to four months and produce up to 2×10^{20} neutrons. DTE-1 will also address the important technology issues of reactor relevant tritium processing and remote handling. In particular, DTE-1 will demonstrate the ability of the Active Gas Handling System to process tritium while supporting a reacting tokamak plasma.

In a six-month shutdown in early-1997, the Mark IIA divertor target structure will be exchanged for a second target structure, an ITER-specific divertor of the "Gas-Box" class (Mark IIGB).The exchange will be accomplished by remote handling since radiation levels will be too high during 1997 for a manned in-vessel intervention. This remote handling operation will demonstrate for the first time one of the central technologies required for ITER and for a fusion reactor.

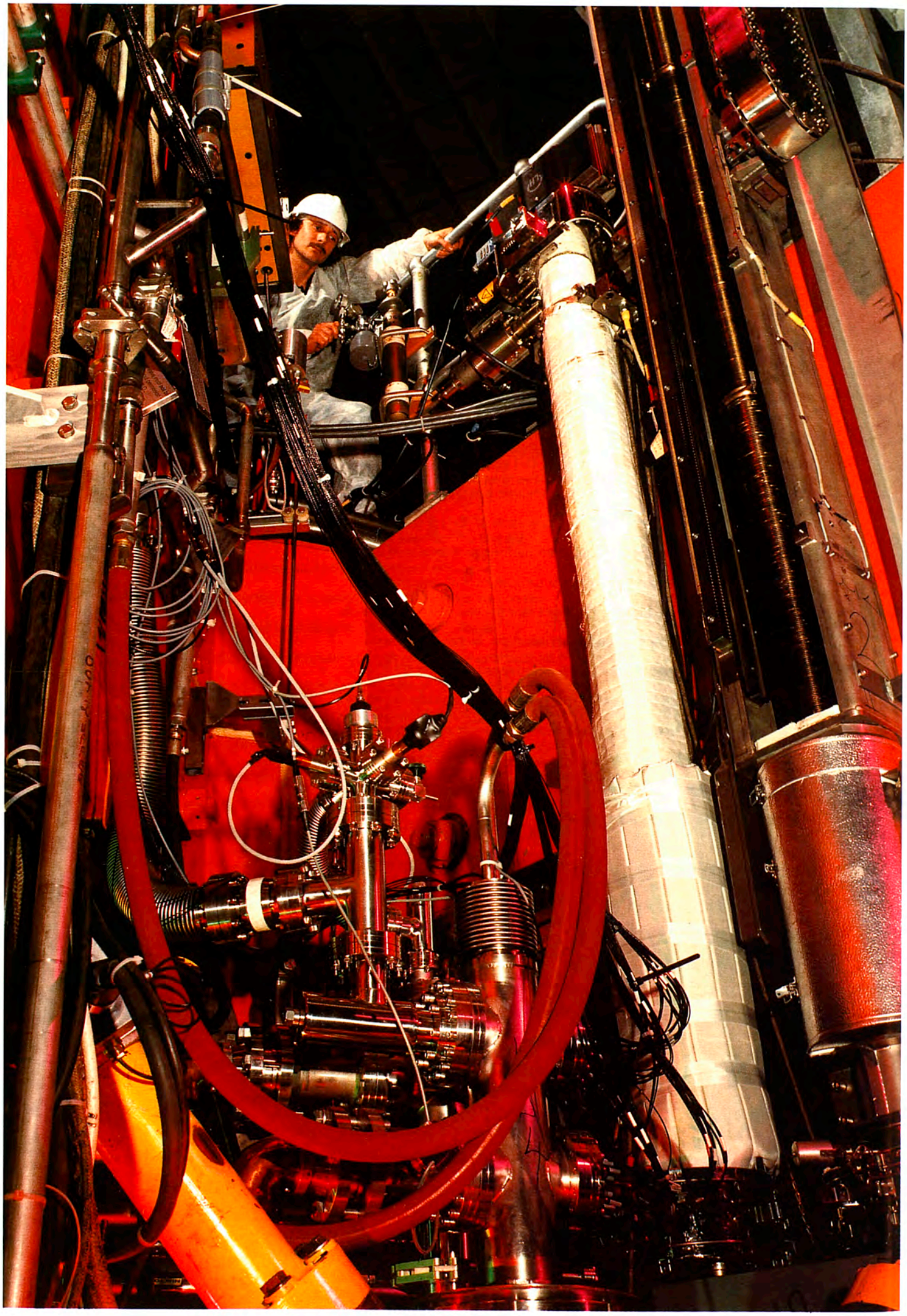
In a divertor of the "Gas-Box" type, such as the Mark IIGB, energy and momentum are removed from the divertor plasma and spread more uniformly over the divertor sidewalls. Neutrals can recirculate freely

through a large relatively open volume below a fairly narrow entrance baffle, which is placed as high as possible near the X-point. The 1997/98 operations period will validate experimentally high power, high performance physics in a closed divertor configuration similar to that currently proposed for ITER.

Final Phase of D-T Operation (mid-1998 to end-1999)

A four-month shutdown in 1998 will permit any necessary modifications to the divertor and final preparations for DTE-2. Manned interventions will again be possible in this shutdown. The subsequent experimental programme in late 1998 and early 1999 will then prepare for the final phase of D-T operation by optimising plasma performance and establishing reliable operation.

A period of high performance D-T operation (DTE-2) is scheduled to take place in 1999. This will capitalise on performance improvements achieved in the preceding experimental campaigns with deuterium. DTE-2 experiments could last up to eight months and could produce up to 5×10^{21} neutrons. Actual neutron production, within this defined limit, will be reassessed in light of the experience gained on JET (DTE-1 in 1996) and on TFTR (in 1994/95). Every effort will be made to reduce the activation produced while still satisfying JET's role in supporting ITER and the World Fusion Programme. This period of D-T operation will also provide a full-scale test of the technology of processing tritium in conjunction with an operating tokamak.



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 de l'École Royale Militaire - Laboratorium voor plasma-physica van de Koninklijke Militaire School') and on behalf of the Université Libre de Bruxelles' ('Service de physique statistique, plasmas et optique non-linéaire de l'ULB'); and of the 'Centre d'Études de l'Energie Nucléaire (CEN)/'Studiecentrum voor Kernenergie' (SCK);

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

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

The 'Ente per le Nuova Tecnologie, l'Energia e l'Ambiente' ('ENEA') representing all Italian activities falling within the Euratom Fusion Programme including that of the 'Consiglio Nazionale delle Ricerche', (CNR);

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 Forschungszentrum Jülich GmbH (KFA), Germany;

The Max-Planck-Gesellschaft zur Förderung der Wissenschaften e.V. Institut für Plasmaphysik (IPP), Germany;

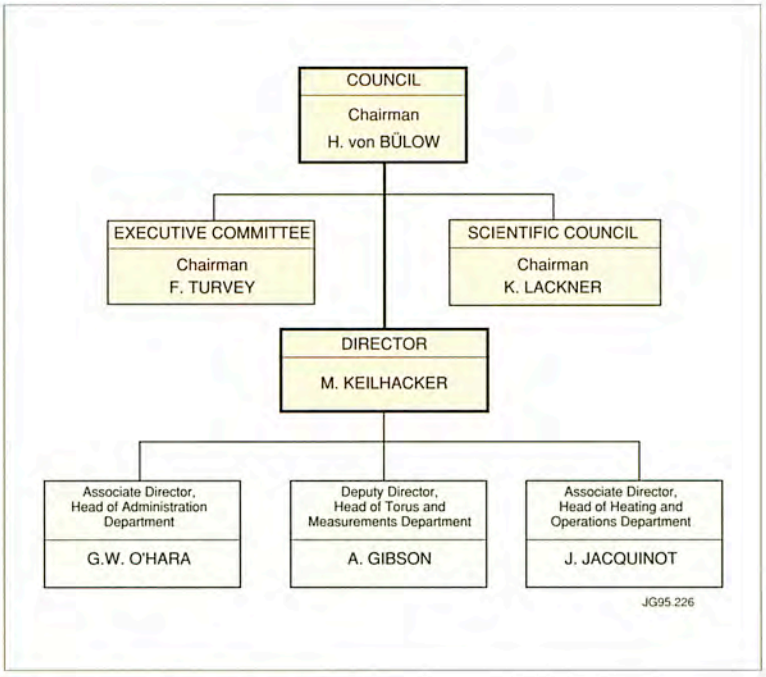


Fig.63: Overall Project Structure

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 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.63).

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 7th February, 24th March, 6th-7th July and 12th-13th October 1994. 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 10th-11th February, 21st April, 14th July, 15th-16th September, 27th-28th October and 8th-9th December 1994.

JET Scientific Council

The 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 Scientific Council met three times during the year: on 8th-9th February, 14th - 15th June and 21st - 22nd September. It also

organised a one day workshop, which was also attended by other experts from the Associated Laboratories, to examine:

- how the JET programme can best satisfy the divertor research needs of ITER.

The JET SC Chairman reported to the JET Council on three occasions, on:

- the JET Director's proposal to extend the JET Programme beyond 1996 (the Mark II divertor programme; options for D-T experiments including the scope and strategic importance of DTE-1; and enhanced plasma performance and coherent advanced tokamak concepts);
- the 1994 Experimental Campaign with the Mark I divertor; and
- the reliability of the JET device for future operations (following the recommendations of a Joint Assessment Group involving JET Scientific Council members and JET staff).

The full Scientific Council membership is detailed in Appendix III.

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. M. Keilhacker, 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 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 three Departments and the Directorate. The three Departments are:

- Torus and Measurements Department;
- Heating and Operations Department;
- Administration Department.

The Project Departmental and Divisional structure is shown in Fig.64.

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.

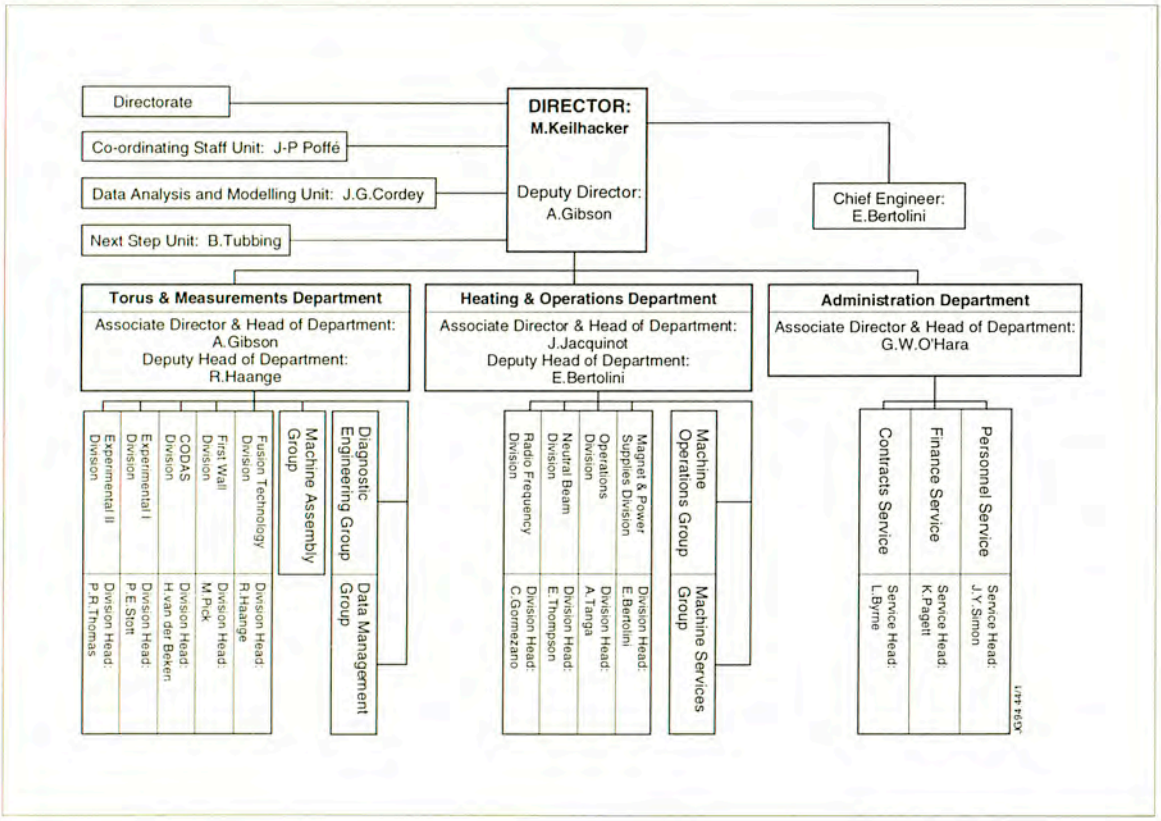


Fig.64: JET Departmental and Divisional Structure

Within the Directorate, are four technical units and a Chief Engineer, reporting directly to the Director. The main responsibilities are as follows:

(a) *The Co-ordinating Staff Unit* is responsible for the availability of a comprehensive health physics and safety project organisation; and for the provision of centralised engineering support services. It comprises four Groups: Health Physics and Safety Group; Quality Group; Technical Services Group; and Drawing Office.

(b) *The Data Analysis and Modelling Unit* is responsible for the provision of software for the acquisition and processing of data from JET diagnostics; for confirming the internal consistency of the processed data and assembling it into public databases; and the development and testing of theoretical models against JET data. In addition, the Unit is responsible for prediction by computer simulation of JET performance, interpretation of JET data and the application of analytic plasma theory to gain an understanding of JET physics. It comprises three groups: Analytic Theory Group; Simulation Group; and Data Processing and Analysis Group

(c) *The Next Step Unit* is responsible for co-ordinating contributions from JET to the European effort in support of the ITER-EDA. This responsibility includes drawing up proposals, initiating relevant work programmes on JET and taking part in their execution and evaluation.

(d) *The Divertor Interface Unit* is responsible for assessing the impact of developments in the experimental programme and operation on the design requirements for JET divertors. This includes a high level of participation in the experimental programme on divertor physics, thermomechanical analysis of plasma induced loads on the divertor, and the definition of advanced divertor concepts.

In addition, there is a *Chief Engineer*, who reports to the Director, and is responsible for ensuring the overall coherence of technical aspects of JET operations.

Torus and Measurements Department

The Torus and Measurements Department has overall responsibility for the performance capacity of the machine: this ncludes enhancements directly related to this (excluding heating) and the long term planning

associated with integration of these elements to achieve ultimate performance. The Department is also responsible: for fusion technology requirements for the active phase including tritium handling and processing; for construction and operation of necessary measurement diagnostic systems and the interpretation of experiment data; and for data systems comprising data control, acquisition and management. The main functions of the Department are:

- to design, procure and implement enhancements to the JET device;
- to provide and maintain clean conditions inside the vessel which lead to high quality plasma discharges;
- to conceive and define a set of coherent measurements;
- to be responsible for construction of necessary diagnostics;
- to be responsible for diagnostics operation, quality of measurements and definition of plasma parameters;
- to organise and implement data acquisition and computing;
- to design and develop remote handling methods and tools to cope with JET requirements;
- to design and construct facilities for handling tritium and for waste management.

The Department consists of five Divisions and three Groups (Machine Assembly, Diagnostic Engineering and Data Management):

(a) *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 the first wall systems and its components such as limiters, wall protections and internal pumping devices. The area of responsibility encompasses the mechanical integrity of the vacuum vessel as a whole and the development and implementation of mechanical and remote handling techniques;

(b) *Fusion Technology Division*, is responsible for all nuclear engineering aspects of the Project including tritium and gas handling, vacuum systems, waste management and regulatory approvals;

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

(d) *Experimental Division 1 (ED1)*, which is responsible for specification, procurement and operation of about half the JET diagnostic systems. ED1 undertakes electrical measurements, electron temperature measurements, surface and limiter physics and neutron diagnostics;

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

Heating and Operations Department

The overall responsibility of the Heating and Operations Department is for the efficient and effective day-to-day operation of the machine. In addition, the Department has responsibility for plasma heating and auxiliary equipment and related physics; the design and operation of power supplies as well as contributing to the execution and evaluation of JET's experimental programme. The main functions of the Department are:

- preparing and co-ordinating operation of the machine across Departments and Divisions;
- heating and current drive and analysis of its effects in the plasma;
- plasma fuelling, including pellet injection;
- designing and employing power supplies for ensuring efficient operation and control of the machine.

The Department consist of two Groups (Machine Operations and Machine Services) and four Divisions:

(a) *Operations Division* plays a major role in the efficient planning and execution of JET's experimental programme and in the integration of existing or imminent systems into an effective experimental programme. In addition, it is responsible for effective methods of fuelling the plasma including the development of methods based on solid high speed hydrogen pellets; development of new plasma wall conditioning techniques; plasma control systems; development of disruption control methods; training of operations staff; and monitoring of machine operations;

(b) *Neutral Beam Heating Division*, which is responsible for construction, installation, commissioning and operation of the neutral injection system, including development towards full power operation. The Division is also responsible for all cryo-systems and also participates in studies of physics of neutral beam heating;

(c) *Radio Frequency Heating Division*, which is responsible for the design, construction, commissioning and operating RF heating and current drive systems during the different stages of its development to full power. The Division is also responsible for the TAE excitation system and also participates in studies of the physics of RF heating;

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

Administration Department

The Administration Department is responsible for providing Contracts, Finance and Personnel services to the Project. In addition, the Department is responsible for the administration of Publications and Public Relations Groups.

Administration

Introduction

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

Finance

The initial budgets for 1994 were approved at 81.70 MioECU for Commitments and 100.16 MioECU for both Income and Payments, subject to a 10.0 MioECU blockage in all Budgets only to be released on the agreement of the JET Council. As a result of the efforts made to achieve longer term reductions in annual costs, the JET Director did not need to request release of the 10.0 MioECU. Therefore, the final 1994 budgets were 71.70 MioECU for Commitments and 90.16 MioECU for both Income and Payments.

Due to the temporary blockage by the European Parliament of part of the Community Fusion Programme Budget for 1994, the original budget could not be adopted until August. In the meantime, the Project operated within a series of constrained budgets and rigorous expenditure restrictions were enforced.

The Commitments and Payments Budgets each are divided into two phases of the Project - Extension to Full Performance and the Operational Phase; subdivisions distinguish between investment, operating, and personnel costs, each with further detailed cost codes.

Commitments

Of the total final appropriations in 1994 of 89.48 MioECU (including 17.78 MioECU brought forward from previous years), 79.36 MioECU

COMMITMENT APPROPRIATIONS	MioECU
INITIAL COMMITMENTS BUDGET FOR 1994	71.70
AMOUNTS BROUGHT FORWARD FROM PREVIOUS YEARS.	17.78
	89.48
COMMITMENTS MADE DURING THE YEAR	79.36
BALANCE OF APPROPRIATIONS AT 31 DECEMBER 1994 AVAILABLE FOR USE IN 1995	**10.12**

Table 7: Commitment Appropriations for 1994

was committed and the balance of 10.12 MioECU was available for carrying forward to 1995. The details of the commitment appropriations available (Table 7) and of the amounts committed in each Phase during the year (Table 8) are summarised as follows:

- In the extension to Full Performance Phase, 0.62 MioECU was committed leaving 1.13 MioECU commitment appropriations not utilised at 31 December 1994, to be carried forward to 1995;
- In the Operational Phase, 78.74 MioECU was committed leaving a balance of 8.99 MioECU to be carried forward to 1995.

		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	1.75	0.62	3.00	2.73
PHASE 3	**OPERATIONAL**				
TITLE 1	PROJECT INVESTMENTS	5.55	4.81	8.42	7.14
TITLE 2	OPERATING COSTS	33.92	28.82	35.78	34.17
TITLE 3	PERSONNEL COSTS	48.26	45.11	48.31	46.19
TOTAL PHASE 3		87.73	78.74	92.51	87.50
PROJECT TOTAL - ALL PHASES		**89.48**	**79.36**	**95.51**	**90.23**

Table 8: Commitments and Payments for 1994

INCOME AND PAYMENTS	MioECU	
INCOME		
BUDGET FOR 1994		90.16
INCOME RECEIVED DURING 1994		
(I) MEMBERS' CONTRIBUTIONS	81.67	
(II) BANK INTEREST	0.95	
(III) MISCELLANEOUS	0.02	
(IV) UNUSED APPROPRIATIONS BROUGHT FORWARD FROM PREVIOUS YEARS	7.52	
TOTAL INCOME		90.16
VARIATION FROM BUDGET		-
PAYMENTS		
BUDGET FOR 1994	90.16	
AMOUNTS AVAILABLE IN THE SPECIAL ACCOUNT TO MEET OUTSTANDING COMMITMENTS AT 31 DECEMBER 1993.	5.35	
TOTAL AVAILABLE APPROPRIATIONS FOR 1994		95.51
ACTUAL PAYMENTS DURING 1994	90.23	
FROM SPECIAL ACCOUNT TRANSFERRED TO INCOME.	0.14	
		90.37
UNUTILISED APPROPRIATIONS AT 31 DECEMBER 1993 CARRIED FORWARD IN THE SPECIAL ACCOUNT TO MEET OUTSTANDING COMMITMENTS AT THAT DATE.		5.14

Table 9: Income and Payments for 1994

Income and Payments

The actual income for 1994 was 82.64 MioECU to which was added 7.52 MioECU available appropriations brought forward from previous years, giving a total of 90.16 MioECU. Total payment appropriations for 1994 were 95.51 MioECU; payments in the year amounted to 90.23 MioECU and 0.14 MioECU was transferred from the Special Account to income. The balance of 5.14 MioECU was transferred to the Special Reserve Account to meet commitments outstanding at 31 December 1994. (Payments are summarised in Tables 8 and 9).

Contributions from Members

The budget for Members' contributions was 81.67 MioECU as follows:

- 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 10 gives contributions from Members for 1994.

MEMBER	%	Mio ECU
EURATOM	80.0000	65.34
BELGIUM	0.2745	0.22
CIEMAT, SPAIN	0.2809	0.23
CEA, FRANCE	1.8143	1.48
ENEA, ITALY	1.9316	1.58
RISO, DENMARK	0.0756	0.06
LUXEMBOURG	0.0018	0.00
JNICT	0.0752	0.06
KFA, GERMANY	0.7947	0.65
IPP, GERMANY	2.4602	2.01
KFK, GERMANY	0.7316	0.60
NFR, SWEDEN	0.2400	0.20
SWITZERLAND	0.5171	0.42
FOM, NETHERLANDS	0.3403	0.28
UKAEA	10.4622	8.54
	100.0000	**81.67**

Table 10: Percentage Contributions to JET for 1994

Bank Interest

During the year, funds are normally received on a quarterly basis in respect of Members' contributions and intermittently for other items. Therefore, the Project has funds not immediately required for discharge of its commitments; these funds are placed on deposit accounts at market interest rates. During 1994, earned interest was 0.95MioECU.

Appropriations from Earlier Years

Unused payment appropriations and excess income over budget of 1.632 MioECU arising in 1992 and 5.883 MioECU arising in 1993 were transferred to income in 1994.

Summary

Table 11 summarises the financial transactions of the JET Joint Undertaking as at 31 December 1994, which have yet to be audited. The final audited accounts will be published in due course.

FINANCIAL TRANSACTIONS	MioECU
CUMULATIVE COMMITMENTS	1,485.5
CUMULATIVE PAYMENTS	1,462.3
UNPAID COMMITMENTS	23.2
AMOUNT CARRIED FORWARD IN THE SPECIAL ACCOUNT	5.1
AMOUNT AVAILABLE FROM 1993 AND 1994 TO SET OFF AGAINST FUTURE CONTRIBUTIONS FROM MEMBERS	1.5

Table 11: Summary of Financial Transactions at 31 December 1994

Contracts Service

Contracts Activity

During 1994, tender/contract work was undertaken by the Contracts Service as shown in Table 12. Many larger contracts involved advance and retention payments for which bank guarantees were required by JET. The value of guarantees held at 31 December 1994 was 4.5MioECU.

Imports and Exports Services

Contracts Service is also responsible for the import and export of JET goods. 712 imports and 292 exports were handled in 1994. There were also 1274 issues of goods to UK firms. The total value of issues to all countries for the year was 8.027MioECU.

Stores Organisation

The bulk of material is procured on a "just in time" basis and stores organisation provides a receipts and delivery service for this material. The total number of such receipts in 1994 was 17,700.

Administration of Contracts

The distribution of contracts between countries is shown in Tables 13 and 14. Table 13 includes all contracts with a value of 10,000ECU and above placed prior to 1984, together with all contracts placed during the period 1984-94. Table 14 is an allocation of "high-tech" contracts, which is based on figures shown in Table 13 but excludes all contracts below 5,000ECU and contracts covering civil works,

FORMAL TENDER ACTIONS	NUMBER	
SUPPLY	55	
SERVICE	23	
PERSONNEL	29	
TOTAL	**107**	
CONTRACTS PLACED	**QUANTITY**	**VALUE (MioECU)**
MAJOR (>75 kECU)	50	7.851
MINOR (<75kECU)	4,402	16.058
DIRECT ORDERS	11,852	2.516
AMENDMENTS AND WARRANTS	658	23.010
TOTAL	**16,962**	**49.435**

Table12: Formal Tender Actions and Contracts placed during 1994

COUNTRY	VALUE (kECU)	% OF TOTAL
UK	564,354	56.07
GERMANY	163,473	16.24
FRANCE	88,530	8.80
ITALY	58,914	5.85
SWITZERLAND	43,068	4.28
DENMARK	13,340	1.33
NETHERLANDS	17,465	1.74
BELGIUM	11,662	1.16
SWEDEN	6,848	0.68
IRELAND	975	0.10
OTHERS	37,716	3.75
TOTALS	**1,006,345**	**100.00**

Table 13: Allocation of JET Contracts

installation, pipework, consumables (including gases), maintenance operations and office equipment (including PCs).

Personnel Service

Staffing

The JET Team strength has reduced again this year (Table 15). Departures of Euratom staff continued at a high rate (over 10%) and replacement with Euratom recruits was restricted. However, UKAEA staff numbers have increased as a result of active recruitment to fill vacancies in 1994. The aim is to have some growth in team staff

COUNTRY	VALUE (kECU)	% OF TOTAL
UK	138,916	27.34
GERMANY	142,951	28.14
FRANCE	77,775	15.31
ITALY	51,242	10.09
SWITZERLAND	34,795	6.85
DENMARK	7,448	1.47
NETHERLANDS	16,277	3.20
BELGIUM	5,062	1.00
SWEDEN	4,482	0.88
IRELAND	389	0.08
OTHERS	28,649	5.64
TOTALS	**507,986**	**100.00**

Table 14: Allocation of JET "High-Tech" Contracts

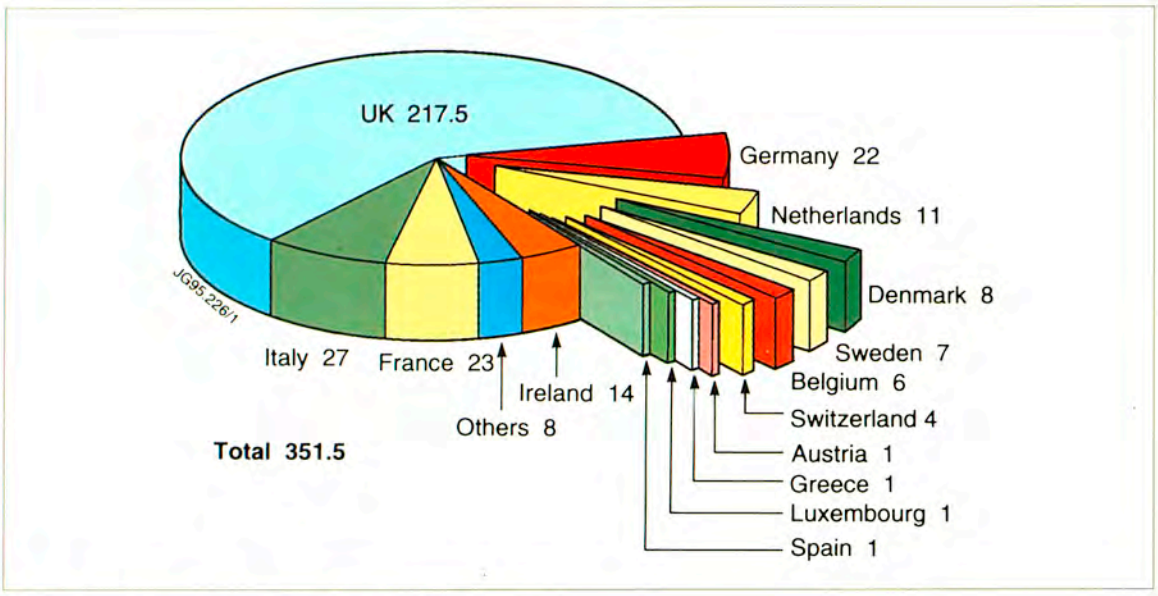


Fig.65: Composition of the JET team by nationality

numbers in 1995 and 1996 to meet work programme needs. Figure 65 shows the composition of the JET team by nationality. In addition to the JET Team staff, 222 contract staff were on site at 31 December 1994, charged to Title 3 of the JET Budget.

All team staff assignments expired at the end of 1994, and once the budgetary position for 1995/6 was clarified, all existing team staff were reassigned until 31 December 1996.

Thirty-four staff (including one part-time member) left the Project during 1994 (16 Euratom, 1 DGXII and 17 UKAEA) the same total as in 1993. Among UKAEA leavers were five retirements but most leavers from the Euratom category obtained posts in the services of the Commission. Two leavers (both UKAEA) moved to ITER.

Recruitment and Promotions

Recruitment was restricted to UKAEA candidates during 1994. There were six recruits: four were already employed by UKAEA and two were newly recruited on fixed term contracts.

TEAM POSTS	POSTS FILLED END 1993	POSTS FILLED END 1994
UKAEA	228.5	218.5
EURATOM	143	127
DGXII	7	6
TOTAL JET TEAM	**378.5**	**351.5**

Table 15: Team posts filled

Three existing team staff were selected to fill Group Leader vacancies during the year. Following the annual promotion exercise 32 staff were promoted to higher grades. Changes in the Commission's promotion procedures necessitated a review of the promotion procedures for Euratom staff at JET. Revised arrangements for assessing eligible staff (Euratom and UKAEA) were introduced.

Staffing Policy

JET staffing policy was reviewed by the JET Executive Committee in the light of the need to maintain a team of trained, experienced and dedicated personnel to operate JET if a proposed extension to 1999 was approved. The retention of staff remains a problem due to uncertainty about the immediate and longer-term future of the Project, and concerns about post-JET careers. The Project is exploring possible measures with the respective employers to address this problem.

The employers' conditions of service were administered, as appropriate. The UKAEA Retention of Experience Allowance was paid to the majority of UKAEA Team staff in December 1994.

Staff Relations

At the beginning of May 1994 the European Parliament formally voted to make available from the Community Budget a sum of 2MioECU for disbursement to UKAEA staff assigned to JET. These disbursements were to be part of a package of measures to help resolve the UKAEA staff petition to the European Parliament. The JET Council agreed that JET could make the disbursement, acting as an agent outside its normal budget

Consultants

Twelve consultants were appointed by JET in 1994 to work for a total of 132 man-days on scientific/technical matters. This compares with 204 man-days in 1993.

Assigned Associated Staff

1994 saw an increase of 58% in man-years of effort provided at JET from the Associations, other than the UKAEA, under the Assigned Associate Staff Agreement. The total contribution for 1994 was 29.1 man-years compared with 23.8 man-years in 1993; the UKAEA

LABORATORY		MAN-YEARS
UKAEA	(UK)	12.0
IPP	(GERMANY)	4.6
CEA	(FRANCE)	2.9
NFR	(SWEDEN)	2.8
ENEA	(ITALY)	2.7
CRPP	(SWITZERLAND)	2.1
JNICT	(PORTUGAL)	1.1
CIEMAT	(SPAIN)	0.9
TOTAL		**29.1**

Table 16: Staff Assignments from Associated Laboratories during 1994

contribution within this was 12 man-years in 1994. There appeared to be an increased interest in assignments when JET operations resumed at the end of February 1994. Tables 16 and 17 show the contribution made by the Associations in 1994 and the distribution of the Assigned Associated Staff within the Project.

Visiting Scientists

Appointments to the Visiting Scientist scheme are made through the UKAEA as Temporary Research Associates. During 1994, three new appointments were made, one each from USA, Japan and Russia. There were seven Visiting Scientists on site at the year end.

Fellows

During 1994, eight new JET Fellowships (under the Commission's Fellowship Scheme) were awarded for two years, three to engineers and five to scientists. At the end of 1994, there were 26 Fellows on site (14 engineers and 12 scientists). Of the research projects undertaken, five were at post-doctoral level and 21 at a lower post-graduate level.

DEPARTMENT	MAN-YEARS
TORUS AND MEASUREMENTS DEPT	17.5
DATA ANALYSIS AND MODELLING UNIT	4.7
HEATING AND OPERATIONS DEPT	5.9
DIRECTORATE	1.0
TOTAL	**29.1**

Table 17: Assigned Staff within the Project during 1994

Students

The number of student contracts awarded reduced slightly during 1994: 53 students worked at JET during 1994, contributing 881 man-weeks. 70% of the appointments were placements of 13 weeks or more. No applications were received from students in Luxembourg, Portugal, Switzerland and Sweden during 1994, but students were recruited from all other Member states.

Training

During 1994, individual staff needs for skills training were met where possible, through attendance on external short courses (about 230 man-days). Safety courses are described in the section on Safety and Health Physics. Nine Team staff were supported by JET to obtain vocational or academic qualifications, in addition to the 53 undergraduate students gaining work experience at JET and the 26 Fellows and 7 other PhD students.

There were 60 JET-based Science Meetings and Symposia. In addition, training opportunities in languages and European Union institutions and policy were provided for Team staff, following the recommendations of a joint JET staff-management working party. Tuition was offered in three Community languages, French, German and Italian. 100 Team staff were allocated places for tuition between October and December 1994.

Administration

Cost reductions have been achieved, in particular in the areas of JET's telecommunications and missions. The number of missions undertaken by JET staff in 1994 fell by 39%.

JET hosted 16 meetings of its governing and advisory bodies, for which transport and accommodation was organised.

Health Physics And Safety

The 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 and EC legislation and, in accordance with the Host Support Agreement, JET complies with the safety regulations of the Host Organisation. Responsibilities for Safety and Health Physics

matters are discharged by the Health Physics and Safety Group within the Coordinating Staff Unit.

Safety

The Group provides a general safety service that incorporates safety related training, monitoring, co-ordination and planning of statutory inspections. It ensures there is an awareness of any new legal requirements or changes to existing legislation. During 1994, new requirements for Pressure Systems resulted in the need for a written scheme of examination to be in place. This was in place to meet the legislative implementation date of July 1994.

During the year, safety related training continued but as the year was predominantly associated with machine operation the numbers show some differences to those of 1993, which was a long shutdown year. For comparison, figures for 1993 are set out in brackets. Throughout 1994, 320 (470) people attended Safety Induction and 98 (137) attended the Basic Safety courses. Radiological Protection courses were attended by 67 (125) people, while 46 (125) attended the Beryllium Introduction courses. Cardio-pulmonary resuscitation courses continue to be well attended with 220 (250) attending. In total, there were 1474 (2299) attendances at safety related courses for 28 course topics during 1994. Throughout 1994, there were 78 (135) accidents reported to the Safety Section. All were classed as minor except for 2 (6) which were reported to the UK Health and Safety Executive since they resulted in more than three days absence from work. Neither accident was work related but occurred within JET buildings. The total lost time from all accidents in 1994 was 45 (163) days. There was only one Incident Safety Review Panel that needed to be set up in 1994 and the Panel report was published during the year.

Health Physics

The Group provides a comprehensive radiological protection and occupational hygiene service, dosimetry service, beryllium analysis and environmental monitoring, both on and off-site.

Following the resumption of operations in January, after the divertor shutdown, there were three short interventions into the torus vessel during March, September and November 1994. The highest radiation dose-rate measured in the torus during the year

was 78μSvh^{-1}. The collective dose accrued as a result of torus intervention was 0.021 man-Sv, which was approximately 50% of that recorded in 1993. The collective dose during the calendar year 1994 for the Project was 0.026man-Sv and reflects the focus of work during long operational periods. The maximum individual dose arising from in-vessel work was 1.37mSv, which is similar to the 1.43mSv individual maximum recorded in 1993. The radiation doses incurred by individuals during 1994 remained low and within the JET dose limitation policy and well within UK statutory limits. On- and off-site radiological monitoring of grass, soil, water and crop samples continued to indicate a negligible impact on the local environment as a result of JET operations.

The Ionising Radiation (Outside Workers) Regulations came into force on 1st January 1994 and these Regulations are aimed at ensuring the radiological safety of those staff who work at a number of locations. There are requirements for the provision and exchange of information between the employer and the outside worker. This is achieved by the issue of Radiation Passbooks and the Group have set the arrangements to ensure compliance with the regulations.

Beryllium is commonly in use at JET but its hazardous properties are well known and controlled through the application of the JET Code of Practice for its safe use, which includes the designation of Beryllium Controlled Areas and Beryllium Workers. Work in these controlled areas continued throughout 1994 with the exposure to airborne beryllium quantified by the means of personal air sampling. Samples are processed and analysed in the Health Physics Beryllium Laboratory and 4715 (14,281) personal air samples were analysed in 1994. All exposures were below the maximum exposure limit of 2μgm^{-3} for beryllium as specified in the UK's Control of Substances Hazardous to Health Regulations.

Safety-related Committees

There are currently two committees on safety-related matters:

The Fusion Safety Committee: this Committee chaired by the Head of CSU is attended by representatives of relevant operational Divisions/Groups at JET and has several independent members, including one from another site handling tritium. Its function is to keep under review the safety aspects of the project during design,

commissioning and operation, which arise from the use of tritium. Its members provide advice on safety issues arising from safety documents prepared by JET and related peer reviews, and in particular it considers proposals for modifications which could significantly affect safety. It also receives reports of significant incidents.

The JET Safety Working Group: the Group is concerned with day-to-day aspects of safety and in particular it monitors and makes recommendations on the JET Safety at Work System. It follows up safety inspections and actions from all incident investigations. Meetings are chaired by the Head of the Coordinating Staff Unit and there is representation from all operational and advisory groups at JET involved in safety, plus an independent UKAEA member and staff representatives.

Press and Public Relations

Whilst the year was a relatively quiet one from a public relations viewpoint, there was a continuing interest from the media and the general public in both nuclear fusion matters in general and JET progress in particular. The restart of JET operations in the new divertor configuration attracted interest mainly from the specialist media.

During 1994, five television crews visited JET to record interviews with senior staff and to film background shots of JET and its facilities. Among these was a BBC crew accompanied by Lord Marshall of Goring, formerly Chairman of the UKAEA. He recalled, the protracted political discussions during 1976/77 on the siting of JET when he was Chief Scientific Advisor to the UK Government.

Four radio reporters visited the Project during the year to record interviews with senior staff. In addition, five interviews were given for local radio stations. In the written media, eight science correspondents came to JET to prepare articles on the progress and developments of the Project. The main technical aspects in which the media took an interest, revolved around the 'new JET' which embodies the pumped divertor and on the results of the tritium experiments on the TFTR fusion experiment at Princeton Plasma Physics Laboratory, USA. Other topics which provoked press interest included the blocking of the Euratom Fusion Budget by the European Parliament and the outcome of the UKAEA staff's petition to the European Parliament.

Among the more distinguished visitors were Dr R Dautray, High Commissioner of the CEA, France; Sir William Stewart, Chief Scientific Advisor to the UK Government; the Russian theoretician, Academician B Kadomtsev; the former Director General of CERN, Professor C Rubbia, and UK energy expert Professor Ian Fells. Politicians visiting JET during the year included Mr Tim Eggar, the UK Energy Minister.

As in previous years many groups from Universities, Colleges and Schools from several European countries made visits to JET - all of whom were given introductory talks and a tour of JET laboratories. In addition, groups from professional bodies and local organisations visited the Project. In all, about 100 groups were given tours of the laboratories. Also JET hosted visits from prize winners of national science competitions in Germany and Portugal.

In November, during the European Week for Scientific Culture organised by the European Commission, younger members of JET staff gave illustrated talks on the European Fusion Research Programme to local schools. This resulted in requests for follow up visits to JET.

Publications Group

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

Conferences

JET provided contributions to a number of major meetings, as follows:

- 11th International Conference on Plasma Surface Interactions, Mito-shi, Japan, May 1994, (2 Invited Papers and 16 Posters);
- 21th European Conference on Controlled Fusion and Plasma Physics, Montpellier, France, June 1994 (3 Invited Papers and 41 Posters);
- 18th Symposium on Fusion Technology (SOFT-18), Karlshrue, Germany, August 1994, (2 Invited Papers and 33 Posters);
- 15th IAEA Conference on Plasma Physics and Controlled Nuclear Fusion Research, Seville, Spain, September 1994, (9 Oral Contributions and 2 Posters);

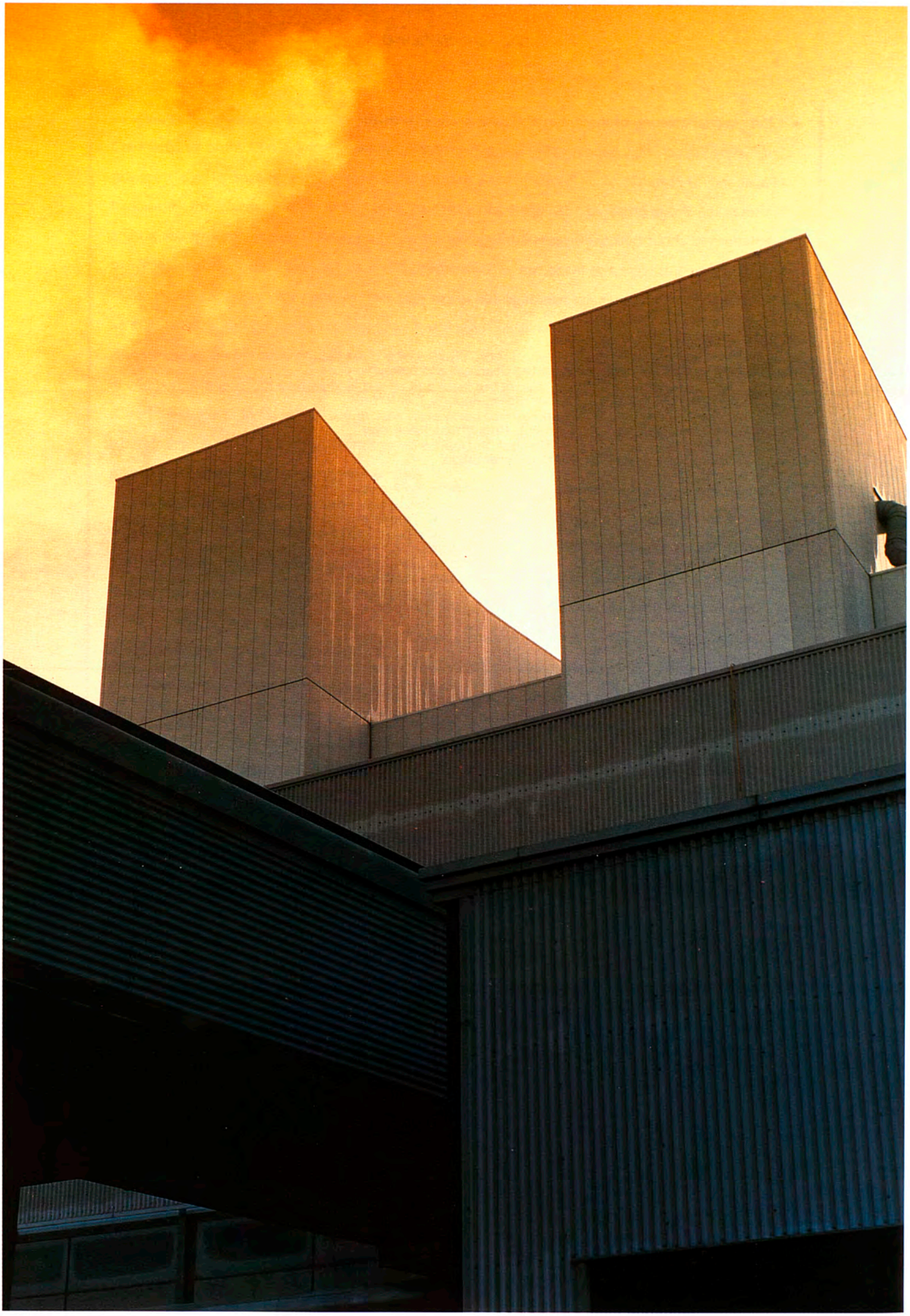
- 36th Annual Meeting of the American Physical Society - Division of Plasma Physics, Minneapolis, USA, November 1994, (1 Invited Paper and 10 Posters).

In total, the Group prepared 168 Papers and 142 Posters for presentations to about twenty different Conferences throughout the world. Arrangements were also made by the Group for 157 participants to attend these major meetings during the year.

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 1994, over 360 publications were cleared for external presentation.

During the year, 218 documents were published from the Project and the full list is included as an Appendix to the 1994 JET Progress Report. This total included 12 JET Reports, 74 JET Preprints, 7 JET Internal Reports, 2 JET Technical Notes and 13 JET Divisional Notes. All these documents are produced and disseminated by the Group on a wide international distribution. In addition, 81 papers were published in scientific Journals. In total, the Group produced 3097 new illustrations and figures and took 1998 new photographs for publications and other disseminated material during 1994.



APPENDIX I
The JET Council

Member	Representative
The European Atomic Energy Community (EURATOM)	P. Fasella C. Maisonnier
The Belgian State, acting for its own part ('Laboratoire de Physique des Plasmas de l'École Royale Militaire - Laboratorium voor plasma-physica van de Koninklijke Militaire School') and on behalf of the Université Libre de Bruxelles' ('Service de physique statistique, plasmas et optique non-linéaire de l'ULB'); and of the 'Centre d'Études de l'Energie Nucléaire (CEN)/'Studiecentrum voor Kernenergie' (SCK)	P.E.M. Vandenplas T. van Rentergem (to April) G. Michaux (from April)
The Centro de Investigaciones Energéticas Medioambientales y Tecnológicas (CIEMAT), Spain	A. Grau Malonda
Commissariat à L'Énergie Atomique (CEA), France	D. Escande R. Aymar (to October) J-P. Schwartz (from October)
The 'Ente per le Nuova Tecnologie, l'Energia e l'Ambiente' ('ENEA') representing all Italian activities falling within the Euratom Fusion Programme including that of the 'Consiglio Nazionale delle Ricerche', (CNR); 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 (Chairman) J. Kjems
The Grand Duchy of Luxembourg (Luxembourg)	Mrs. S. Lucas J. Hoffmann (to October) R. Becker (from October)
The Junta Nacional de Investigação Científica e Tecnológica (JNICT), Portugal	C. Varandas Mrs. M.E. Manso
Ireland	F. Turvey M. Benville (to September) P. O'Neill (from September)
The Forschungszentrum Jülich GmbH (KFA), Federal Republic of Germany	G. von Klitzing
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	G. Leman H. Wilhelmsson
The Swiss Confederation	F. Troyon (Vice-Chairman) 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

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 de l'École Royale Militaire - Laboratorium voor plasma-physica van de Koninklijke Militaire School') and on behalf of the Université Libre de Bruxelles' ('Service de physique statistique, plasmas et optique non-linéaire de l'ULB'); and of the 'Centre d'Études de l'Energie Nucléaire (CEN)/'Studiecentrum voor Kernenergie' (SCK)	R. Vanhaelewyn
The Centro de Investigaciones Energéticas Medioambientales y Tecnológicas (CIEMAT), Spain	F. Manero
Commissariat à L'Énergie Atomique (CEA), France	R. Gravier C. Samour (to January) Mrs. P. Livanos (from January)
The 'Ente per le Nuova Tecnologie, l'Energia e l'Ambiente' ('ENEA') representing all Italian activities falling within the Euratom Fusion Programme including that of the 'Consiglio Nazionale delle Ricerche', (CNR); dell'Energia Nucleare e delle Energie Alternative (ENEA), Italy	A. Coletti M. Samuelli (to May) F. Pecorella (from May)
The Hellenic Republic (Greece)	N. Chrysochoides
The Forskningscenter Risø (Risø), Denmark	Mrs. L. Grønberg V.O. Jensen
The Grand Duchy of Luxembourg (Luxembourg)	R. Becker (to October) C. Bartocci (from October)
The Junta Nacional de Investigação Cientîfica e Tecnológica (JNICT), Portugal	J. Bonfim (to April) J. da Costa Cabral (from April) F. Serra
Ireland	F. Turvey (Chairman) D. Taylor
The Forschungszentrum Jülich GmbH (KFA), Federal Republic of Germany	V. Hertling
The Max-Planck-Gesellschaft zur Förderung der Wissenschaften e.V. Institut für Plasmaphysik (IPP), Federal Republic of Germany	Mrs. I. Kramer
The Swedish Natural Science Research Council (NFR), Sweden	G. Leman (Vice-Chairman) L. Gidefeldt
The Swiss Confederation	M. Tran L. de Faveri (to February) S. Berthet (from February)
The Stichting voor Fundamenteel Onderzoek der Materie (FOM), The Netherlands	A. Verhoeven
The United Kingdom Atomic Energy Authority (UKAEA)	D.C. Robinson T.J. Elsworth (to April) T. Conlon (from April)

Secretary: J. McMahon, JET Joint Undertaking

APPENDIX III
The JET Scientific Council

Members appointed by the JET Council

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Staff Secretary: M.L. Watkins, JET Joint Undertaking