EFDA

Joint European Torus

Report of Activities
1st January 2000 - 31 March 2001

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Preface

With the planned winding up of the JET Joint Undertaking at the end of 1999, two committees chaired respectively by C. Mancini and H. Bruhns were established to determine how the continued use could best be made of the JET Facilities. Following the recommendations of the Mancini Committee, the European Commission and the fusion Associates agreed to continue the exploitation of JET under the European Fusion Development Agreement (EFDA). The initial Workplan, including the Task Force structure under which the scientific exploitation of JET would be organised, was prepared by the Bruhns Committee with input from the EFDA Associates.

The management structure of the new EFDA-JET organisation was defined in the JET Implementing Agreement (JIA). It specifies that the EFDA Associate Leader for JET (J. Paméla) has overall responsibility for the implementation of this Agreement and is responsible to the EFDA-JET Subcommittee for the execution of the JET Workprogramme. Under the framework of a separate JET Operation Contract (JOC), the UKAEA (the Operator) is responsible for the operation and safety of the JET Facilities. A UKAEA Senior Manager (F. Briscoe) is responsible for the execution of this Contract.

Despite the large upheaval caused by the change of organisation and staff, the JET Facilities were restarted by the Operator in time for the first experiments in May 2000. Since then four experimental campaigns (C1-C4) have been carried out, involving the participation of three hundred scientists from the Associates. The success of these campaigns is a testimony to the skill and hard work of a large team of people and is crucial to the continued success of the European fusion programme as a whole.

The EDFA JET Subcommittee Chairman
M. Kaufmann

Introduction by the EDFA Associate Leader for JET

Since 1 January 2000 the JET Facilities have been used under the new European Fusion Development Agreement (EFDA). The most notable features of this Agreement with respect to the Joint European Torus (JET) are: (1) that the scientific and technical activities constituting the JET programme are integrated within a larger set of activities focused on the preparation of the next step, ITER; (2) that the facilities are now used in a spirit similar to that of other large scientific facilities in fields such as particle physics, neutron physics and synchrotron radiation: the programme is proposed by scientific teams formed across several European laboratories, who come to the JET Facilities to conduct experimental campaigns.

The change in organisation has triggered a reinforced co-ordination of scientific activities, providing a fruitful cross-fertilisation between fusion facilities and scientific teams, driven by the unprecedented mobility of European fusion researchers and has enabled fusion laboratories to assemble and combine complementary expertise. On the basis of overall objectives the scientific and technical programme is elaborated in close co-operation with all the EFDA Associates. International collaborators have shown a strong interest in participating. This represents a working example of the European Research Area proposed by the European Commission for the Sixth Framework Programme.

The present Report of Activities covers the period of 1 January 2000 to 31 March 2001. This period corresponds to the first four JET experimental campaigns under EFDA, which were conducted under similar experimental conditions and brought a coherent set of scientific results, in particular for the preparation of the ITER operating scenarios. The main scientific achievements relate to the consolidation of the ITER reference scenario and to the development of scenarios with strong internal transport barriers, which showed significant progress and promise for a steady state scenario.

Besides the four Campaigns the main achievements during the period covered are: the establishment of the new organisation and management structure under EFDA; the setting up of the required administration; the organisation of the Operator's work and of the Task Forces; the handing-over of the Facilities to the UKAEA at the termination of the Joint Undertaking; the restart of the operation after a short shutdown following hand-over; the launching of enhancement projects for implementation in 2001-2002; the studies of possible further enhancement of the JET Facilities for the period beyond 2002.

These achievements have been made possible by the combined efforts of a large number of people, the staff from the Associations involved in Task Forces and Projects, the Task Force Leaders and Project Leaders, the UKAEA Operator staff under the direction of Frank Briscoe and the staff from the EFDA Close Support Units. The Heads of all the European laboratories involved in EFDA and the European Commission have fully supported and facilitated these activities. A number of non-European collaborators have also contributed to the success of the experimental campaigns. My thanks go to all those involved for their contribution to the undoubted success of JET under EFDA.

The EFDA Associate Leader for JET
J. Paméla

Introduction by the UKAEA Senior Manager

UKAEA, as Operator, has faced a big challenge in assisting the transition to the new arrangements for exploiting JET and in making their realisation a success. In particular, there has been a large organisational change - the UKAEA staff returning from the JET Joint Undertaking have joined with those in the former UKAEA Culham Division to form a new integrated whole. At the same time to replace the loss of many key staff from the JET Joint Undertaking in 1999, there has been much redeployment of the existing staff and appointment of new staff, including over 30 seconded Associate staff.

UKAEA has also had to introduce its own management systems in many areas where the ones used by the JET Joint Undertaking had lapsed with the disappearance of JET as a legal entity, e.g. the systems for finance, procurement, and safety, while adopting some of the old systems which could be retained, e.g. the technical control system for making modifications to the JET Facilities. In addition, UKAEA has worked with the EFDA Close Support Unit and with the Task Forces to establish the user-facility nature of the new arrangements in helping to define the interfaces between Scientific and Technical (S/T) staff and operator staff in relation to both experiments and enhancements.

UKAEA has delivered a high percentage of the planned experiment days and achieved a good level of system performance. Inevitably there have been a number of equipment faults, but there has been a good response to fix these or to work around them with the co-operation of the Task Forces. Good progress has also been made on the first wave of enhancements notwithstanding the complexity of the new arrangements with equipment being designed by Associates and manufactured by industry while meeting UKAEA standards for installation and commissioning. Some of these enhancements are now ready to be installed during the 2001 shutdown period.

The UKAEA Senior Manager
F. Briscoe

1. On the Path to Controlled Thermonuclear Fusion

Can we produce useful power from the fusion of the nuclei of the light elements? This question has driven a large international research effort which has now provided sufficient understanding for the construction of an International Thermonuclear Experimental Reactor (ITER). The scientific and engineering input for the design of ITER came from many fusion facilities world wide. Among these, the Joint European Torus (JET) has played a particular role, being one of the two largest fusion facilities with the Japanese tokamak JT60-U, and having several unique capabilities:

– a tritium handling plant,
– remote handling capability,
– beryllium-coated walls.

Since 1 January 2000 JET has been operated under new arrangements, in the framework of the European Fusion Development Agreement (EFDA), with a continued and reinforced focus towards ITER. Further key scientific results were obtained, contributing in particular to the consolidation of the ITER operating scenarios. The new organisational framework is described in Section 2 of the present report, while the main scientific achievements obtained during the period 1 January 2000 to 31 March 2001 are summarised in Section 3. An overview of the operation of the JET facilities during this period is given in Section 4. The enhancements of the JET facilities are described in Section 5. Significant contributions from non-European laboratories are subject to international collaborations listed in Section 6. The ITER project is described in Appendix 7.1 and Nuclear Fusion Basics are given in Appendix 7.2.

The present section gives a brief history of the Joint European Torus and of its contribution to the ITER design and describes the JET tokamak.

1.1 History of JET

Construction of the JET tokamak started in 1978 under a Joint European Undertaking. The facility has been operational since 1983 (see aerial view of site Fig 1.1). Since the closure of the US tokamak TFTR in Spring 1997, JET is the only experiment in the world capable of operating with the deuterium-tritium fuel mixture foreseen to be used in future commercial fusion power stations.

In November 1991, JET (Fig. 1.2) was the first machine worldwide to produce controlled fusion power, nearly 2 MW peak and averaging 1 MW over two seconds with a fuel mixture of 90% deuterium and 10% tritium.

Thereafter JET has been enhanced by the installation of a divertor to handle higher levels of exhaust power. Deuterium experiments in the ITER geometry have made essential contributions to the ITER divertor design and provided key data on heating, confinement and fuel purity. This has contributed to define the size, heating requirements and operating conditions of ITER.

Figure 1.1: Aerial view of the JET site

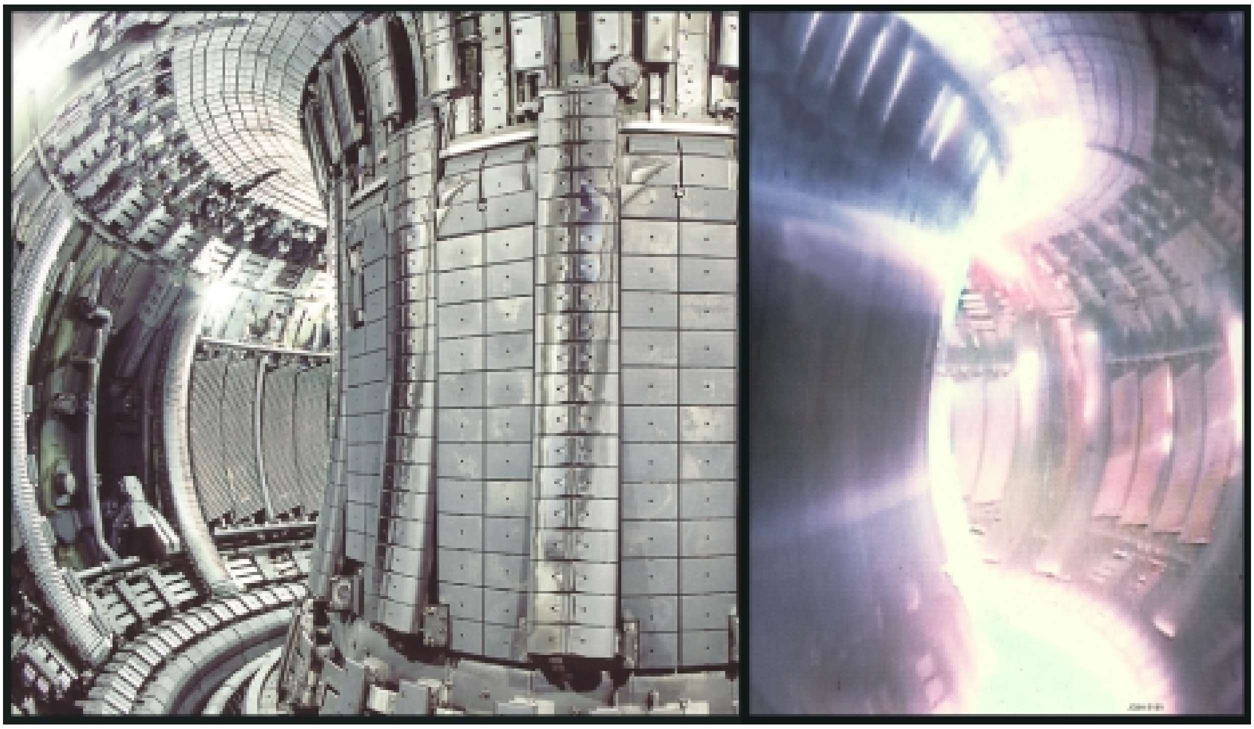

Figure 1.2: Inside JET (double view. The right hand side shows an infrared image in the presence of a plasma)

During 1997 the JET operations included a three months campaign of highly successful experiments using a range of deuterium-tritium fuel mixtures. The results were of major significance to the development of the fusion power option. JET set three new world records:

- 22 MJ of fusion energy in one pulse
- 16 MW of peak fusion power
- a 65% ratio of fusion power produced to total input power.

Subsequently, in Spring 1998 the fully remote handling installation of an ITER-specific divertor was successfully completed on time, demonstrating another technology vital for both ITER and a fusion power station. Experimental work continued in 1999, in particular to characterise the new divertor configuration to control impurities and plasma density and to develop Internal Transport Barrier scenarios in preparation of ITER.

The organisation which built and operated JET over 16 years, the JET Joint Undertaking, reached its end on 31st December 1999. The ownership of the JET Facilities was transferred into the custody of the UK Atomic Energy Authority (UKAEA), whereas the overall implementation and co-ordination of further scientific exploitation would in future be carried out under the European Fusion Development Agreement.

1.2 The JET Tokamak

JET uses the tokamak magnetic field confinement system to maintain isolation between the hot plasma and the walls of the surrounding vacuum vessel. The basic components of the Tokamak's magnetic confinement system are (see Fig. 1.3 and 1.4):

- The toroidal magnetic field which is produced by coils surrounding the vacuum vessel.
- The poloidal magnetic field produced by a current in the plasma; the plasma current is induced by transformer action, with the plasma itself forming the secondary circuit.

Additional coils, around the outside of the vacuum vessel, shape and position the plasma.

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 coils (outer poloidal field coils) used for positioning, shaping and stabilising the position of the plasma inside the vessel.

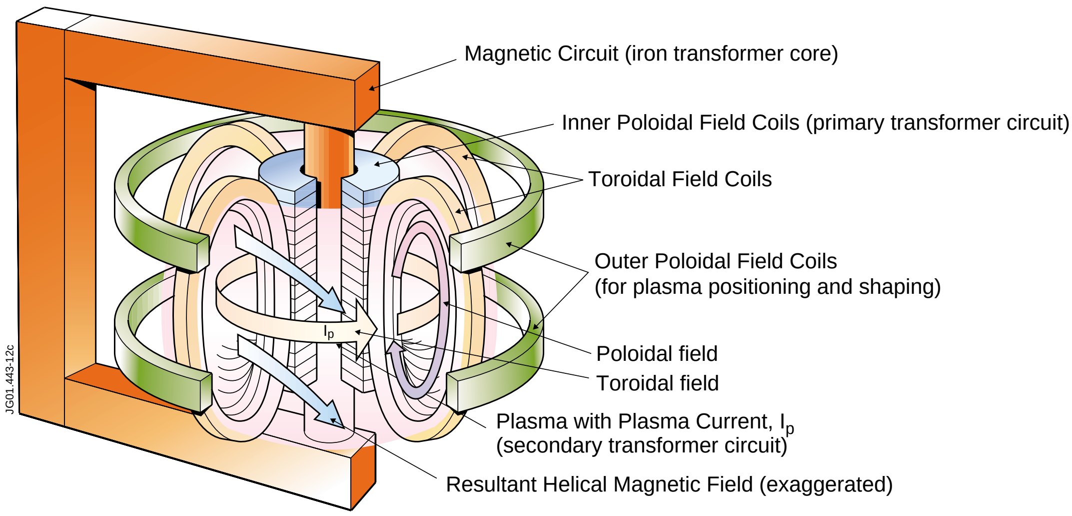

Figure 1.3: Magnetic Field Configuration

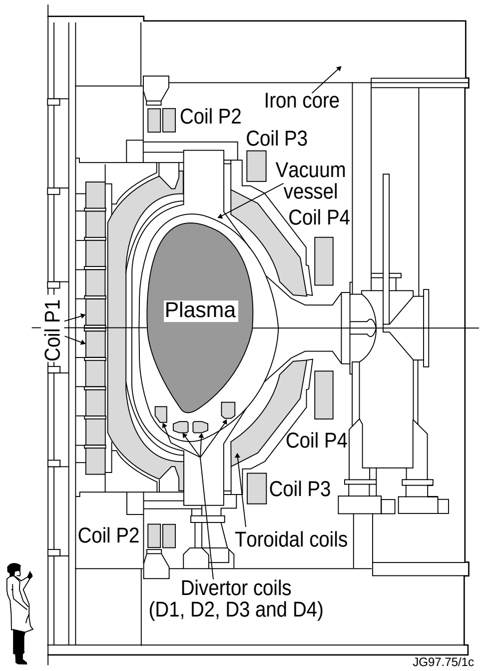

Figure 1.4: Cross-section showing the toroidal, poloidal and divertor coils

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 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 original main design parameters are presented in Table 1.1. In some cases these parameters have been far exceeded at a later stage.

PARAMETER	VALUE
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 1.1: Original JET parameters

2. Organisation

2.1 The European Fusion Development Agreement

Since 1 January 2000, the collective use of the JET Facilities is based on a new set of Agreements signed by the European Community and Associates from the European Union and associated countries:

- the “European Fusion Development Agreement” (EFDA),
- the “JET Implementing Agreement” (JIA), and
- the “JET Operation Contract” (JOC).

The main agreement, EFDA, provides the framework for implementing research, development and design work relating to technology activities in the Associations, the collective use of the JET Facilities and the European contribution to international collaborations, in preparation for the possible construction of an experimental reactor such as ITER. The conditions for the collective use of the JET Facilities are provided by the second agreement, the JET Implementing Agreement (JIA). In the frame of the JIA, the JET Operation Contract (JOC) entrusts the operation of the JET Facilities to the UKAEA. This set of agreements/contracts is complemented by “Rules for the Secondment of Associates’ Staff” which are used to second professional staff to Culham for temporary periods.

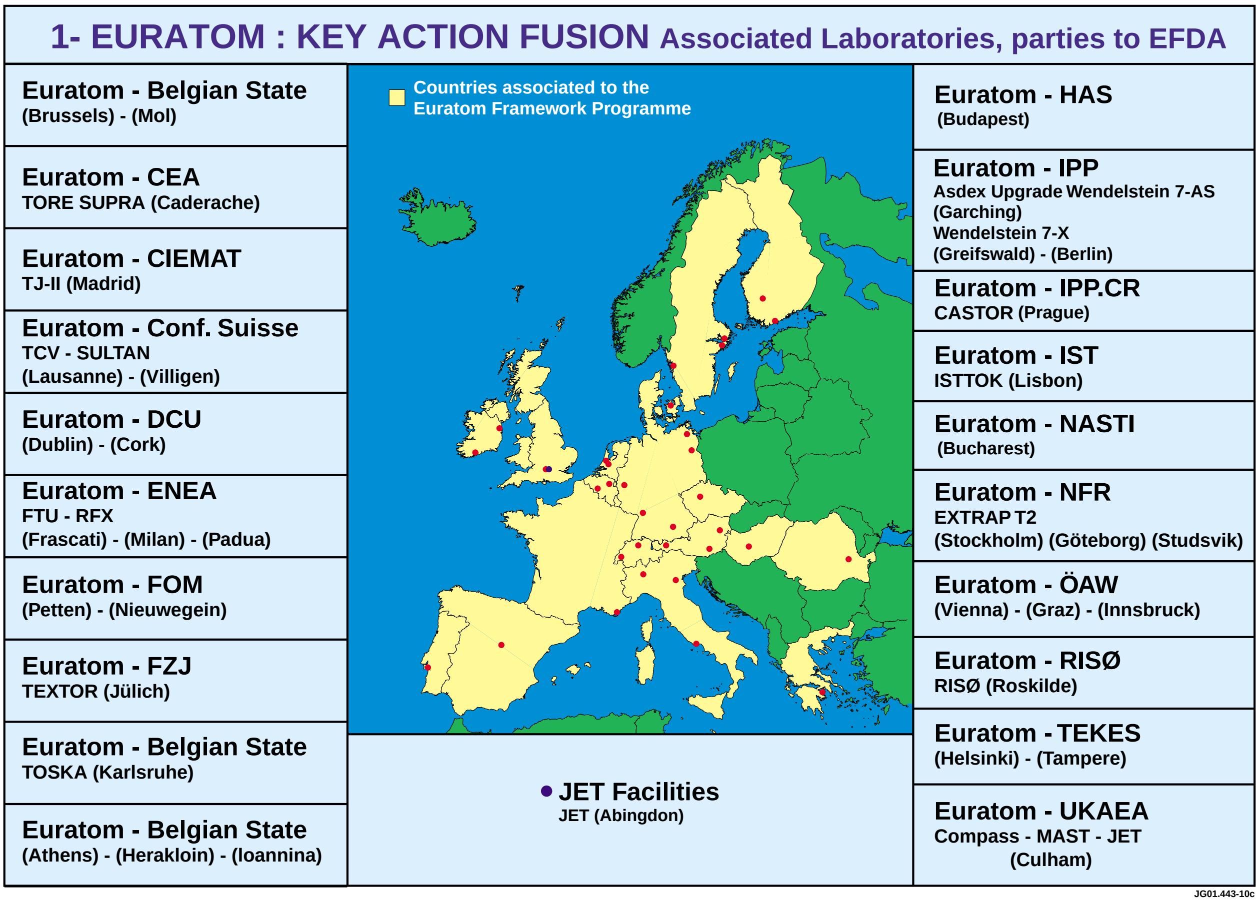

Figure 2.1: Map of Europe with Association sites

The objectives, schedule and overall resources of the EFDA activities are defined in the three-year Workplan. Within this the yearly JET Workprogramme consists of scientific and technical tasks (Experimental work and Enhancements) as well as the Operation of the JET Facilities required for its implementation. The implementation of the various tasks is conferred to EFDA Associates (Fig. 2.1) under specific sub-contracts which are placed further to calls for proposals. In addition to the work performed by Associates, specified work required for Enhancements is contracted to Industry.

An EFDA Steering Committee established by the contracting parties is responsible for the overall planning and supervision of the activities. For the supervision of the implementation of activities in specific areas two sub-Committees have been set up, one for the use of the JET Facilities and the other for Technology Activities*. The Steering Committee has appointed an EFDA Leader and Associate Leaders for JET and Technology, respectively. The EFDA Associate Leader for JET has the overall responsibility for the use of the JET Facilities. In order to assist the three Leaders in their duties, two Close Support Units have been established at the two host institutions, the UKAEA at Culham, United Kingdom and the Max Planck Institut für Plasmaphysik at Garching, Germany.

2.2 Use of JET under EFDA

One of the key features of the new way of working is that the experimental programme is organised in a campaign oriented manner. There are no permanent scientific staff on site (apart from UKAEA staff). The experimental programme is conducted by scientists visiting JET for limited periods and most of the complementary scientific work is undertaken from the Associates' laboratories. More generally, the work is conducted by EFDA Associates themselves. Consistent sets of tasks meeting programmatic objectives are eloborated in contact with the Associates. Full technical responsibility is conferred to Associates as follows:

- experiments are conducted by Task Forces led by Task Force Leaders, all of whom belong to the Associates and are based in their home laboratory;
- the operation of the JET Facilities is UKAEA's responsibility under the JOC;
- the enhancements of the JET Facilities are organised in projects, involving Associates and under the technical responsibility of Project Leaders belonging to Associates.

The overall co-ordination is achieved by a small team, the Culham Close Support Unit (CSU) under the EFDA Associate Leader for JET.

Extensive use is made of Remote Participation (see appendix 7.3).

The respective roles and structures of the CSU, Task Forces, Operator and Enhancement Projects are further described in the next sections.

2.3 The Culham Close Support Unit

The Culham Close Support Unit (CSU) assists the EFDA Associate Leader for JET. It comprises professional staff seconded to Culham by the EFDA Associates. The organisation of the Departments of the CSU reflects the specific duties conferred upon the EFDA Associate Leader for JET (see Fig 2.2).

The Programme Department is in charge of co-ordinating the development of the experimental programme,

*Appendix 7.7 lists the meetings of the EFDA committees held during the reference period.

the preparation of the experimental Campaigns, as well as the execution and monitoring of these campaigns and post campaign work. It deals with data consistency and co-ordinates the analysis and predictive code development and scientific reporting. The head of the Department co-ordinates the Task Forces and is also responsible for international collaborations.

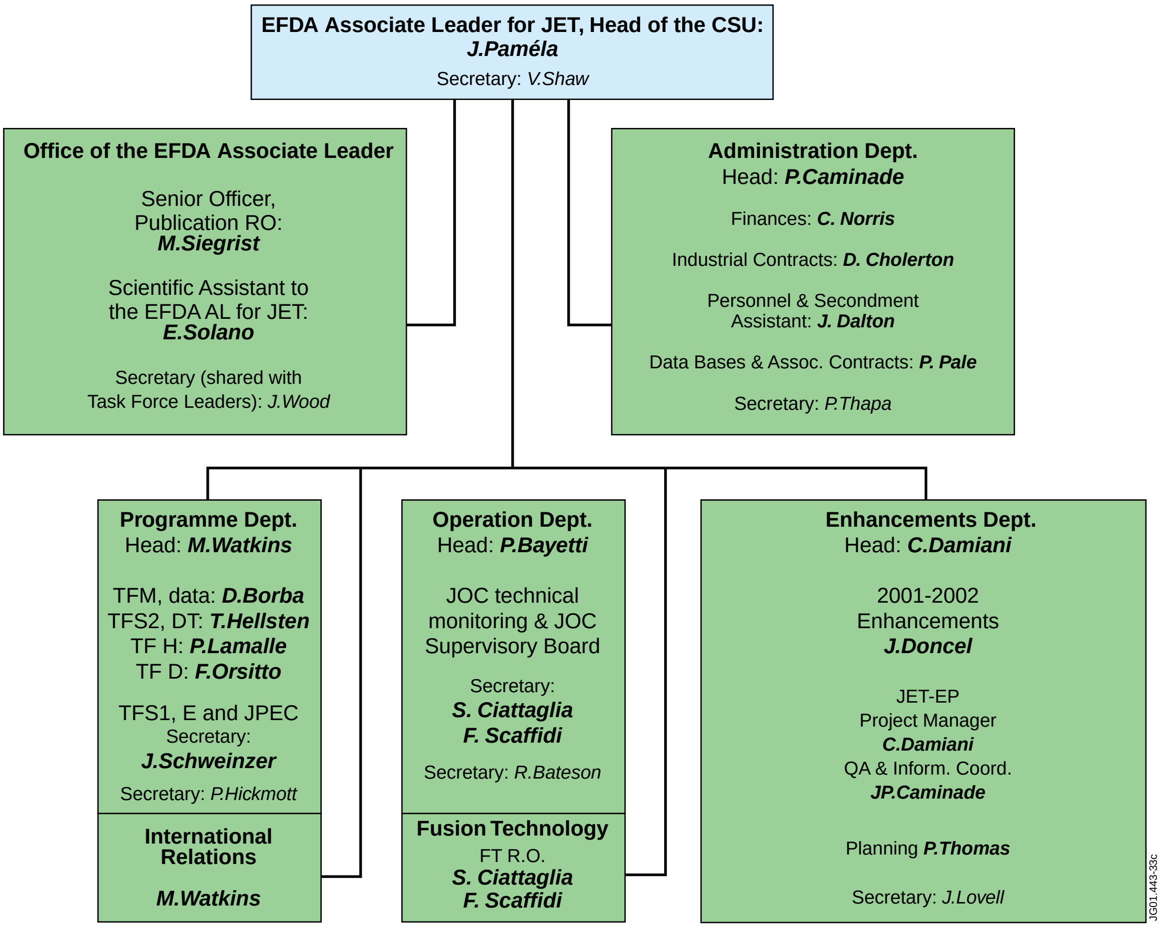

Figure 2.2: EFDA Close Support Unit-Culham Organisation Chart (As of 31 March 2001)

The **Operation Department** provides the main link with the Operator and is responsible for the technical monitoring of the JOC. It prepares the Reference Plan of Exploitation of the JET Facilities required for the implementation of the JET Workprogramme. It follows up Fusion Technology actions in liaison with the CSU Garching.

The **Enhancement Department** plans, organises, co-ordinates and monitors the enhancements of the JET Facilities.

The **Administration Department** is in charge of finances and budget planning, contracts, personnel administration and logistical support for the Scientific and Technical (S/T) secondees.

The **Office of the EFDA Associate Leader** provides scientific assistance and deals with scientific communication and publications, as well as remote participation in liaison with the CSU Garching and is responsible for Public Information issues.

2.4 The Task Forces

Task Force Leader	Deputy	Task Force	Association	Location
Jef Ongena		S1	ERM	Belgium
	Wolfgang Suttrop	S1	MPI	Garching, Germany
Alain Bécoulet		S2	CEA	Cadarache, France
	Robert Wolf	S2	IPP	Garching, Germany
	Paolo Buratti	S2	ENEA	Frascati, Italy
Volker Philipps		E	FZJ	Jülich, Germany
	Guy Matthews	E	UKAEA	Culham, U.K.
Olivier Sauter		M	CRPP	Lausanne, Switzerland
	Tim Hender	M	UKAEA	Culham, U.K.
Angelo Tuccillo		H	ENEA	Frascati, Italy
	Jean-Marie Noterdaeme	H	IPP	Garching, Germany
Joaquin Sanchez		D	CIEMAT	Madrid, Spain
Ralf-Dieter Penzhorn		FT	FZK	Karlsruhe, Germany
Derek Stork		DT	UKAEA	Culham, U.K.

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Figure 2.3: The Task Force Leaders

The experimental Campaigns on JET are conducted by Task Forces which address key scientific topics in preparation for the ITER experimental programme in a co-ordinated and consistent manner. The Task Forces are led by Task Force Leaders assisted by deputies (see Fig.2.3). The Task Force members, including the Leaders, are scientists employed by the EFDA Associates. These scientists come to Culham for limited periods of time to participate in part or all of the experimental Campaigns on JET. The stays at Culham are generally covered by EFDA “Secondment Agreements”.

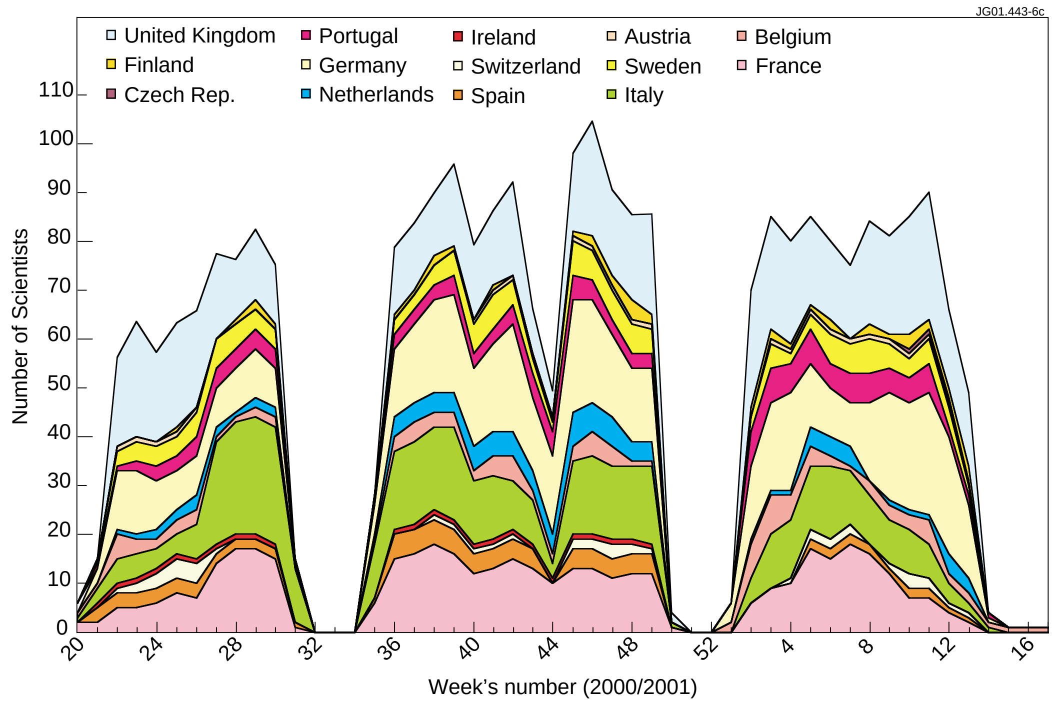


Figure 2.4: Scientists involved in campaigns C1-C4 by European country

More than 300 European physicists participated in the experimental campaigns C1-C4 (May 2000-March 2001), of which 248 from 14 laboratories came to Culham (Fig 2.4). Up to 104 physicists from Task Forces were involved at any one time.

Organising the logistic support for such a number of visitors with a rapid turn-over was a new type of challenge in the field of fusion. The CSU, UKAEA and Administrative Contact Persons designated by the Associates worked in close collaboration to prepare and organise the necessary support. A network for assistance to secondees was organised. This has provided a wealth of information: a "Welcome Pack" and a "Secondee's Handbook" were prepared; an internal website allowing fast access to key and up-to-date information was developed. A team jointly organised by the CSU and UKAEA (Secondees Assistance Team) dealt with all administrative procedures when secondees arrived on site to allow them to become operational as rapidly as possible: help in finding accommodation, organisation of collective transport, provision of offices and computing tools, flexible e-mail facilities, access to JET data etc.

2.5 The Operator

Under the terms of the JOC, UKAEA is responsible for operating the JET Facilities to meet the requirements of the EFDA S/T programme. The operation of JET is integrated, together with the work of the Euratom/UKAEA Fusion Association, into the activities of UKAEA at Culham, the structure of which is shown in figure 2.5 . The roles of the various UKAEA management units in the operation of JET are as follows:

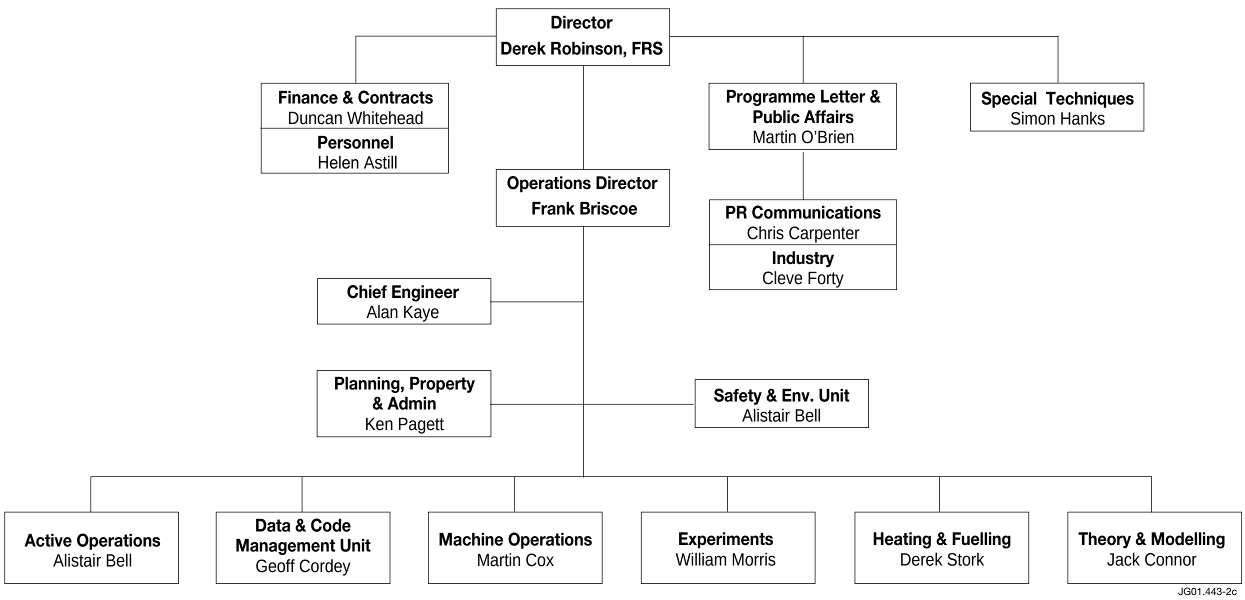


Figure 2.5: Operator Structure

The **Active Operations Department** provides services to the other operational Departments related to the safe supply and handling of tritium for the JET machine, management of radioactive and special wastes, decontamination, obtaining official approvals for operation with radioactive materials, including liaison with the UK Environment Agency. The Department is also responsible for maintenance of some services including non-pulsed power, ventilation systems and building safety systems.

The **Experiments Department** operates systems and provides services in support of machine operations, in particular in diagnostic systems. It also contains the core of expert Session Leaders and acts as the main point of contact for Task Force staff.

The **Heating and Fuelling Department** develops and operates the additional heating systems in support of machine operations. It is responsible for: Neutral Beam Injection, Ion Cyclotron Resonance Heating and Lower Hybrid Current Drive heating; Pellet fuelling and Cryogenic systems.

The **Machine Operations Department** is responsible for: machine operations; vacuum systems; shutdown maintenance; equipment installation and planning; remote handling; engineering of in-vessel components and diagnostics; Design Office services. The Department is also responsible for the pulsed power supplies to the JET systems and for the control and data acquisition systems.

The **Planning, Property and Administration Department** is responsible for the financial management of the JET Operations Contract. The Department also provides non-technical services including support to EFDA secondees and publication services.

The **Safety and Environment Department** provides independent advice to the Head of Site on safety-related aspects of operations at Culham. The Department is also responsible for health physics services (including beryllium safety), industrial safety and emergency planning and the control and development of the management system including quality assurance.

The **Chief Engineer's Unit** provides support to the Chief Engineer in ensuring the safe and effective operation and development of JET.

The **Data and Code Management Unit** is responsible for the processing of data from the JET diagnostics, the maintenance and development of the main physics database PPF (Physics Pulse File) and the high level database CPF (Central Pulse File). The Unit also manages the provision of a suite of advanced analysis codes for use by members of the Task Forces.

Other Divisions of UKAEA provide commercial and human resources services through Departments based at Culham. The Departments are:

- **Culham Commercial Department**, part of UKAEA Finance and Commercial Division, consisting of Contracts, which approves the issue of all external contracts and provides assistance with procurement strategies, contract conditions and commercial negotiations, and the Finance, which provides financial services to Culham Division.

- **Culham Personnel Department**, part of UKAEA Corporate Services Division, which provides human resources services to Culham Division and its staff.

2.6 Enhancements

As part of the activities conducted under EFDA, enhancements are prepared and implemented to increase the performance capabilities of the JET Facilities. Project Teams, involving one or several Associations, are established to undertake the work related to Enhancement Projects. Project Leaders are appointed by the EFDA Associate Leader for JET as proposed by the Associations involved. Project Leaders are responsible to the EFDA Associate Leader for JET. Each Project Leader is in charge of co-ordinating the design and construction of the equipment needed for the implementation of the Projects. For each project, an Operator representative is nominated by the UKAEA Senior Manager. He ensures that the Operator's safety and quality requirements for integrating enhancements in the JET facilities are taken into account; he also organises the installation, commissioning and future operation of the new equipment for which the Operator is responsible.

The list of all the approved Enhancement Projects, Project Leaders, Operator Representatives and involved Associations is given in tables 2.1, 2.2 and 2.3.

2.7 Resources

The resources for the use of JET under EFDA are provided by the European Commission (EURATOM), a Joint Fund from the EFDA Associates and direct funding from Associates. During the period of reference, up to 19 professionals and 8 support staff from 11 different Associations worked in the Close Support Unit. The Operation of the facility required on average about 400 persons during the year 2000. The average yearly expenditure under the JET Operation Contract (JOC) is foreseen to be 54.3 MEuros over the period 2000-2002. The involvement of EFDA Associates in the experimental Campaigns and enhancement projects during the year 2000 was about 124 ppy in total under EFDA contracts. The breakdown between Association is given in Fig. 2.6. Industrial Contracts for about 8 MEuro have been launched to conduct enhancements.

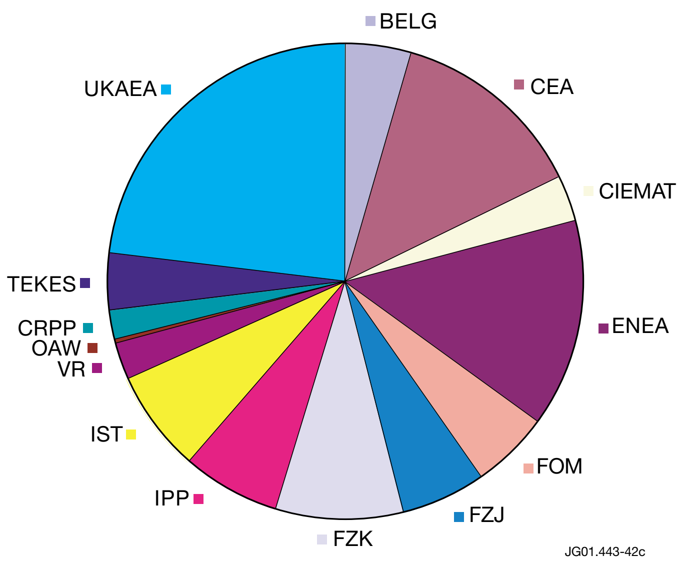


Figure 2.6: Breakdown of Associations' Involvement (manpower) in the JET 2000 Work programme

Enhancements to be Implemented in 2000-2002

Project	Project Leader	Operator Representative	Other involved Associates/Laboratories
Improvement of Neutral Beam Injector in Octant 8	T Jones D Edwards, D Martin (UKAEA)	D Edwards	–
Error Field Correction Coils	M Bigi (UKAEA)	S Hotchin	CNR
Quartz Micro-Balance	G Esser (IPP)	G Neill	UKAEA
Edge LIDAR Thomson Scattering Detection System Upgrade	R Pasqualotto (RFX)	C Gowers	UKAEA
Reciprocating Probe Heads Upgrade	C Hidalgo (CIEMAT)	K Erents	VR CRPP UKAEA
Microwave Reflectometry for Turbulence Studies	L Cupido (IST)	R Prentice	UKAEA IPP PPPL(US)
Upgrade of Motional Stark Effect System	N/A	N Hawkes	PPPL(US)
Spectrometer for High Time Resolution Pellet Ablation Measurements	N/A	M Stamp	PPPL (US)
D2 Pellet Injector Upgrade	M Watson (UKAEA)	D Wilson	CEA
Control system for improving extreme shapes on JET Plasmas	F Crisanti (ENEA)	F Sartori	UKAEA CRPP
Upgrade of Electron Cyclotron Emission Radiometer	E de la Luna (CIEMAT)	J Fessey	IPP
Real Time Control Upgrade	E Joffrin (CEA)	R Felton	IST
Upgrade of the Li Beam Diagnostics	S Zoletnik (HAS)	P Morgan	IPP
Fast He Beam Spectroscopy	H-D Falter (IAP)	Y Andrew	FZJ
New Fast ADC digitizers for the Central Acquisition and Trigger System (CATS)	B Alper (UKAEA)	S Dorling	–
Michelson Interferometer for Electron Cyclotron Emission System (to be moved from FTU Frascati)	M Zerbini (ENEA)	J Fessey	–

Table 2.1:

Studies and R and D in Preparation of Possible Enhancements

Project	Project Leader	Operator Representative	Other involved Associates/Laboratories
Ion Cyclotron Second Harmonic Protection	F Durodie (ERM)	T Stanley	–
Improvement of Lower Hybrid Current Drive Coupling	A Tuccillo (ENEA)	J Mailloux	CEA CNR Milano
3DB Coupler for Ion Cyclotron Resonance Frequency heating	F Durodie (ERM)	I Monakhov	IPP
Improved Neutral Beam Injector neutraliser	B Ellingboe (DCU)	S Cox	UKAEA

Table 2.2

Design Studies for other and Longer Term Enhancements

Project	Project Leader	Operator Representative	Other involved Associates/Laboratories
New Ion Cyclotron Resonance Heating system equipped with an ITER-like antenna	F Durodie (ERM)	J Fanthome	CEA,ENEA, FZJ,IPP,IST, UKAEA
Electron Cyclotron Resonance Heating System	T Verhoeven (FOM)	C Fleming	CIEMAT, CRPP ENEA, FZK
Divertor and Divertor Diagnostics	Ph Chappuis (CEA)	E Villedieu	ENEA, IPP, FZJ, IST
New IR Viewing camera	E Gauthier (CEA)	P Coad	FZJ, IPP,
Halo Current Sensors Upgrade	P Fiorentin (RFX)	V Riccardo	–
Micro-wave Access (new waveguides for ECE and reflectometry)	L Cupido (IST)	TBD	CNR Milano (TBC) PPPL (US)
Tritium Retention Studies	P Coad (UKAEA)	P Andrew	TEKES, FZJ, IPP (TBC)
Charge Exchange Recombination Spectroscopy Upgrade	K-D Zastrow (UKAEA)	A G Meigs	SCK-CEN (TBC)
High Resolution Thomson Scattering System	R Pasqualotto (RFX)	S Saunders (Window only) T Edlington	IST, SCK-CEN (TBC) General Atomics (US)
Vertical Bolometer (KB5)	Giannone (TBC) (IPP)	TBD	CEA, RFX (TBC)

Table 2.3

3. Scientific Programme

3.1 Introduction

The collective use of the JET facilities for the period beyond 1999 provides a unique contribution to the consolidation of the scientific basis in plasma physics and plasma engineering and to the demonstration of high performance in operational modes relevant to the objectives and configuration of the Next Step (ITER).

3.1.1 Planning and Machine Operations

The machine activity during the period 1 January 2000 to 31 March 2001 (Fig. 3.1) started with a five month shutdown and restart phase. After the hand over of the JET Facilities in a very passive state to the UKAEA during the last week of 1999, the machine was first brought back into the standard state of JET maintenance (shutdown) and left in this state until the restart of operation on 16 March 2000. During restart, the various sub-systems were progressively brought back to their normal performance.

During the period 31 May 2000 to 30 March 2001, there were four Experimental Campaigns on JET. These comprised 155 of the 250 days target for Scientific and Technical (S/T) experiments during the three years 2000-2002. Approximately 20 useful pulses were achieved each day.

The first plasma for the Campaign C1 took place on 31 May 2000, as originally planned.

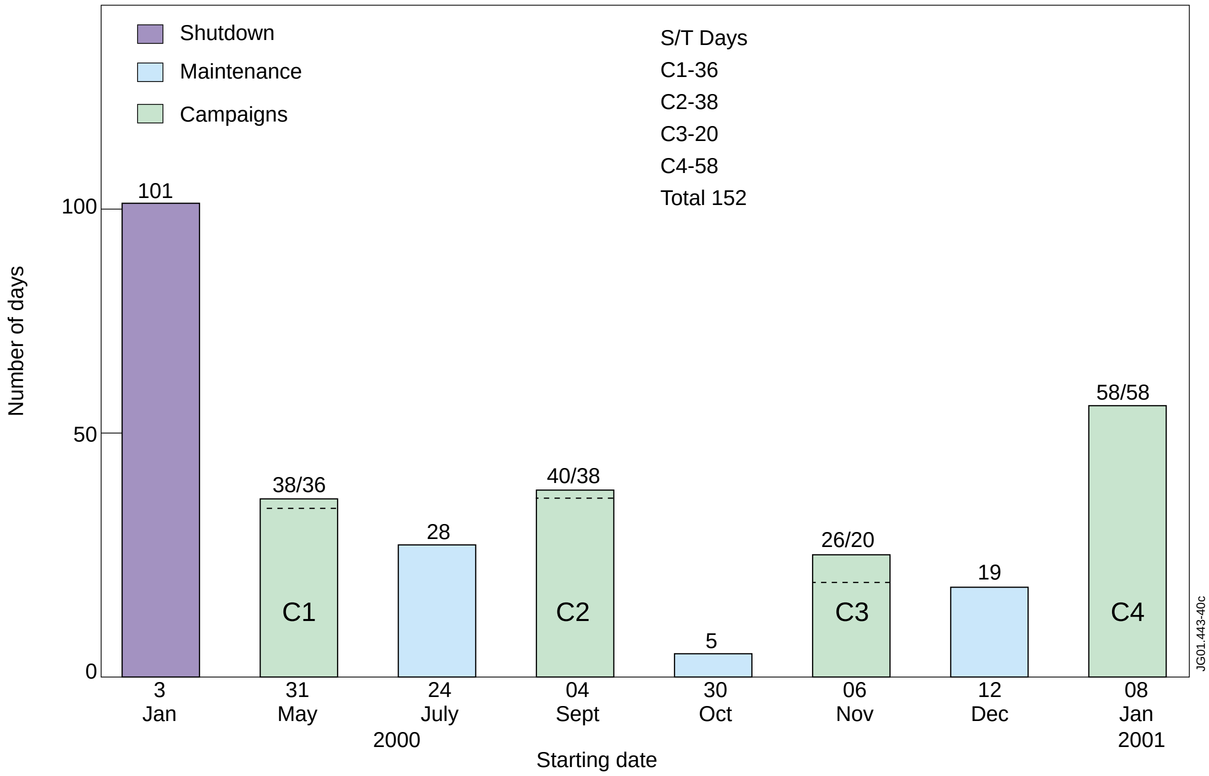


Figure 3.1: Machine activity during the period 1 January 2000 to 31 March 2001. Planned/achieved operation days are indicated for each campaign

The only significant failure during Campaign C1 was a water leak on a Neutral Injector Box. The system had to be withdrawn from service immediately, but the programme continued with reduced Neutral Beam power for

2 weeks. It was curtailed 1 week early to allow repair in time for the start of the second Campaign.

During the August intervention the leaking element was quickly identified and repaired and the system was re-commissioned up to the required power levels. Other planned activities were also completed on time. Two unexpected problems occurred which could not be fixed for Campaign C2, but they concerned systems which were not crucial to operation.

The only problem which occurred during Campaign C2 was a leak on one of the rotary valves; this was easily solved with only one day of operation lost. During the week between Campaigns C2 and C3 the Lithium Beam diagnostic was removed for refurbishment because its performance had become unsatisfactory.

Campaign C3 started on time on 6 November 2000 with the machine and heating systems performing well. However, an air leak terminated operations on the 15 November following a disruption. The leak was in a weld on a redundant diagnostic which was still connected to the torus vacuum. For repair the torus needed to be cooled to ambient temperature. Operation recommenced after a loss of 6 days of operation.

Campaign C4 started on 8 January 2001 after a period of standby during the turn of the year. Operation was with a lower, more ITER-relevant vessel temperature (200^{o}C rather than 320^{o}C) and included a period of helium discharges.

3.1.2 Focus of the Experimental Programme

The focus of the experimental programme was on consolidation of the physics basis of ITER. Strong priority was therefore given to experiments which:

- were required prior to the submission of the ITER Final Design Report;
- would have an influence on the details of the ITER design; and
- contributed to the preparation of the ITER operating scenarios.

The preparation and execution of the experiments and the subsequent analysis of the experimental results was organised under two scenario Task Forces, S1 (standard mode of ITER operation) and S2 (advanced mode of ITER operation), four specialised Task Forces, E (plasma exhaust, edge and divertor studies), M (MHD instability studies), H (heating, rotation and current drive studies) and D (diagnostics), and two Task Forces which had no explicit experimental time, DT (Deuterium-Tritium) and FT (Fusion Technology).

The key issues which the Task Forces addressed included the following elements:

- Improvement of overall plasma confinement and stability for operational scenarios with both modest (Task Force S1) and extensive (Task Force S2) requirements for profile control;
- Characterisation and improvement of plasma operation close to operational boundaries, in particular close to density and pressure limits; plasma shaping studies; confinement scaling studies; instability studies (Task Forces S1, S2 and M);
- Simultaneous optimisation of core and edge physics (Task Forces S1, S2 and E);

- Development of heat and particle exhaust physics and characterisation of the Mark IIGB divertor (studies on divertor operation, radiative layers, detached plasmas, in-out asymmetries between high and low field legs of the divertor, effect of differential gas puffing); studies on impurity control and plasma wall interaction (Task Force E);
- Helium plasma operation to improve the understanding of plasma edge and divertor physics, plasma wall interaction; divertor radiation, detachment and density limits; ELM behaviour; atomic physics; and impurity production, particularly in the absence of chemical erosion (Task Force E);
- Erosion of and (co-) deposition on plasma facing components; impurity flows in the scrape-off layer; migration of wall material and hydrogen retention in carbon surfaces (Task Forces E and FT);
- Steady state aspects of confinement optimisation: long pulse current drive, heating and fuelling (Task Forces S2 and H);
- Physics and engineering of heating and current drive systems; improvement of coupling; validation of radio frequency physics modelling; test of specific heating and current drive schemes (Task Force H);
- Mitigating techniques for off-normal events and improvement of the corresponding database (Task Forces M and E);
- Global Alfvén Eigenmodes which can be destabilised by fast ions if the damping of the waves is insufficient. The testing of models which predict that such modes in ITER are influenced mainly by mode conversion in the high shear edge region (Task Forces M and DT);
- Effect of lower vessel temperature on the formation of Internal Transport Barriers, recycling and impurity sources under high performance conditions (Task Forces S2 and E);
- Development and validation of diagnostics, control methods and systems in the specific environment of a large DT fusion device (Task Force D).

Joint experiments with medium-sized tokamaks were also undertaken to obtain important scaling information in several areas.

3.2 Task Force S1

3.2.1 Scientific Issues

Task Force S1 aims to consolidate the physics basis for the ITER reference operational scenario. This scenario is based on long pulse operation in which the plasma current is driven inductively and there is little control of the radial distribution of the current by other (non-inductive) means. The most promising type of discharge for this scenario is a regime with high confinement (the "H-mode") in which energy and particle losses at the plasma edge are reduced considerably (in comparison with other possible modes of operation), leading to steep temperature and density gradients and high overall plasma energy. These steep edge gradients can lead, however, to instabilities which manifest themselves as short periodic events Edge Localised Modes (ELMs), which help maintain quasi-stationary conditions, but which also produce high, unwanted, transient heat loads on plasma facing components, such as the targets of the divertor. This ELMy H-mode is a highly developed and robust scenario which has been demonstrated on many tokamaks. A full demonstration which combines **high performance** with **acceptable power exhaust** is of paramount importance.

3.2.2 Summary of Results

High Performance

To reach the ITER levels of performance with the reference operational scenario based on a quasi-stationary ELMy H-mode requires the simultaneous achievement of sufficiently **high levels of energy confinement** (that is, a confinement quality factor $H_{98(y,2)}$=1, where $H_{98(y,2)}$ is the confinement relative to the ELMy H-mode scaling expression, IPB98(y,2)*, **plasma pressure** (as given by the normalised beta value β_N=1.8), **density** (85% of the Greenwald density, n_{GW} [$10^{20}m^{-3}$] = I_p [MA]/(πa^2[m^2])) and low **impurity content**.

High Confinement at High Density

Experiments on JET and other tokamaks prior to Campaign C1 showed that it was increasingly difficult to maintain high confinement at high density. During Campaigns C1-C4, Task Force S1 concentrated its efforts on achieving discharges with the required levels of normalised confinement, density and pressure. Several techniques were used, including:

- Shaping the plasma, by increasing its triangularity;
- Seeding the plasma with controlled fluxes of impurities, at both high and low plasma triangularity;
- Fuelling the plasma with either (a) controlled and moderate gas puffs or (b) optimised pellet injection.

The results obtained using these techniques, alone or in combination, are presented in Fig. 3.2 which shows the dependence of the confinement quality factor $H_{98(y,2)}$ on the Greenwald density factor n/n_{GW}. It is clear that the requirements of the ITER scenario have been achieved, or even exceeded during the 2000-2001 Campaigns.

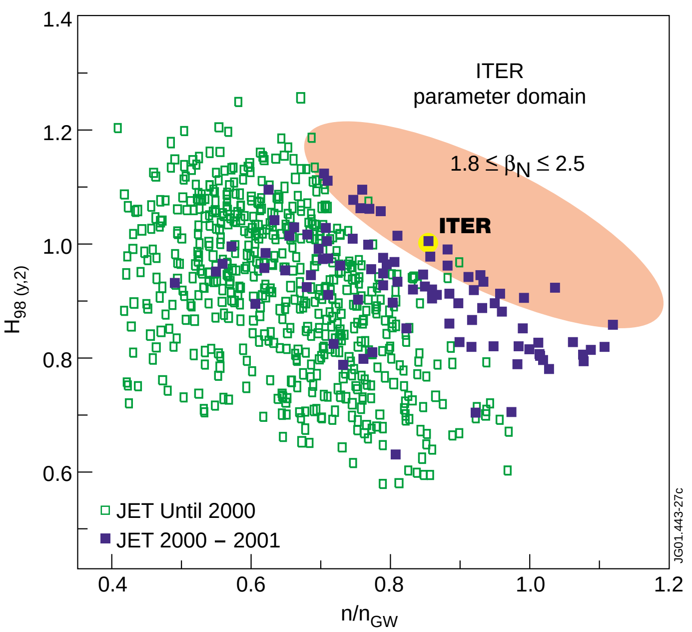


Figure 3.2: Data plotted in a $H_{98(y,2)}$ - n/n_{GW} diagram, from the JET ELMy H-Mode steady state database. Applying the different techniques listed above, this diagram shows clearly data with higher densities and confinement than obtained previously, consolidating the projected values for the ITER operating point.

Acceptable Power Exhaust

The power exhaust from a reactor must be compatible with first wall requirements. Of particular importance

*see Nucl Fusion 39 (1999) 2204

for the reference operational scenario based on a quasi-stationary ELMy H-mode is the power loss during ELMs, since this can cause serious damage because of ablation and/or melting of the target material. An estimate for ITER operating at Q=10 and 400MW fusion power, shows that the power loss per ELM should be at most 1% of the plasma energy in order to avoid the melting of tungsten and evaporation of carbon target materials. ELM mitigation is therefore of the utmost importance.

ELM Mitigation

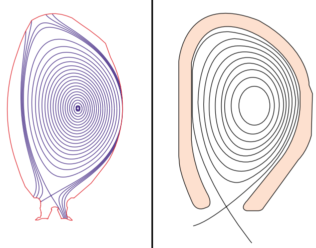
During Campaigns C1-C4, Task Forces S1 and E conducted several studies related to the mitigation of ELMs. Consideration was given to both the peak power flux during ELMs and the power flux averaged over many ELMs. It was found that the power load on the divertor target plates is indeed reduced in discharges with:

- Impurity seeding; and with
- Broadband MHD activity between ELMs.

3.2.3 Detailed Scientific Results

(i) High Confinement at High Density by Plasma Shaping

The plasma shape in the poloidal cross-section was varied in order to study its influence on plasma confinement. These experiments used a large number of different plasma shapes, including a shape which is very close to that projected for ITER (elongation κ=1.74 and average triangularity δ=0.45-0.5).



	SHAPING	
	JET "ITER shape" Pulse No: 53299, 2.5MA/2.7T	**ITER**
$H_{98\,(y,2)}$	0.91	**1.0**
$\beta_{N,th}$	1.90	**1.81**
n_e / n_{GW}	1.1	**0.85**
Z_{eff}	1.5	**1.7**
P_{rad} / P_{tot}	0.40	**0.58**
κ, δ	1.74, 0.48	**1.84, 0.5**
q_{95}	3.2	**3.0**
τ_{pulse} / τ_E	15	**110**

JG01.416-1c

Figure 3.3: Parameters of a JET ITER-like discharge compared with the required ones for the ITER conventional scenario.

In this so-called "ITER-like" configuration (κ=1.73 and δ~0.47) with a plasma current and toroidal magnetic field of 2.5MA and 2.7T and about 14MW of neutral beam heating, the best quasi-stationary conditions (several seconds, very close to the maximum duration possible under such operational conditions on JET) obtained simultaneously were a confinement quality factor $H_{98(y,2)}$ ~ 0.9, a normalised pressure β_N = 1.9 and a density n/n_{GW}=1.1, i.e. simultaneously at or above ITER requirements for all these key parameters (Fig. 3.3).

(ii) High Confinement at High Density in ITER-like Plasmas with Impurity Seeding

At high triangularity, argon seeded plasmas have a radiating mantle, increased levels of radiation (P_{rad}/P_{tot} increases from 30% to 60-70%) and a somewhat higher effective ionic charge Z_{eff} (~2.2 rather than ~1.9 (Fig. 3.4)). The plasma density is increased to 10% above the Greenwald level, generally above that achieved without impurity seeding. Furthermore, confinement is not degraded with argon seeding, remaining high ($H_{98(y,2)}$ ~ 1) and similar to that in non-seeded discharges. This scenario is very promising in particular when the beneficial effect on ELM mitigation (see iv) is taken into account as well.

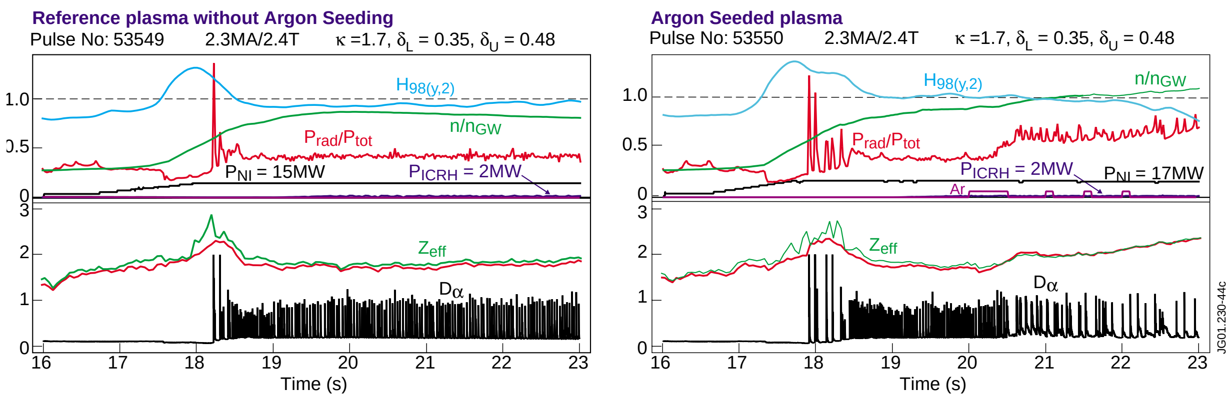


Figure 3.4: Illustration of the effect of Ar seeding in a high triangularity plasma (δ~0.47). Shown is a non-seeded reference plasma (left, Pulse No: 53549) and a plasma with Ar seeding (right, Pulse No: 53550). Both are Neutral Beam heated with a further 2MW of centrally deposited ICRH (H-minority heating). Ar seeding in Pulse No: 53550 starts at t=20s, and allows the good confinement and high density properties of high δ discharges to be combined with a high radiation level (P_{rad}/P_{tot} ~ 70%), in a radiative mantle around the discharge. In comparison with the reference discharge, similar confinement and higher densities are obtained : confinement stays at a high level $H_{98(y,2)}$ ~ 1, and the density rises slowly and reaches a value of n/n_{GW} =1.1. Note the marked decrease in the ELM frequency, as shown in the trace of the D_α light.

(iii(a)) Density Peaking during Long Pulses without Pellet Injection

For a given average plasma density and energy, fusion power increases with increased peaking of the density profile, thus optimising the fusion yield in a fusion reactor. Recently, several tokamaks have observed density peaking during long duration pulses. Dedicated experiments during Campaigns C1-C4 have shown such density peaking (Fig. 3.5) for a wide range of plasma currents (0.95-2.5MA) and plasma shapes. The plasma has been fuelled from various locations with the best results being obtained with fuelling through the private flux region (region beneath the X-point) of the divertor.

(iii(b)) High confinement at High Density with Pellet Injection

A 2.5MA/2.4T plasma with modest triangularity (δ~0.34) showed a confinement collapse when attempts were made to increase the density above 80% of the Greenwald density with gas puffing alone. On the other hand, long pulse pellet fuelling experiments have allowed densities close to the Greenwald density to be achieved, while still avoiding the strong confinement degradation usually associated with strong gas puffing.

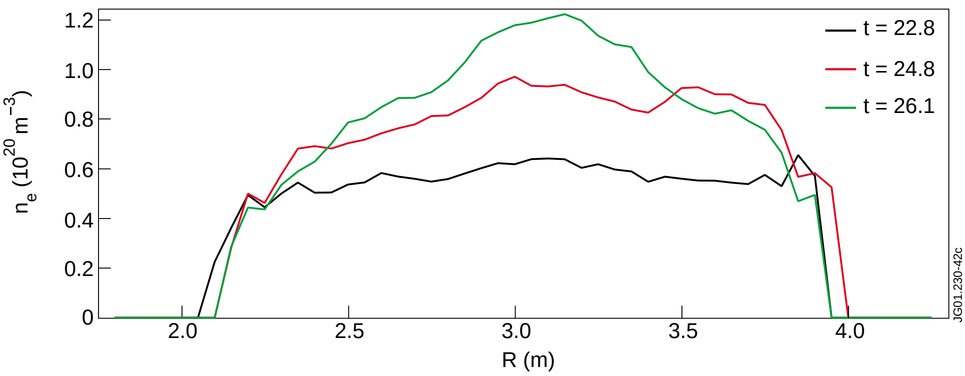


Fig. 3.5: Density profiles at the times indicated show significant peaking near the end of the discharge, with central density values of about 1.2 10^{20} m^{-3} and a peaking factor $n(0)/n_{ped}$ ~ 2.

These pellet fuelling experiments used solid ($4mm^3$) deuterium cubes launched at a speed of 160 m/s into the plasma from the high magnetic field side via an injection tube tilted by 44 degrees with respect to the horizontal plane and with a tangency radius at a normalised minor plasma radius ρ~0.6-0.7. The pellets can induce ELMs which lead to confinement losses, but this effect can be mitigated by adapting the pellet fuelling cycle. This leads to a recovery of the plasma energy while the particle inventory remains high. An optimised pellet sequence was found; this comprises an initial density build-up phase with pellets being injected at about 6 Hz followed by a density sustainment phase with pellets being injected at about 2 Hz. The density build-up is then sufficiently gentle for strong local cooling of the plasma to be avoided. Densities in excess of the Greenwald density can be achieved without significant persistent loss of plasma energy.

(iv) ELM Mitigation in ITER-like Plasmas with Impurity Seeding

Although ELMs have the same characteristics with and without impurity seeding, their frequency is reduced in the former case and the strong radiation results in strongly reduced temperatures on the surface of the divertor targets. This is shown clearly by infra-red thermography measurements in the inner divertor, and even more spectacularly in the outer divertor, where (i) the base level falls from about 500°C to about 100°C and (ii) the maximum temperature spikes are reduced from 1300°C to about 600°C with argon seeding (Fig. 3.6). While further work is needed to consolidate these findings, these results already show promise as a technique for ELM mitigation.

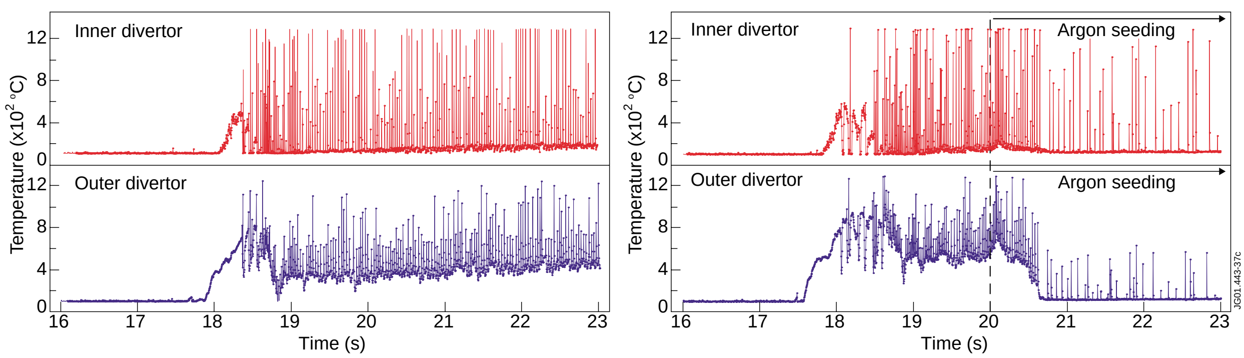


Figure 3.6: Infra-red thermographic measurements at a fixed location of the target plates in the inner and outer divertor legs for the discharges of Fig 3.4. Compared to the reference discharge (left), the maxima of the temperature spikes in both the inner and outer divertor legs are diminished. In addition, the base level temperature for the outer divertor leg is substantially reduced by a factor of 4-5.

(v) ELM Mitigation in the Presence of Broadband MHD Activity

Discharges which achieve high confinement at high density without impurity seeding show significantly different ELM behaviour with medium/high and low triangularity plasmas. In particular, at medium/high triangularity:

- the ELM frequency remains low, even with high rates of gas fuelling;
- the relative losses per ELM ($\Delta W_{ELM}/W_{dia}$) do not increase with decreasing ELM frequency; and
- the power flux averaged over ELMs decreases with increasing fuelling.

The second of these points is particularly surprising since until recently the relative losses per ELM showed a clear correlation with frequency ($\Delta W_{ELM}/W_{dia} \sim 1/f_{ELM}$), thus seeming to confirm the physical picture that the time between ELMs "accumulates" plasma energy which is then released during the ELM. The new "anomalous" and more favourable ELM behaviour may indicate that power is lost due to other mechanisms. When the edge density is in excess of 70% of the Greenwald density, the ELM frequency is often anomalously low and continuous broadband edge-localised MHD activity is observed between ELMs. A comparison of the power input and energy loss by Type I ELMs shows that additional energy is lost during the interval between ELMs. While the exact nature of the mechanism which leads to these enhanced losses is the subject of further studies, these increased levels of power and particle exhaust by continuous MHD activity rather than by ELMs could be of significant advantage to an operational scenario which is compatible with the first wall.

Remarkably, the various observations of ELM behaviour with pellet injection, impurity seeding, high and medium triangularity can be summarised uniquely in a relation which shows that the relative energy loss per ELM ($\Delta W_{ELM}/W_{ped}$) decreases with decreasing values of the collisionality of the edge plasma prior to the ELM. However, this could be due to processes associated with the relative duration of the phase of enhanced MHD activity and the characteristic time of the parallel energy loss, both of which depend in different ways on collisionality. Depending on the exact mechanism, the extrapolation of these results to ITER can differ substantially.

3.3 Task Force S2

Advanced scenarios based on enhanced confinement with a significant fraction of the plasma current generated via the bootstrap effect provide alternatives to the ITER reference ELMy H-mode scenario. The aim of Task Force S2 is to develop high performance steady state operation based on enhanced confinement using Internal Transport Barriers (ITB).

3.3.1 Scientific Issues

Plasma Confinement and Transport Barriers

In general, heat and particle transport in thermonuclear plasmas are governed by anomalous transport mechanisms. Such transport reduces substantially the temperature and density gradients for given heat and particle fluxes compared to those, which would have been achieved if the transport was governed by Coulomb collisions alone. Under some conditions the transport losses are substantially reduced, sometimes down to the neo-classical levels, the minimum caused by collisions in a toroidal plasma. The reduction in the transport is usually restricted to annular toroidal regions, transport barriers, and may appear either at the edge or internally. The existence of internal transport barriers is a fairly recent discovery. The conditions under which they appear are not fully understood, although substantial progress has been made. The formation and evolution of an internal transport barrier as seen in the ion temperature profile is demonstrated in Fig. 3.7. As the barrier develops, the electron temperature and density gradients also increase in the barrier.

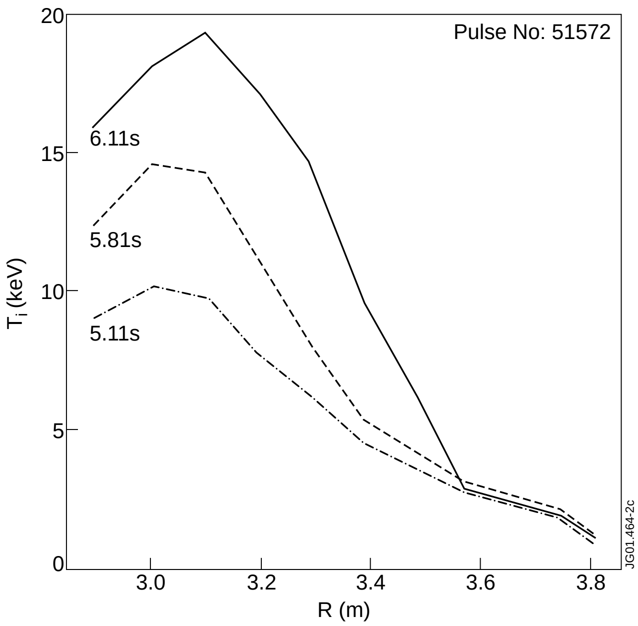


Figure 3.7: Development of an internal transport barrier. The ion temperature profile before the development of the barrier (dashed dotted line), after the barrier developed at R=3.45m (dashed line) and after the barrier expanded to R=3.65m (full line). The heating power is constant for a few energy confinement times before and after the barrier has developed. The temperature profile had reached a quasi steady state before the barrier developed.

Steady State Operation

Steady state operation of tokamaks is constrained by the need to drive the plasma current, which is required to confine the plasma. Currents inductively driven by a transformer restrict operation to long pulses, but currents driven non-inductively, using waves and neutral particle beams, can provide steady state operation. These methods have a rather low efficiency, but this can be improved if a large fraction of the plasma current is driven by the bootstrap current arising from the pressure gradient in a magnetically confined toroidal plasma. High performance discharges with internal transport barriers offer a promising route to steady state operation with a large fraction of the plasma current driven by the bootstrap current. The build-up of the total plasma current, the fractions of bootstrap and beam driven currents, for a discharge in which a transport barrier develops, are shown in Fig. 3.8.

Real Time Control of Plasma Profiles

Steady state operation requires control of the profiles of plasma current, temperature, density and rotation in order to prevent instabilities, which may destroy the transport barrier or terminate the discharge. The pressure and current gradients are in general limited by instabilities, which can produce soft limits, preventing steep gradients, or hard limits, destroying confinement. Since internal transport barriers appear in plasmas with hollow, or nearly flat current profiles and the electric resistivity in the plasma decreases with increasing temperature there is a natural tendency for the plasma current to peak centrally. This makes the real time control of the profiles a challenging task of paramount importance for maintaining the transport barrier and

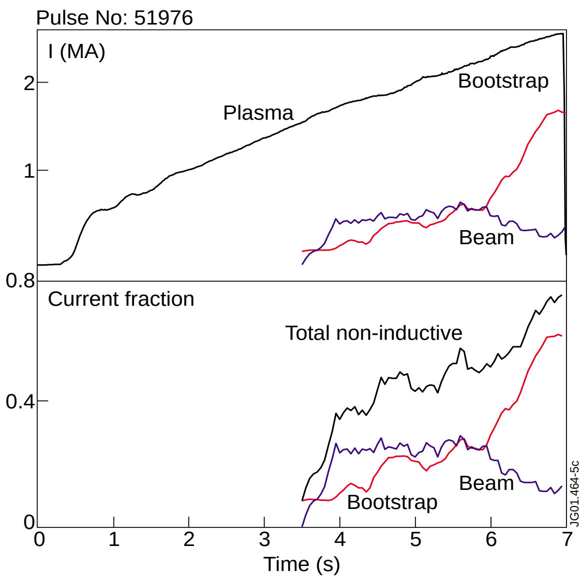


Figure 3.8: The build up of plasma current in Pulse No. 51976. The beam power, 17MW, and ICRH power, 4MW, are constant after t=5s. As a high performance internal transport barrier develops the plasma pressure increases rapidly and the non-inductively driven bootstrap current increases. The total non-inductively driven current approaches 80% of the total current before the discharge disrupts. The beam driven current reduces as the plasma density increases.

avoiding instabilities. For example, instabilities may appear when confinement improves and the pressure gradient increases. In this case the instabilities can be avoided by lowering the heating power. Furthermore, current diffusion across the barrier can lead to the disappearance of the barrier or a reduction in confinement. In this case the re-establishment of the barrier may require a reduction of the current inside the barrier.

3.3.2 Detailed Scientific Results

During Campaigns C1 to C4, the understanding of the conditions for forming and maintaining internal transport barriers was advanced by the improved measurement and control techniques for the plasma current and rotation profiles.

Thresholds for Internal Transport Barriers

Experiments were carried out to compare the performance and threshold between plasmas with nearly flat central current profiles, low (magnetic) shear, and reversed (magnetic) shear with (strongly) hollow current profiles (Fig 3.9). In contrast to the low shear plasmas, very low (or even no) power threshold was needed to establish barriers in reversed shear plasmas. Similar levels of high performance were achieved in both low and reversed shear plasmas.

Location of Internal Transport Barriers

With low magnetic shear, with nearly flat q-profile in the centre of the plasma, the internal transport barriers are formed near integer values of the safety factor, q, i. e. a magnetic surface for which the magnetic field lines form a closed curve after a finite number of turns around the torus given by the integer value of q. The importance of the location of the barrier with respect to the integer value of q is illustrated in Fig. 3.10, which shows the evolution of the ion temperature profile during the formation of the ion transport barrier. Barriers formed at integer q-values of 1, 2 and 3 have been achieved in JET.

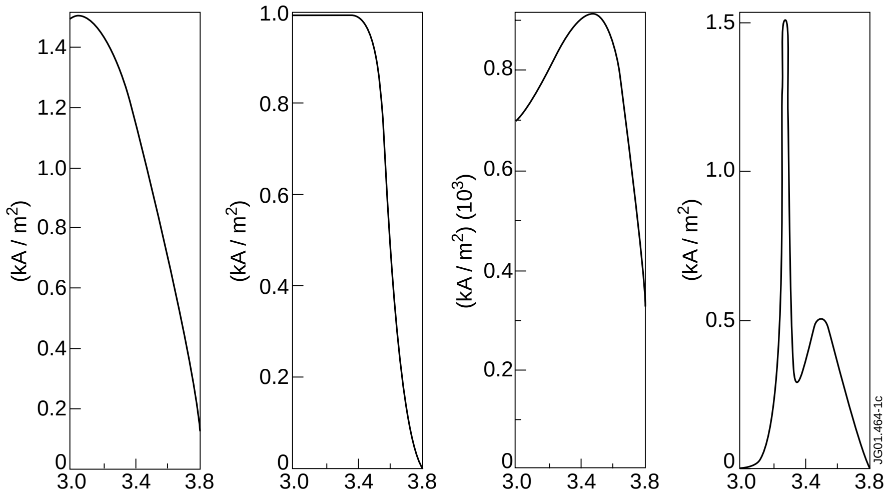


Figure 3.9: Current profiles for a normal discharge, a low shear discharge, strongly reversed shear profile and from a discharge with a current hole. The high current peak in the figure furthest to the right is caused by LHCD and the lower peak arises due to diffusion of the inductively driven current similar to the second figure from the right.

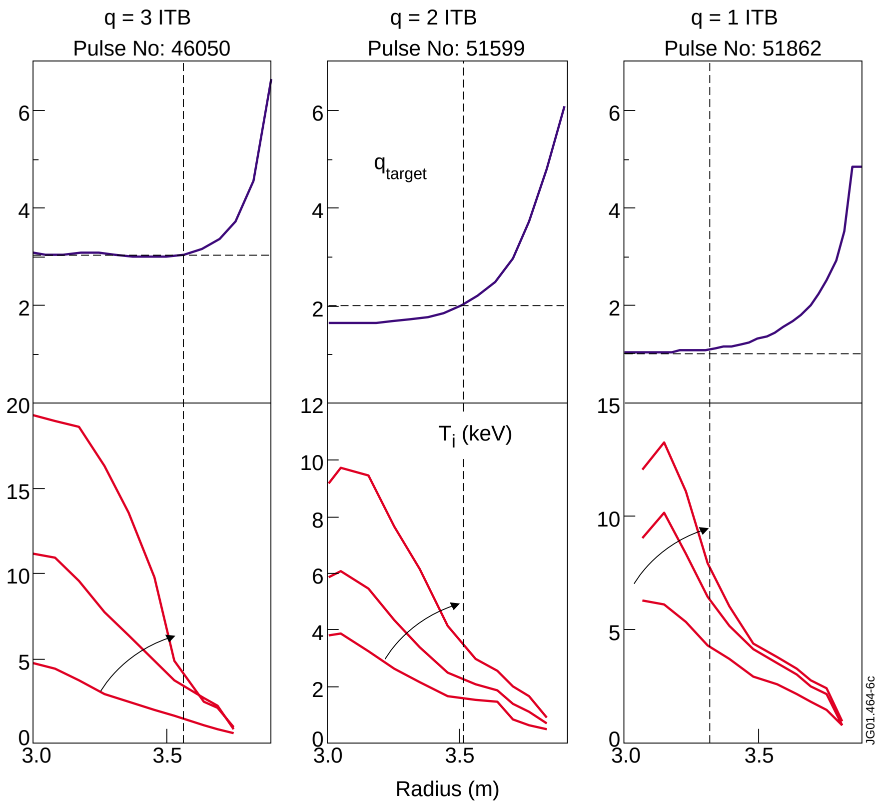


Figure 3.10: The q-profile of the plasma (upper curve) and the evolution of the ion temperature profiles during the formation of the barriers (lower curves). Note that Pulse No. 46050 was obtained under the JET Joint Undertaking.

Strongly Reversed Shear with Lower Hybrid Heating & Current Drive

Strongly reversed shear plasmas were created by first pre-heating the plasma with Lower Hybrid, (LH), waves and then rapidly ramping up the inductively driven current. The advantage of using LH waves for pre-heating the electrons is that the electrons absorbing the wave produce a current. This method is called Lower Hybrid Current Drive, (LHCD). In JET, LHCD is obtained by launching a directed spectrum of microwaves with a frequency of 3.7GHz. Both the pre-heating of the electrons to high temperatures and the current drive were found to be important for establishing hollow current profiles as targets for reversed shear plasmas.

The currents generated in the barrier by LHCD, neutral beams and the bootstrap effect can reduce strongly the inductively driven current in the centre. In the extreme case a large current deficit, the current hole, *(see Fig. 3.31 in Section 3.7)*, is formed near the centre as deduced from the measurements of the inclination of the magnetic field using the Motional Stark Effect, MSE.

The bootstrap current in the barrier can constitute a significant fraction of the total plasma current because of the large pressure gradient and the relatively low poloidal magnetic field in that region (the bootstrap current is proportional to the density and temperature gradients and inversely proportional to the poloidal magnetic field).

Long Pulse Non-inductive Operation

Long pulses with approximately 80% non-inductively driven current have been achieved. 40% of the total plasma current of 2MA consists of bootstrap current and another 40% of the current is driven by neutral beams and lower hybrid waves. Fig 3.11 illustrates a pulse with these features. The coupling of microwaves for the LHCD during the main heating phase was facilitated by injecting CD_4, i.e. methane containing the isotope deuterium instead of the normal hydrogen isotope, in front of the mouth of the microwave launcher.

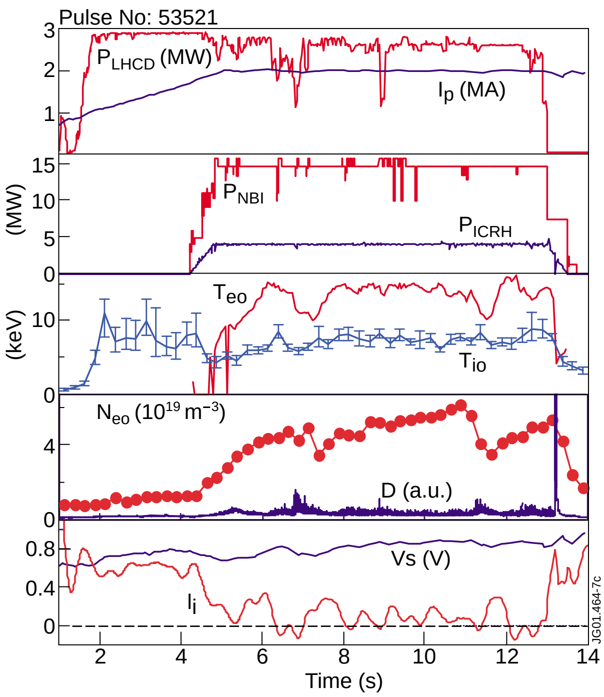


Figure 3.11: The evolution of some parameters of Pulse No. 53521. As the barrier is formed at t=4.5s the fraction of non-inductively driven current increases and the loop voltage driving the inductively induced current falls from 0.6V to 0.1V. High density and temperature are maintained during 8s, which equals 27 energy confinement times.

Impurity Accumulation

Long pulse discharges with internal transport barriers allowed detailed studies of particle transport, confirming its neo-classical characteristic of leading to the accumulation of high-Z impurities in the plasma core in cases with strong density barriers (Fig. 3.12).

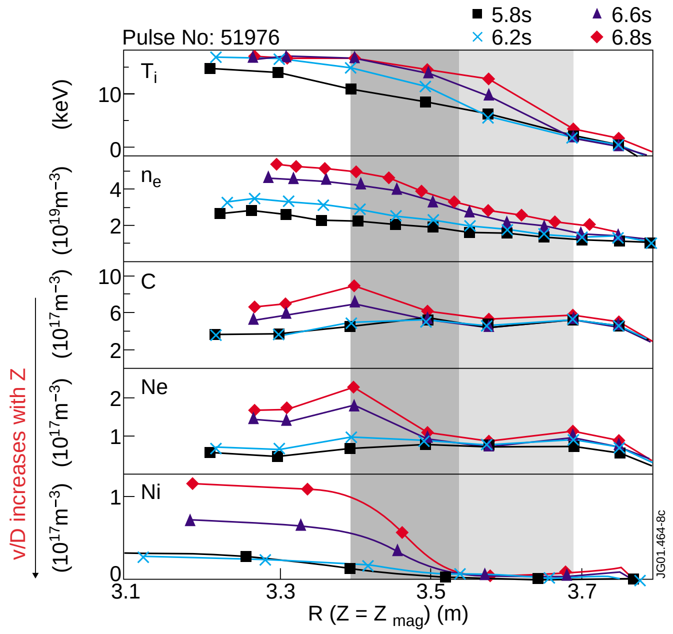


Figure 3.12: Impurity accumulation is consistent with neo-classical theory. As the density profile peaks up the profile of the heavy impurities such as nickel peaks more than that of the light impurities such as carbon.

Real Time Control of Internal Transport Barriers

Significant progress in the real time control of localised internal transport barriers was achieved by adjusting the heating power to the normalized temperature gradient length as the barrier develops. A criterion for the existence of a transport barrier was established and tested allowing the position and strength of the barrier to be identified in real time. This was used successfully in the real time control of the internal transport barrier (Fig. 3.13).

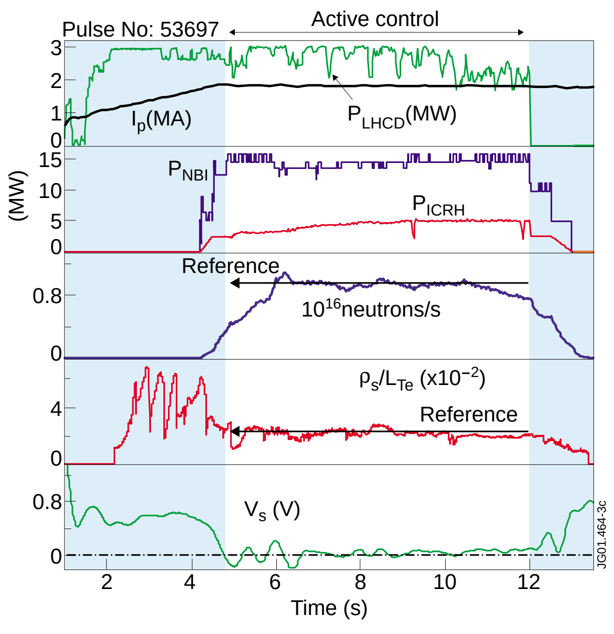


Figure 3.13: Time evolution of Pulse No 53697 using feedback control of the neutron rate by NBI and the normalized electron-temperature gradient length L_{Te}/ρ_S by ICRH. The plasma current is sustained fully non-inductively (V_s = 0).

ELM Behaviour in Discharges with Internal Transport Barriers

The importance of the behaviour of plasma in the outer confining region was confirmed, as large edge localised modes (ELMs) often degraded severely the performance and even led to the termination of the discharge. However, in highly non-inductive discharges stationary mild ELM activity was observed.

Internal Transport Barriers with Similar Electron & Ion Temperatures

In most high performance discharges in JET the ion temperature is much larger than the electron temperature, but in reactor plasmas heated by alpha particles the electron and ion temperatures would be comparable. Experiments with internal transport barriers under such conditions have also been carried out on JET.

Turbulence Suppression

Formation of transport barriers is assumed to be caused by the destruction of the drift wave turbulence causing the anomalous transport. With improved measurement techniques for the plasma current (MSE) and rotation profiles (NBI "blips") it was found that in JET the suppression of the turbulence is consistent with stabilisation via ExB-shear, when the effect of the magnetic shear is included. This is illustrated in Fig. 3.14.

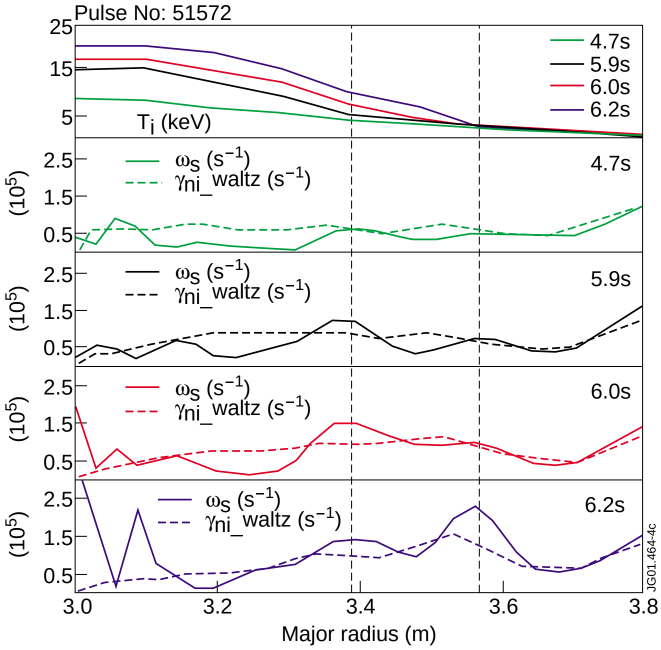


Figure 3.14: The ExB shear flow suppression of ITG driven electron turbulence seems to be confirmed in JET. Drift modes are expected to be stabilised when the ExB shear flow exceeds the growth rate of the drift mode including the effect of magnetic shear.

3.4 Task Force E

3.4.1 Scientific Issues

The major goal of Task Force E is to predict for ITER the levels of power and particle exhaust for the scenarios of Task Forces S1 and S2 and to ensure that they are compatible with the technical specifications of ITER plasma facing components. In this context areas of interest are erosion of plasma facing components, migration of wall material and subsequent re-deposition, fuel recycling, impurity production and transport to the core plasma, and retention of hydrogen in carbon. A major issue for future fusion power generating plants is the retention of tritium within the machine, since this reduces the efficiency of the recycling process (leading to costs for extra tritium) and introduces additional safety constraints.

Task Force E also works closely with Task Forces S1 and M in the process of improving the understanding and scaling of ELMs but with special emphasis on the impact of ELMs on the divertor and wall surfaces. In collaboration with Task Force M both the impact of disruptions on plasma facing components and techniques for their mitigation are developed.

Besides these experimental activities Task Force E develops simulation packages to model the edge and divertor plasmas with improved predictive capability.

3.4.2 Summary of Results

The exhaust of power from a tokamak plasma is one of the key constraints on the design of a fusion reactor. As a result Task Force E has characterised the edge, Scrape-Off Layer (SOL) and divertor plasmas for the discharge scenarios of Task Forces S1 and S2. Power exhaust properties have been studied using improved techniques such as slow sweeps of the plasma footprint across the divertor target plates, together with finite element analysis of thermocouple data and improved diagnostics of the edge (edge Lidar and Li-beam) and of divertor plasmas (IR thermography). In H-mode discharges a narrow power carrying SOL layer with a midplane decay length of 2-3 mm has been found.

Carbon is the principal impurity which is released from the walls of JET by physical and chemical sputtering. These atoms are ionised in the SOL and flow back to the divertor by collisional coupling with the deuterium ions. This “screening” of impurity atoms by the SOL has been quantified by injecting methane from various locations and observing the contamination of the plasma core. The screening is twice as effective for methane puffed from the vessel top than from the midplane and 10 times more effective when puffed from the divertor.

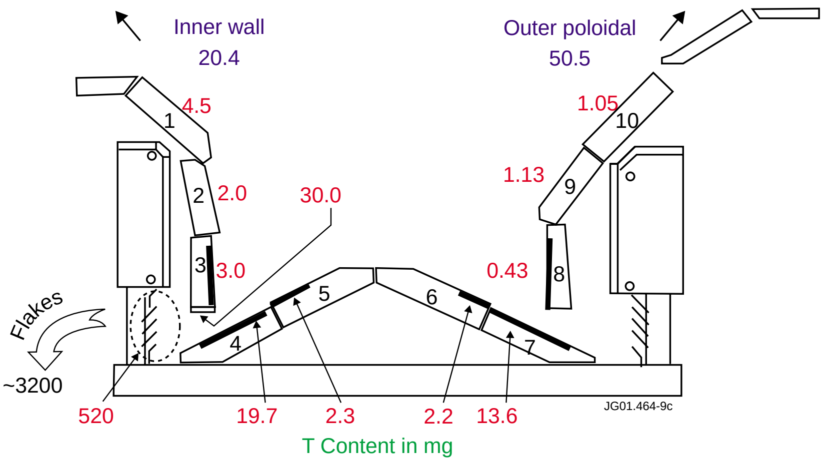


Figure 3.15: Poloidal cross section of the JET divertor with the content of tritium (in mg) in the divertor tiles. Most of the remaining tritium is believed to be located in the sub divertor region (3.2 g). The wall is a net erosion area.

Mixtures of helium and deuterium and pure helium have been studied mainly to improve the understanding of the physics of impurity release (chemical erosion versus physical sputtering) and to assess the suitability of helium operation during the low activation phase which would precede the introduction of DT into ITER. Due to the non-existence of chemical erosion processes with pure helium plasmas a significant decrease of the carbon impurity release was observed compared to similar deuterium plasmas. The power threshold for H-modes is about 50% higher in helium compared to deuterium. Energy confinement decreases by 15-20%

compared with similar deuterium plasmas. Power deposition profiles on the divertor plates are broader by 30-80% in helium plasmas.

Surface analysis of divertor tiles revealed that after operation with deuterium-tritium mixtures in previous JET campaigns, most of the tritium trapped in the vessel was in carbon-based deposits that formed flakes. The retention of tritium in JET is primarily associated with the re-deposition of carbon eroded from contact points between the plasma and the surrounding walls (Fig. 3.15)

3.4.3 Detailed Scientific Achievements

Power to the Divertor, Decay Lengths

Under normal operating conditions about three quarters of the energy entering the SOL reaches the outer divertor. Analysis of thermocouple and target probe data shows that in lower density H-mode discharges the power profile on the outer target consists of a narrow peak corresponding to a midplane decay length of 2-3 mm which carries about 70% of the ELM-averaged power and a wider profile wing (5-7 mm midplane) (Fig. 3.16). The contribution of the narrow peak decreases with increasing density, whereas the wider part does not change. No narrow peak is observed on the inner target. Modelling indicates that the most likely cause of the narrow power peak is the loss of energetic ions whose orbits cross the magnetic separatrix. Based on this result an unfavourably narrow power width is predicted for ITER, but this is probably ameliorated by a higher neutral density expected in the ITER divertor to suppress direct orbit losses due to charge exchange processes. However, it is clear that due to the low plasma collisionality in the mid-plane, ion orbit losses will remain important in the ITER SOL and divertor behaviour.

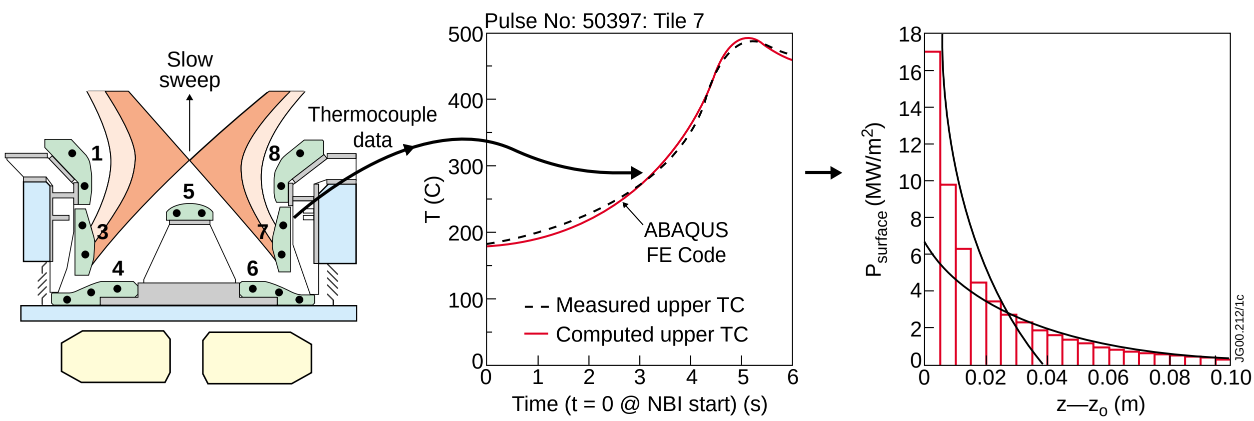


Figure 3.16: Average power profiles extracted from thermocouple data.

Steady state and transient power loads on the targets have been investigated in parallel with the fast infrared (IR) thermography system (Fig. 3.17).

In the outer divertor, the power balance agreement between IR and thermocouple data is excellent whereas the IR analysis indicates a much larger power on the inner target compared with thermocouple and other data. This is most probably caused by "overheating" of surface layers which originate from carbon impurity deposition and have reduced thermal properties. Type I ELMs carry typically about 30% of the power to the outer target but a much larger proportion (> 50%) to the inner. The ELM power deposition profile is only slightly larger (20-50%) compared with that found in between ELMs and with no significant or only a very slight shift of the position of deposition maximum (< 5mm).

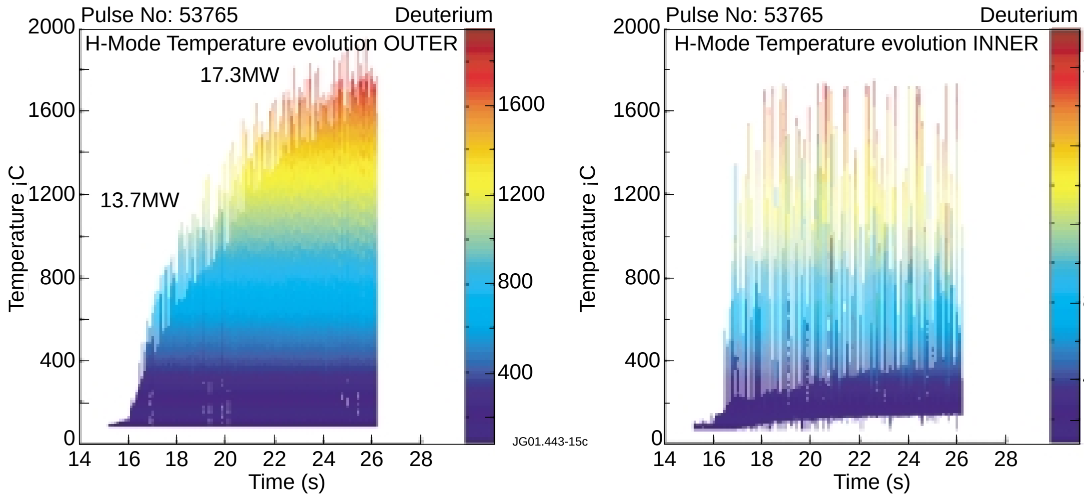


Figure 3.17: Temperature evolution of the outer (left fig.) and inner (right fig.) target plates in a ELMy H-mode discharge.

Carbon Impurity Production

a) Effect of Lower Vessel Temperature

The reduction of the temperature of the JET vessel from 320°C to 200°C reduced the temperature of the divertor tiles from about 220°C to 140°C. As a result the chemical erosion yields and the carbon ion line emission decreased by about 50% in the inner divertor but did not change in the outer divertor. With reduced vessel temperature the density at which an X-point MARFE forms is increased by 20% in L-mode discharges. However, no beneficial effect on the main chamber carbon impurity content was measured.

b) Operation in Helium

JET operated with mixtures of deuterium and helium as well as pure helium plasmas in nearly all operational scenarios which had been exploited previously in pure deuterium. Figure. 3.18 compares in helium and deuterium discharges the CIII light emission integrated over the inner and outer divertor legs and along a midplane horizontal chord from (a) two similar L-mode density limit discharges and (b) two 11MW NB heated H-mode discharges at the same density. The inner divertor carbon source decreases by factors 3-10 under all plasma conditions. In the outer divertor the carbon release is similar for helium and deuterium operation at lower densities but, in contrast to deuterium operation, increases with rising density in helium operation. In the main chamber similar CIII light emission (Fig. 3.18) is observed for the limiter phases of both helium and deuterium plasmas whereas a significant reduction occurs in the diverted phase of the helium discharge. This reduction is mainly due to the reduced charge exchange fluxes.

This comparison of helium with deuterium operation proves that in the inner divertor chemical erosion dominates the carbon release in deuterium discharges. In the outer divertor, however, physical sputtering dominates at low densities with an increasing contribution from chemical erosion at higher densities.

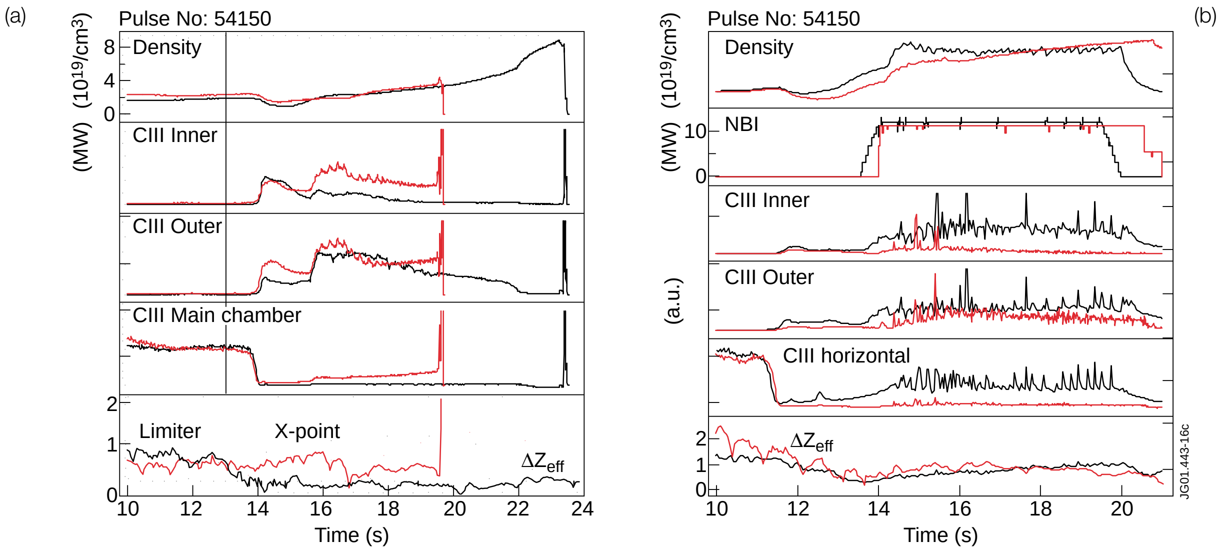


Figure 3.18: Evolution of the density, CIII light in the outer and inner divertor as well as in the main chamber together with Z_{eff} for L-mode density limit discharges in He and D (left). Similar time traces for ELMy H-mode discharges in He and D at 11 MW NB heating power (right). (D: red trace; He: black trace)

In L-mode discharges the reduced carbon influx leads to a considerable reduction in the main plasma carbon content (Fig. 3.18). In H-mode discharges the carbon source reduction is similar but does not manifest itself in a reduced Z_{eff} (Fig. 3.18, right part). This surprising result points to a dominant contribution of the outer divertor to the core-plasma carbon content during H-mode.

The largely reduced carbon content results in L-mode density limits which are about a factor of two above those observed in deuterium-plasmas (Fig. 3.18) with the density limit determined by an almost poloidally symmetric radiative collapse as compared with a much more localised radiative collapse (MARFE) in deuterium operation.

3.5 Task Force M

Task force M (MHD) aims to study the stability of the plasma to low and high frequency disturbances and related phenomena. The objective is to understand, control, avoid or mitigate the consequences of such instabilities. The basic model for understanding is the so-called MagnetoHydroDynamic (MHD) model, which describes a large variety of plasma phenomena, including the most common instabilities that can lead to plasma disruptions and the degradation of confinement. The control of such instabilities improves discharge reliability and performance, extending the operational domain of the tokamak. The validated models are used in the extrapolation to ITER parameters and the development of reactor scenarios.

In the ELMy H-mode scenario, the ITER reference operational scenario, the occurrence of **"Neoclassical Tearing Modes" (NTMs)** and **Edge Localized Modes (ELMs)** can limit the maximum plasma pressure achievable. Near operational boundaries, and in particular in advanced tokamak regimes, **disruptions** can occur. It is important to improve the understanding and predictive capability in this area and develop techniques to mitigate the consequences of **Vertical Displacement Events (VDEs)** and the generation of **runaway electron** beams through experiments, modelling and data base comparisons.

3.5.1 Scientific Issues

Disruptions and Runaway Electron Beams

The abrupt termination of the plasma current due to an MHD instability is normally referred to as a plasma disruption. This process deposits the entire energy content of the plasma onto the first wall in a very short period of time causing the erosion of the plasma facing materials. The forces resulting from the currents induced in the vessel components can limit the maximum plasma current in a given plasma configuration. In addition, the strong electric field associated with the abrupt termination of the plasma current can generate beams of runaway electrons. These high energy electron beams may intercept and damage the first wall. It is important to improve understanding and the predictive capability, including the development of techniques to avoid and mitigate the consequences of disruptions.

Neo-classical Tearing Modes

In the ideal MHD framework the magnetic field lines are frozen in the plasma. However, finite dissipation allows the magnetic field lines to diffuse. In the vicinity of magnetic surfaces with rational values of the safety factor q, the topology of the magnetic field lines can change and generate disturbances known as tearing modes. In the tokamak self generated currents, known as bootstrap currents, arise from the plasma pressure. Asymmetric plasma disturbances lead to changes to the bootstrap current, which can reinforce the disturbance, giving rise to an instability known as the Neoclassical Tearing Mode (NTM). The need for an initial 'seed' disturbance is a key aspect of the NTM and so understanding the physics of the seed mechanism and its relationship to the plasma pressure is an important issue for the future. The most common seed mechanism is linked to the periodic plasma relaxations of the central plasma pressure, known as the "sawtooth" instability. The sawtooth crash is caused by the internal kink MHD instability associated with the safety factor on axis being below unity.

Error Field Modes

The tokamak is designed as a toroidally symmetric magnetic confinement chamber. However, due to technical limitations, toroidally asymmetric fields are unavoidable and are known as error fields. These asymmetric fields can lead to instabilities known as error field modes. Such modes can limit tokamak operation at low density and it is necessary that ITER has sufficiently small error fields, to avoid instability. It is important to understand the critical values of the error fields and the requirements on coil systems which can compensate for the error fields.

Alfvén Eigenmodes

The presence of a population of energetic ions, which are sufficiently energetic for their velocity to be above that of the Alfvén waves, can interact with and destabilize Alfvén eigenmodes (AEs). Such unstable AEs could lead to anomalous transport of alpha particles in a fusion tokamak reactor. Alpha particle losses, if they occur, may reduce the ignition margin and can cause damage to the first wall of the reactor. Present models predict (AE) stability in the conventional ELMy H-modes reactor scenarios. However, AEs are thought to be more easily destabilized in advanced tokamak regimes, in particular, in scenarios with strongly reversed profiles of the safety factor. Therefore, present efforts concentrate on studying AE physics in these scenarios and validating the physics for extrapolation to reactor scale.

3.5.2 Detailed Scientific Results

Physics of Low Frequency Modes

Neo-classical Tearing Modes

The focus was the understanding of the main plasma parameters contributing to the physics of low frequency MHD modes. This understanding allowed the control of these modes, in particular Neoclassical Tearing Modes (NTMs), and the improvement of plasma performance.

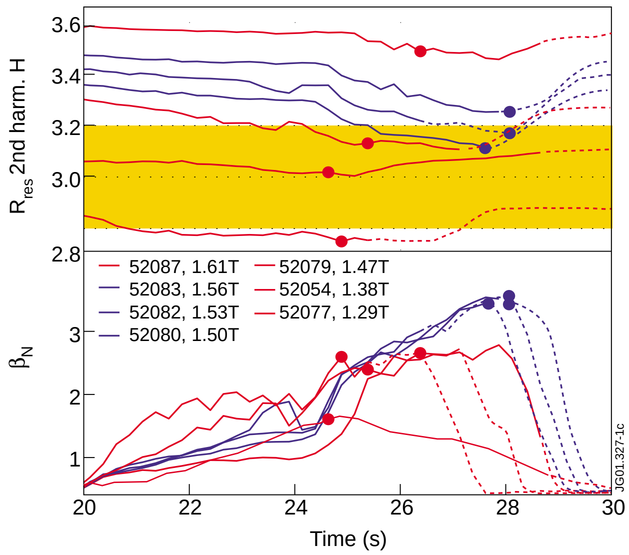


Figure 3.19: Evolution of the ion cyclotron heating resonance position and normalised plasma pressure (β_N) for discharges with the magnetic field (B_0) changed from pulse to pulse. The dots mark the onset of NTMs and the shaded area is the region inside the Sawteeth inversion radius. By applying the ICRH at the optimal position for Sawteeth stabilisation the onset of the NTM is delayed and the overall normalised pressure is increased.

Power ramp down experiments showed that the NTMs are meta unstable for values of the normalised plasma pressure which characterize ELMy H-mode operation in JET. Therefore, the emphasis was on controlling the cause of the NTMs rather than ameliorating their effect. NTMs are caused by the seed islands created by events such as sawtooth crashes leading to the onset of NTMs. Since the sawteeth period and amplitude can be affected by a population of fast ions (as might be created during ICRF heating), detailed experiments were performed, in collaboration with Task Force H, on the effect of the first and second harmonic hydrogen minority heating and current drive on the sawteeth. It was shown clearly that at crashes after long sawtooth periods (large amplitude sawteeth), larger seed islands are created resulting in an onset of NTMs at much lower values of the plasma pressure. In contrast, by destabilising the sawteeth, thereby reducing the sawtooth period (and the amplitude), it has been possible to obtain much higher plasma pressure (see fig 3.19). The role of plasma rotation has also been studied, by changing the ratio of ICRF and NB powers. It has been shown that increased rotation is beneficial, most probably due to the decoupling between the magnetic surfaces with q=1 and q=1.5.

The effects of ICRF on sawteeth has also been used to improve specific scenarios, such as the Radiation Improved (RI) mode or density peaking experiments. Longer wavelength (n=1) NTMs have also been studied and a joint experiment with the US tokamak DIII-D has been conducted. These experiments on JET and DIII-D have established that the pressure limits set by NTMs with a toroidal mode number n=1 are similar for the same dimensionless parameters in the two machines. This has allowed scalings for the NTM threshold to be established.

Error Field Modes

Dedicated experiments have studied the dependence of the threshold for the penetration of error fields. These have allowed the previous scaling for ITER for the error field threshold to be confirmed. Furthermore, a possible reason for the different scalings observed on JET and on the smaller tokamak COMPASS-D may be due to, at least in part, the different dependence of plasma rotation on magnetic field.

Physics of High Frequency Modes and Energetic Particles

Furthermore, specific experiments dedicated to actively measuring the damping of Toroidal Alfvén Eigenmodes (TAEs) with low toroidal mode numbers (n=0,1,2,..) in limiter discharges and to the study of the excitation of TAEs using ICRH minority ions have been performed (Fig 3.20). These will enable a better test of the models used to predict TAEs in ITER. In particular a very weak dependence of damping on magnetic field has been demonstrated and weakly damped Alfvén eigenmodes in X-point scenarios with reversed magnetic shear have been discovered.

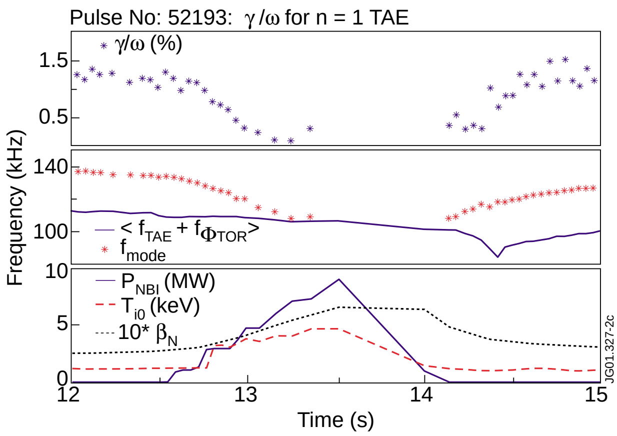


Figure 3.20: Evolution of the frequency and damping of a TAE measured by the saddle coil antenna active excitation system and other main plasma parameters. The decrease in damping as a function of the normalised pressure (β_N) is clearly shown.

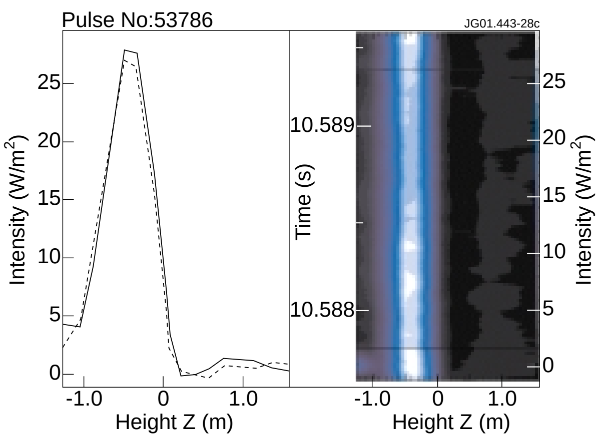


Figure 3.21: Soft X-ray profile and image of the runaway electron beam measured with a shielded soft X-ray camera. This type of data allows the size, position and time behaviour of the runaway to be determined, and in some instances more detailed properties of the magnetic geometry can be inferred.

Initial experiments on the formation of runaway electron beams after plasma disruptions were carried out (Fig.3.21). Optimal conditions for the formation of such electron beams were established consistent with the studies carried out in other tokamaks. These experiments will allow more detailed studies in the future, including methods of mitigating or avoiding runaways.

3.6 Task Force H

3.6.1 Scientific and Technical Issues

Task Force H addresses the physics and engineering aspects of plasma Heating and Current Drive, for which three systems are available on JET: Neutral Beam Injection, Ion Cyclotron Resonance Frequency and Lower Hybrid Frequency.

Neutral Beam Injection

High energy neutral beams, created in a neutralizer from a source of accelerated ions, are injected into the plasma where they are ionized and transfer their energy and momentum to the bulk plasma ions and electrons by collisions. Neutral Beam Injection (NBI) is intensively used in high power experiments as a source of heating, current drive and particles.

Ion Cyclotron Resonance Frequency Heating

Ion Cyclotron Resonance Frequency Heating (ICRF) is a highly versatile heating and current drive technique, relying on antenna arrays to launch radio frequency waves (23 to 57 MHz) into the plasma. These waves are resonantly absorbed by selected particle populations, which eventually transfer their energy and momentum to the bulk plasma by collisions. ICRF can deliver power predominantly either to ions or to electrons, at the centre of the plasma core or off-axis, over a broad range of plasma densities and equilibrium magnetic fields. By suitable phasing of its four current-carrying conductors, each of the four antennae can deliver a variety of radiation patterns: symmetric power spectra are used for plasma heating, whilst highly directive asymmetric spectra can also be applied in non-inductive current drive experiments.

Lower Hybrid Heating and Current Drive

Lower Hybrid Heating and Current Drive systems employ a waveguide array to launch high frequency waves (3.7 GHz) into the plasma, which are subsequently absorbed by the electrons. It is a highly efficient technique to drive off-axis current in the plasma, and thus a powerful tool to control the current profile.
The ability to couple power with ICRF and LHCD depends strongly on the plasma boundary characteristics: distance to the antennae, density, Scrape-Off Layer plasma. The most important challenge faced at present is to reliably couple high levels of power to the **high performance** ELMy H-mode discharges relevant to ITER, characterized by steep edge gradients and unsteady edge conditions.

The scientific and technical work of Task Force H in Campaigns C1-C4 has been organised along three topics:

- Plasma Heating, Current Drive and Rotation physics,
- Launching and deposition issues, and
- Optimisation of the systems.

Apart from its specific activities, Task Force H collaborates intensively with all other Task Forces, and in particular with Task Force S2 in the development of advanced scenarios with a significant fraction of non-inductive current drive.

3.6.2 Scientific and Technical Achievements

Plasma Heating, Current Drive and Rotation Physics

Physics aspects of plasma heating, current drive and rotation have been addressed, such as the capability of Ion Cyclotron (IC) waves to drive the plasma current and to induce toroidal rotation of the plasma, and ICRF heating scenarios in ^4He plasmas, for the phase of ITER operation preceding the use of deuterium/tritium mixtures.

Launching and Deposition Issues

The physics of coupling the power to the relevant plasma scenarios as well as determining and understanding the location of the power deposition has been studied. In particular the problem of coupling Lower Hybrid power to plasmas which exhibit both internal and external transport barriers has been investigated.

Optimisation of the Systems

Experiments which explore the limitations of the heating systems have been carried out, in order to develop tools to improve their performance. Examples are:

- Commissioning a wideband matching system for ICRF and
- Implementing a protection system for the LH launcher based on the monitoring of heavy impurity radiation.

3.6.3 Detailed Scientific and Technical Results

(i) Ion Cyclotron Current Drive Experiments

Plasma current has been driven by various schemes based on ICRF heating and measured by a diagnostic which is based on the Motional Stark Effect (MSE (see Fig 3.22)). In these experiments the most commonly employed phasings of the antenna arrays correspond to successive currents in phase quadrature, hence the terminology +90° and –90° phasings, which respectively correspond to waves preferentially launched along or opposite the plasma current.

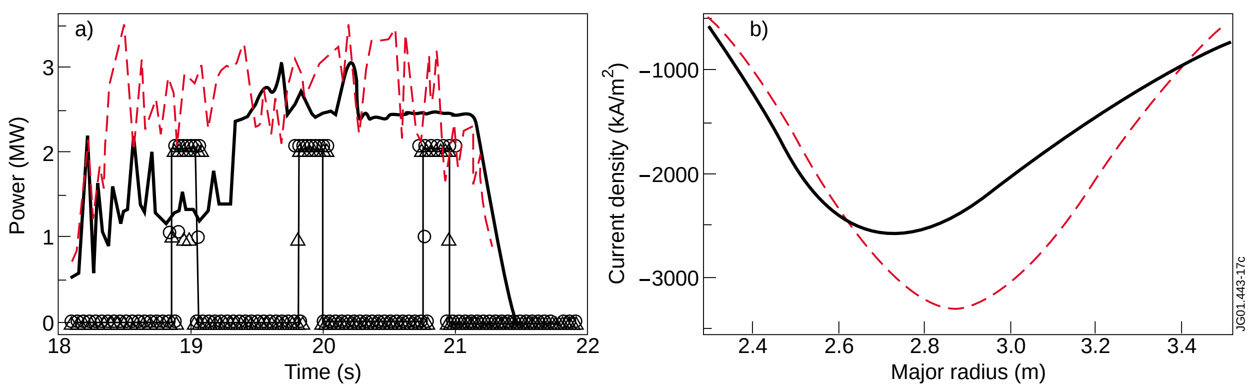


Figure 3.22: a) Injected power: bold line ICRF +90°, dashed line ICRF -90°, circle/triangle NBI. b) Current density profile at t = 20.9s measured by MSE, bold line ICRF +90°, dashed line ICRF -90°.

Sawteeth Control Experiments

In the case of a minority concentration of hydrogen in deuterium plasmas, an internal relaxation phenomenon known as "sawteeth" can be stabilised with +90^o phasing of the antennae, leading to long-period "monster" sawteeth. With the opposite -90^o phasing the resulting "destabilised" sawteeth have shorter periods and smaller amplitudes.

Further experiments have been carried out by Task Force M to control sawteeth and hence prevent the formation of seed islands for Neoclassical Tearing Modes at the q=3/2 surface. This allows the threshold pressure to be increased, thus leading to performance improvement.

Heating Minority Hydrogen Ions at the Second Harmonic of their Cyclotron Frequency

In deuterium plasmas, with low concentrations of Hydrogen (1-2 %), clear evidence of heating at the second cyclotron harmonic of hydrogen (B_0=1.7T, f_{ICRF}=56MHz) was found with Neutral Particle Analysers; no fast deuterons due to heating at the fourth harmonic of deuterium were detected.

Fast Wave Current Drive Experiments

For the first time on JET, moderate levels of RF power (2-3 MW, see fig 3.22a) have been coupled at high toroidal magnetic fields in the so-called Fast Wave Current Drive scenario, where the wave energy is directly absorbed by electrons in the absence of significant cyclotron damping by the ions (B_0=3.45T, f_{ICRF}=37MHz, I_p=2.5MA). MSE data show a difference in current density profile at the plasma centre between +90^o and -90^o (Fig. 3.22b), with a difference in central current of 300kA.

(ii) Plasma Rotation induced by Ion Cyclotron Heating

A systematic study has been made of ICRF-induced toroidal plasma rotation, and in particular of its radial profile, for a wide range of parameters (L/H mode, position of resonance layer, antenna spectrum). Since the experiments rely on short pulses of neutral beam injection (blips) to measure rotation from the Doppler shift of the resonant charge exchange line of carbon impurities, it is of utmost importance to check that the measured rotation is not affected by the measurement technique itself. The overall rotation profiles start to evolve by the end of the blip which lasts 200 ms. The value of the central rotation is affected more quickly (on order of 50 ms). The very first profile however, taken during the short beam blip, can be seen as representative of the rotation profile before the blip. This was independently confirmed by the time evolution, due to the blip, of mild central MHD modes. In L-mode plasmas heated by ICRF alone (hydrogen minority in deuterium), the rotation shows a distinct maximum off-axis and in the direction of the plasma current. The rotation depends on the position of the resonance layer: a resonance layer located on the high field side of the magnetic axis leads to a slightly higher maximum than for a low field side location. The difference between a co- and counter- direction of the antenna spectrum is small.

(iii) Ion Cyclotron Heating Experiments

Minority and Mode Conversion Heating

An extensive series of experiments on ICRF heating scenarios for the phase of ITER operation prior to the introduction of DT mixtures has been carried out in ^4He plasmas. Scenarios based on the use of ^3He were studied systematically by scanning the ^3He concentration from low (the minority ^3He regime) up to high values (the mode conversion regime). At low concentration, the first direct observation of an ICRF-induced pinch of ^3He minority ions was made with a gamma ray emission profile monitor (Fig. 3.23). At high concentration, the mode conversion regime was explored in detail for the first time on JET. In this heating scenario, the decimetric

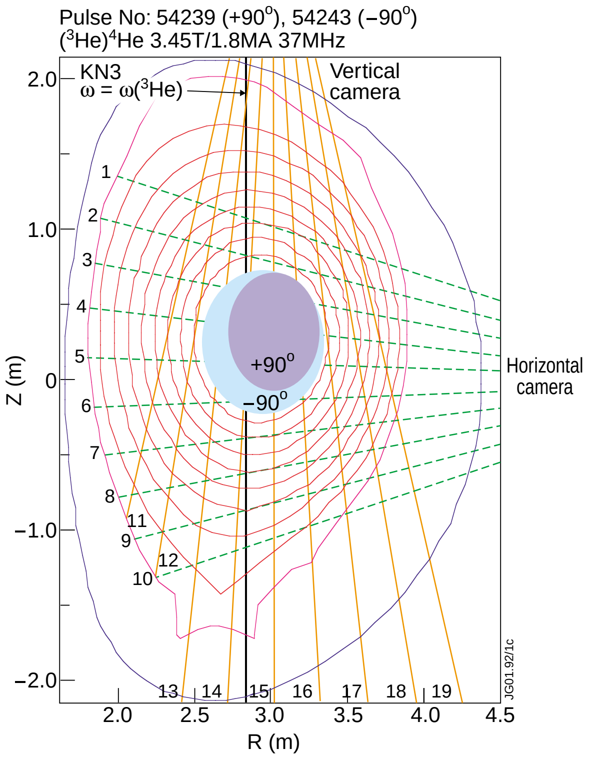


Figure 3.23 :Radial extent of gamma ray emission at half maximum monitored for two discharges, one with +90° and the other with −90° ICRF antenna phasing.

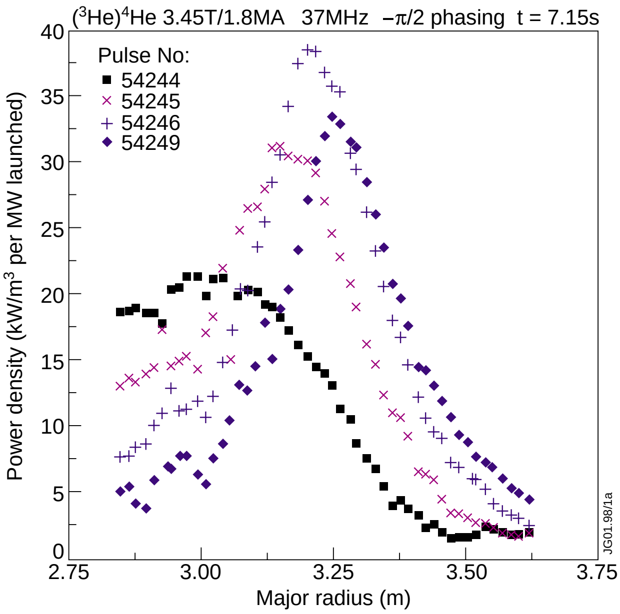


Figure 3.24: With increasing ^3He puff, transition from central to off-axis direct electron power deposition is observed from break-in-slope analysis of T_e data during ICRF power modulation.

waves launched into the plasma by the antennae can strongly excite a much shorter wave mode, itself strongly absorbed by the electrons. Control of the off-axis power deposition was demonstrated by adding ^3He during ICRF heating (Fig. 3.24).

H-Mode Threshold Experiments

The conditions for accessing the H-mode in plasmas of various isotope composition from predominantly ^4He to predominantly D were studied for several ICRF scenarios. In all cases an H-mode was triggered.

Third Harmonic Heating of Helium Neutral Beam Ions

Experiments have also been carried out for the first time in ^4He plasmas with ICRF power applied at the third harmonic of ^4He neutral beam ions, with the objective of creating a strong ^4He tail for alpha particle studies. In these experiments clear experimental evidence of the acceleration of ^4He ions up to energies in excess of 2 MeV has been obtained with a gamma ray diagnostic based on the reaction $^9Be(\alpha, n\gamma)^{12}C$ (Fig, 3.25). In the presence of a strong ^4He tail with an effective temperature around 0.5 MeV, Alfvén eigenmodes, sawtooth stabilisation and H-modes induced by ^4He heating were observed for the first time on JET.

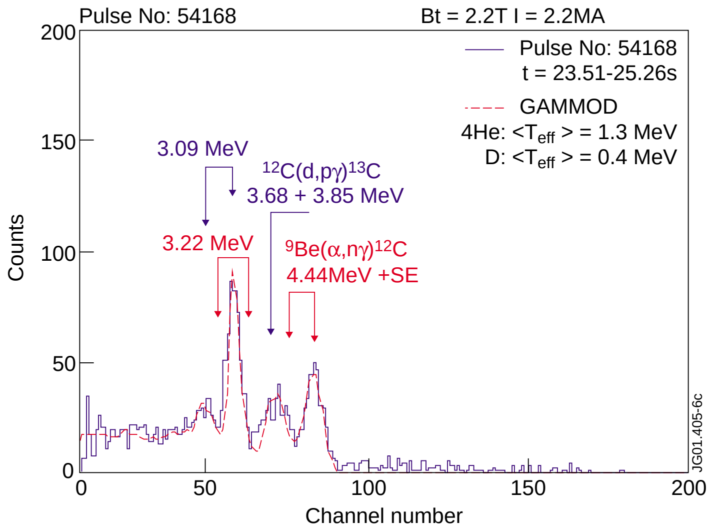


Figure 3.25: Experimental and calculated γ-ray spectra for discharge 54168

Simulation of the Self-heating of Plasmas by Alpha Particles

Interesting experiments have been performed using ICRH to simulate alpha particle self-heating. These experiments studied the dynamic behaviour of deuterium plasmas where part of the ICRH was applied in direct response to real-time plasma parameters (such as the neutron rate) using the Real Time Central Control networks. Since ICRH, via fast ions, mainly heats the electrons and is centrally deposited, this experimental arrangement simulates the plasma self-heating which alpha particles would produce in a fusion reactor. A separately controlled component (of either ICRH or NBI) was used for auxiliary heating. With these experiments several features of self-heated plasmas were demonstrated. Figure 3.26 shows an example of such an experiment where "thermal runaway" (i.e. increasing stored energy with decreasing applied power) is stabilised by decreasing the additional power in real time. Also shown is the capability to control the level of the simulated alpha particle heating power, via real time feedback, to a pre-programmed control reference waveform.

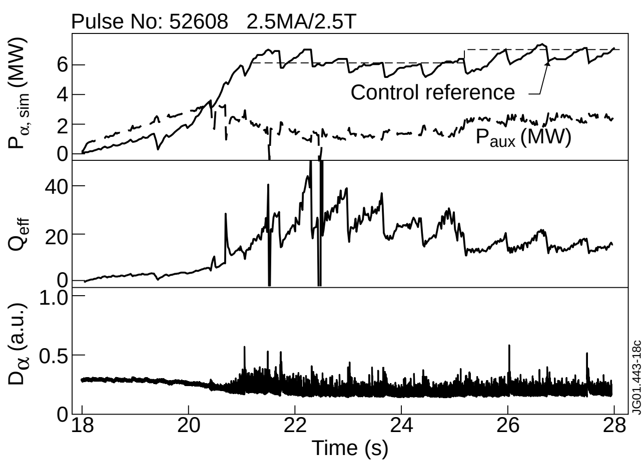


Figure 3.26: Simulated self-heated plasma experiment in which a component of ICRH is applied in proportion to the measured change in DD reaction rate (P_{α},sim). A separate component of the ICRH is used in the role of auxilliary heating (plus constant 2MW NBI), under feedback control after t = 20.5s thus stabilising the simulated "thermal runaway".

(iv) Lower Hybrid Studies

Very significant progress was achieved in coupling LH waves to high performance plasmas during Campaigns

C1-C4. A combination of proper plasma shaping and local gas injection was developed, which enabled systematic operation with lower hybrid powers in excess of 3.5 MW in H-mode discharges with ELMs and with Internal Transport Barriers.

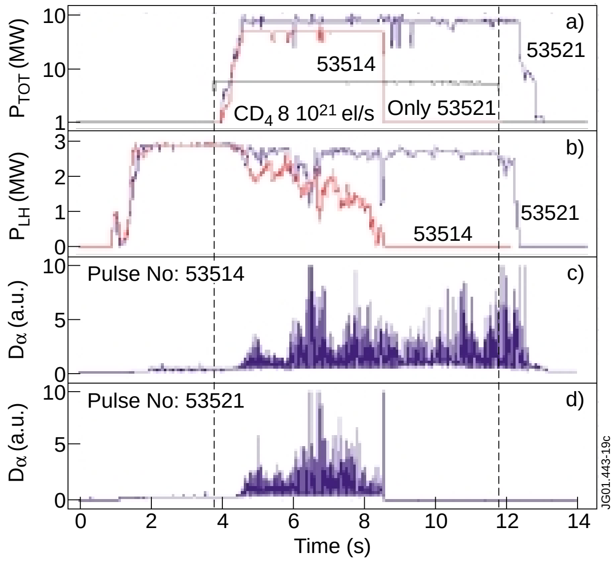


Figure 3.27: a) Time evolution of total auxiliary power: NBI plus 4MW ICRF; b) LH power; c-d) D_α in the two discharges, resp. with and without CD_4.

Deuterated methane (CD_4) is puffed into the plasma from a pipe close to the LH launcher. The injection of CD_4 increases the density locally in the Scrape-Off Layer, while best matching of the plasma and the antenna shapes produces a uniform SOL plasma along the antenna. At the optimum rate of CD_4 injection ($\approx 8\times10^{21}$ electrons/s) the power reflection coefficient decreases from about 10% to 4-5%, very close to that representative of L-mode plasmas. As shown in Fig. 3.27, the injection of CD_4 allows LH power to be coupled throughout the high power phase of the discharge. No detrimental effects on the quality of either the H-mode or the ITB mode are observed, as also suggested by the comparison of the D_α signals of Fig.3.27. As a result, the use of LHCD was increased significantly, especially in Advanced Scenarios with high additional heating power. Fig. 3.28

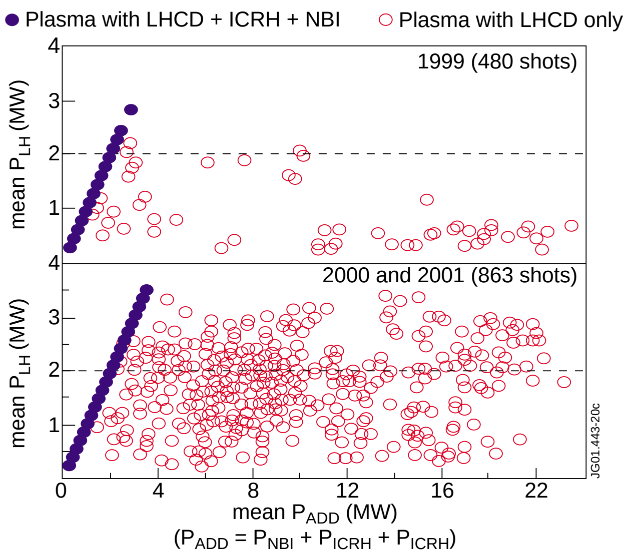


Figure 3.28: Coupled LH power versus total additinal power in 2000-01 JET Campaigns, compared with 1999 data.

illustrates the LH coupled power in the Campaigns C1-C4 versus the total power injected into the plasma and shows a comparison with the 1999 statistics. By applying LH during the early phase of the discharges, a reversed q profile is obtained which can be controlled, from weakly to deeply hollow, by varying the LH power. These LH-produced reversed shear plasmas have been extensively studied by Task Force S2 and, when heated with NBI and ICRH power, ITBs on T_e, T_i and n_e are observed with significantly lower threshold powers. When a significant fraction of the plasma current was driven by LHCD, during the main heating phase, long-lived ITBs have been obtained. In plasmas with B_T=3.4T, I_P=2.0 MA (Pulse No: 53521) ITBs lasting ~8s on the ion temperature and the electron density and ~12s on the electron temperature have been obtained.

(v) Commissioning the ICRF Wide Band Matching System

Edge-localized modes (ELMs), which are characteristic of the ITER reference scenario (the ELMy H-mode), induce large and rapid impedance variations on the ICRF antennae. The conventional radio frequency matching system cannot respond on such a short timescale, and unacceptably large bursts of reflected power occur at the generators, triggering protective power trips and limiting the ability to deliver ICRF to ELMy H-mode plasmas. A wide band matching system has been developed at JET to address this crucial problem. It consists of adjustable prematching elements inserted in series in the RF transmission lines, and of an upgraded generator frequency control allowing faster variations in the presence of ELMs (up to 300kHz in 300µs). One of the four ICRF arrays is fully fitted with a prototype system, and important progress has been achieved in its commissioning during Campaigns C1-C4, with many hardware problems being rectified. The most significant result is the demonstration of successful operation of the fast frequency feedback loop during ELMs. The successful operation of the system with low generator reflection and improved power delivery to an ELMy plasma has however not yet been established, and optimization is continuing in order to achieve the best possible performance. Given the complexity of its operation, the future of the system is currently being reviewed. Complementary measures to improve resilience of the ICRF to ELMs and alternative operating modes are also under investigation.

3.7 Task Force D

Task Force D (Diagnostics) aims to provide all Task Forces with high quality diagnostic measurements. This requires the operation and upgrade of diagnostics, control methods and systems in the specific environment of a large device capable of DT operation. To achieve these objectives, Task Force D has organised: the involvement of the Associations in the exploitation of the JET diagnostic systems during the Experimental Campaigns; experiments for the validation of particular diagnostics; and proposals for the construction of new diagnostics and upgrades, proceeding to their assessment, realisation and exploitation.

3.7.1 Organisation during Campaigns

The Workprogramme is organised through the involvement of European scientists in experiments carried out on JET. With respect to diagnostics during the Experimental Campaigns this corresponds to one third of the total number of scientists (see Fig. 3.29).

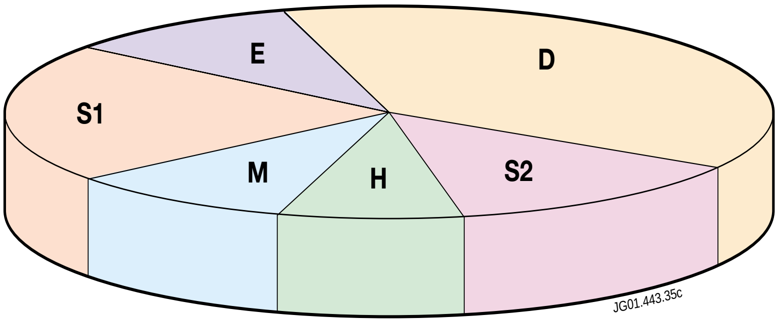


Figure 3.29: Shows the distribution of the number of scientists among the Task Forces in Campaigns C1-C4: about 33% (~10ppy in manpower) of the total number of scientists present during the Campaigns are related to diagnostics competencies, while the remainder are related to plasma physics competencies, linked to other Task Forces.

The competencies covered by Task Force D are related to the exploitation of the various diagnostic systems listed in Table 3.1. This Table shows the division of the diagnostic areas into Topics with the same competence. During an experimental session at least one scientist is dedicated to each topic for the exploitation of the diagnostics. In particular for topics such as Charge Exchange Recombination Spectroscopy or Interferometry/ Polarimetry, 3-4 scientists are required. During each experimental session a Diagnostic Coordinator ensures that data are appropriately collected to fulfil the scientific objectives of the experiment. The exploitation of diagnostic systems consists of taking data, checking their consistency and using them in the evaluation of a physics scenario and finally in published papers. On average, 25-27 scientists are present at a time to cover diagnostics during the Campaigns; they provide important contributions to this complex process. Within each Topic the Operator of the JET Facilities provides an agreed level of support, starting from basic maintenance up to full operation, where not only calibration of the system but also some rough validation of the data is provided.

Specific experiments were dedicated to the collection of data fundamental for testing the capability of some important diagnostic systems (such as the MSE or CXRS Sytems) or the exploitation of new techniques. These required less than 5% of the experimental time.

Topic	Name	JET Identifier	Diagnostic Systems	Measurements
D-1	Thomson Scattering	KE3 KE9D	Core LIDAR EDGE LIDAR	Core and Edge electron temperature and density
D-2	ECE/ Reflectometry	KK1 KK3 KG3 KG8b	Michelson Interferometer Heterodyne Radiometer O-mode reflectometer E-mode correlation reflectometer	Electron temperature density fluctuations correlation of fluctuations
D-3	Magnetics	KC1 KC1D KC1F EFIT Analysis	Pick up coils; saddle loops; diamagnetic loops; halo sensors; full flux loops Pick up coils in the divertor Fast magnetics	Magnetic reconstruction, magnetic fluctuations, TAE modes Equilibrium reconstruction
D-4	Soft X-ray and Bolometry	KJ3/4 KB1/KB4/KB3D CATS	Soft X-ray diode arrays Bolometer cameras Fast Data acquisition system	Soft X-ray and plasma radiation emission tomography
D-5	Neutronics	KN1 KN3 KM2-9	Neutron flux monitor Neutron Profile Monitor and Fast Electron Bremsstrahlung Various 2.5MeV and 14.5MeV neutron spectrometers	Neutron yield, ion temperature profile
D-6	Charge Exchange Recombination Spectroscopy Motional Stark Effect	KS5/6 KS7 KS9	Core CXRS Edge CXRS MSE	Ion temperature, toroidal velocity, Impurity density profile Poloidal field profile; q-profile
D-7	Passive Visible Spectroscopy X-ray spectroscopy	KS3 KT1-2-3 KL1 KL2 KZ3 KX1 KH2 KS6	Visible spectroscopy VUV spectroscopy Video Cameras Laser Blow-off system High Resolution X-ray crystal spectroscopy	Zeff Impurity content Impurity injection Ion temperature
D-8	IR thermography	KL3	Thermography of the divertor	Two dimensional thermographic profile of the divertor tiles
D-9	Probes	KY3 KY4D	Langmuir probes	Electron density and temperature, ion temperature SOL and divertor
D-10	Beams	KY6 KT6D KT7D	Li-beam Thermal He beam	Edge electron density profile Te and ne edge
D-11	Neutral Particle Analyzer	KF1 KR2	High Energy NPA(MeV) Low Energy NPA	Fast and low energy ion distribution function
D-12	Interferometry polarimetry	KG1 KG4	Far Infrared Interferometer Faraday Polarimeter	Electron density Poloidal magnetic field

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Table 3.1: Task force D Topics and Associated Diagnostic Systems

3.7.2 Scientific Issues and Results

3.7.2.1 Data Validation

An important part of the work of Task Force D is the production and validation of data. This process has led to a reconsideration of some important measurements such as the electron temperature, density and Z_{eff}. For example, the differences in the electron temperature as measured by ECE and LIDAR Thomson Scattering have triggered fundamental research in the ECE emission of high temperature plasmas. The comparison of the Z_{eff} measurement made using Bremsstrahlung and charge exchange data is another active field of improvement. Detailed experimental analysis of these issues will be carried out during the Campaigns of 2002.

3.7.2.2 New Measurements

Electron Density Measurements

Measurements of spatial profiles of the electron density in the plasma edge are important for understanding the physics of the H-mode. Fig. 3.30 shows good agreement between the density profile as measured by the edge LIDAR Thomson Scattering and by Lithium beam spectroscopy. There are limitations of the density and temperature measurements associated with the various diagnostic systems. The main limitation is that data does not extend to the top of the pedestal of the H-mode. This triggers further development of diagnostics (see below and section 5).

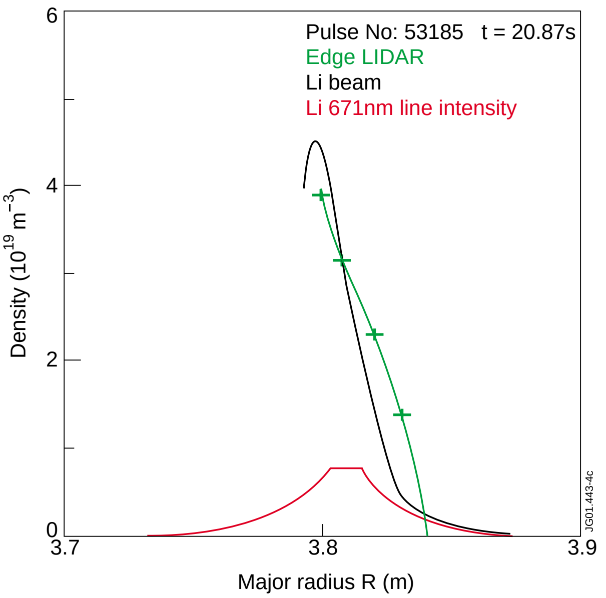


Figure 3.30: Density profile at edge measured by the Edge LIDAR Thomson Scattering and Li-beam spectroscopy.

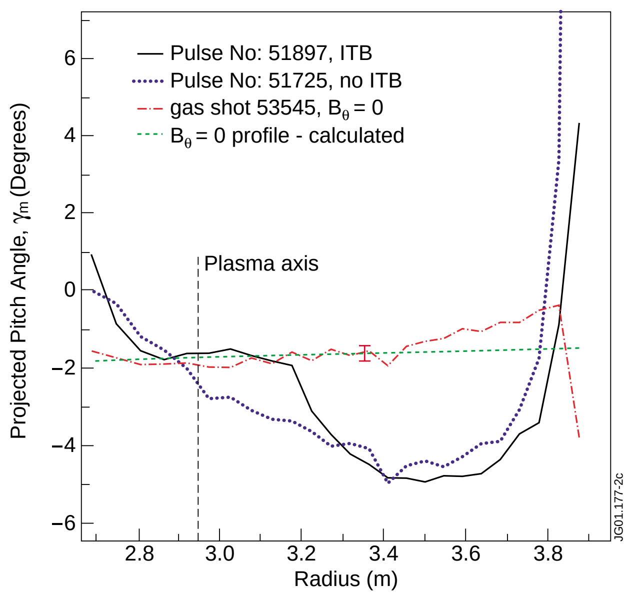


Figure 3.31: Pitch angle (γ_m) measured versus major radius showing that the (γ_m) measured during ITB (black trace) is close to the value measured without plasma current (red trace) in the central region.

MSE Measurements

New measurements were obtained using the Motional Stark Effect (MSE) diagnostics in shear reversed discharges. Fig. 3.31 shows the measurement of the pitch angle made by the MSE system in such a discharge. The measurement (black trace) shows a large region (~20cm) in which the plasma current is very low: the MSE pitch angle value is close to that measured without plasma.

Measurement of Poloidal Velocity

The poloidal velocity is important because it is needed for the determination of the radial electric field (E_r) through the force balance equation. The measurement of E_r is relevant for assessing models for the formation of transport barriers (ITBs) in the plasma core as well as in the edge (H-mode). For the first time the poloidal velocity (v_θ) has been measured in JET in optimised shear discharges. Fig. 3.32 shows the result of the measurement of v_θ (full trace) and the calculated neoclassical value (dotted trace). The resulting v_θ of a few km/s is in rather good agreement with neoclassical values.

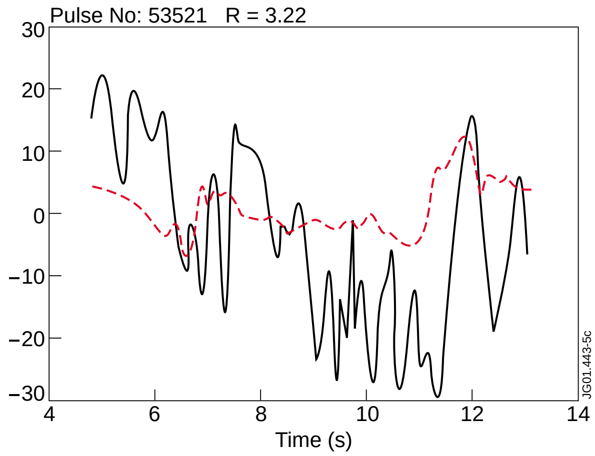


Figure 3.32: Poloidal velocity (v_θ) versus time at R = 3.22m (Solid trace), neoclassical calculation (dotted)

He Beam Emission Spectroscopy

A new measurement on JET has been provided by Beam Emission Spectroscopy of helium. The helium lines from a thermal helium beam can be used for diagnosing the edge of the plasma, i.e. for measuring the electron density and temperature. In addition, energetic He beams provide the advantage that helium lines can also be used for measuring the plasma density and temperature further inside the plasma. Helium doped 130kV deuterium beams have been made available on Octant 8 and tests have been conducted for measuring the intense helium singlet line which can be used for diagnosing the plasma. Fig. 3.33 shows the emission profile of this line at 587.6 nm (in red) well separated from the plasma background emission.

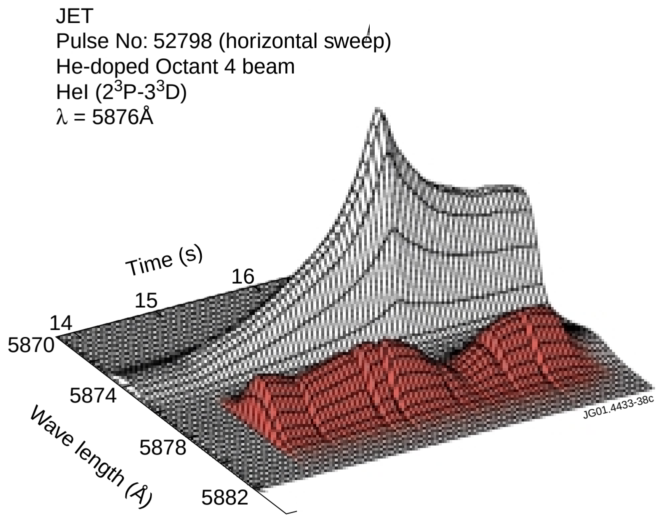


Figure 3.33: Beam emission profile showing the doppler-shifted emission (front red) and the unshifted emission (back) for doped 74kV beam.

3.7.3 Diagnostic Enhancements

New or improved diagnostics will be available for particularly important areas of research during the Experimental Campaigns of 2002.

Advanced Tokamak Physics:

- Motional Stark Effect Upgrade
- Correlation Reflectometry
- Real Time Control
- ECE Michelson Interferometer
- ECE Heterodyne Radiometer
- Error Field Correction Coils

H-mode Physics:

- Edge LIDAR Thomson Scattering
- High Time Resolution Pellet Spectrometer
- Li-Beam
- He-Beam

SOL and Divertor Physics:

- Reciprocating Probe Heads
- Quartz Microbalance

Further details are given in Table 2.1, 2.2, 2.3 and in section 5.1.2.

3.7.4 JET-EP Diagnostic Working Group Evaluation and Assessment

The strategy for the development of the JET diagnostics during the medium/long term has been discussed and assessed by the JET-EP Diagnostic Working Group (DWG). A staged approach has been proposed, considering diagnostics to be installed in 2002, and in the period 2003-2004. The diagnostics to be installed in 2002 are listed in section 3.7.3. For the longer term, high priority has been assigned to diagnostics which could provide an overall improvement of the physics information across the whole JET programme:

- **New Charge Exchange Recombination Spectroscopy (CXRS) system**: for increasing the availability and accuracy of measurements of ion temperature, rotation, impurity content.

- **High Resolution Thomson Scattering System**: for studying the H-mode pedestal and ELMs as well as ITB dynamics, with improved spatial resolution (~1.5cm) and high repetition rate (10-30Hz).

- **Vertical Bolometer**: for guaranteeing the availability of tomographic reconstruction of the emitted radiation, with a possible new improved spatial resolution in the divertor.

Equally high priority is recommended for ITER relevant projects:

- **Infrared Viewing System**: the aim is to measure the power load to the wall and divertor (contributing to ELM studies), the synchrotron radiation from runaways in disruptions, and lost alpha particle distributions.

- **Microwave Access**: a new set of ITER-grade corrugated waveguides for broadband operation is required to increase the transmission of the receiver optics, thus allowing measurements of reflectometry and ECE with strongly improved accuracy .

- **Halo Current Sensors**: for measuring the halo currents in the divertor

- **Mirror Damage Test**: installation of mirror samples of Molybdenum and Tungsten as well as Rhodium is required for studying the erosion and deposition on plasma facing mirrors.

- **Diagnostics for the Study of Tritium Deposition and Retention**.

Other diagnostic systems of high interest for JET but with a narrower scope are also recommended:

- **TAE antennas**: the aim is to be able to study the stability properties of ITER-relevant TAE modes, with toroidal mode numbers n=5-10.

- **Time of Flight Spectrometer**: for neutrons produced in deuterium discharges:
  The aim is to provide measurements of high energy particles and fuel reactivity on high performance deuterium discharges.

- **Lost alpha particle detectors**.

- **Magnetic proton recoil neutron spectrometer for 14MeV neutrons**.

These diagnostic enhancements are studied in the frame of the “longer term enhancement projects” described in Table 2.3 and section 5.2.

3.8 The Fusion Technology Task Force

3.8.1. Scope and Objectives

The activities of the Fusion Technology (FT) Task Force are closely related to the EFDA Technology Workprogramme, and several tasks are even implemented within that frame. In view of the ITER and long term programme requirements, this Task Force places emphasis on the following topics:

- Tritium in the Tokamak (in collaboration with the Task Force E);
- Waste Management and Tritium Processing;
- Plasma Facing Components (in collaboration with the Task Force E);
- Engineering, including Remote Handling, Inspection Techniques and Reliability Assessment;
- Safety.

The Task Force also coordinates the involvement of Associations in the use of the JET Test Beds.
As shown in Fig. 3.34, more than 60% of the activities under the FT Task Force are directly or indirectly related to tritium and its technology. High priority has been given to the assessment of tritium in first wall materials (graphite and CFC tiles as well as flakes), to the investigation of the behaviour of high-Z first wall materials such as tungsten, to the determination of the conditions leading to tritium release from these materials and to the development of techniques for their conditioning. Further development of tritiated waste management technologies and water detritiation has also been given a high priority.

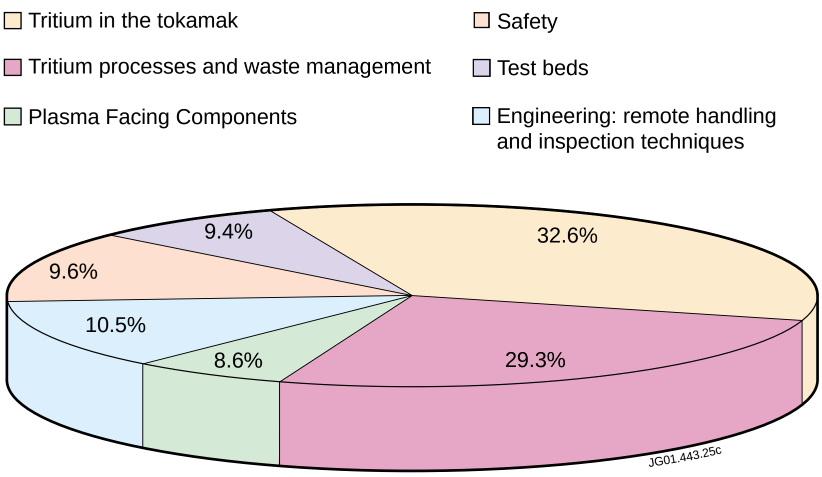


Figure 3.34: JET FT Tasks 2000-2001 distribution by S/T Tasks

3.8.2 Organisation

The activities of the Task Force concentrated mainly on experimental work carried out in the Associations with materials coming from JET (e.g. beryllium and/or tritium contaminated samples and flakes removed from the vacuum chamber) and new tiles produced by the Associations for installation in the machine as well as on experimental work conducted for or at the JET Facilities (e.g. improvement of the performance of the Active Gas Handling System). Some activities involve work during the 2001 shutdown of the machine, while others are performed independently of the machine operation schedule. The Operator is involved in practically all tasks.

3.8.3 Overview of Results

One of the key issues investigated within the FT task force refers to the tritium balance in the JET machine: after the DTE1 campaign at the end of 1999 the residual tritium inventory in the machine was estimated to be about 2.0 grams. Of that tritium, 0.2 g appeared to be trapped in both first wall and divertor tiles while the rest is believed to be trapped in flakes in the JET sub-divertor region. Further understanding Tritium trapping in tiles, flakes etc. therefore constitutes a major topic of the FT Task Force activities.

Tritium trapping in tiles has been measured by a number of techniques and large number of detritiation procedures have been examined in various Associations in screening tests. They include:

a) treatment with oxygen/ozone;
b) exposure to an open flame;
c) direct high frequency heating;
d) dishwasher method using a low viscosity easily vaporisable liquid;
e) exposure to an argon plasma burner;
f) mechanical abrasion;
g) treatment with moist air (or other carrier gas) at moderate temperatures;
h) irradiation with an excimer laser xenon pinch lamp;
i) high temperature outgassing under vacuum or noble gas.

Fig. 3.35 shows a cross section of the JET machine through the divertor. The tiles on the left and right side belong to the inner and outer areas of the divertor, respectively. The numbers indicate the tritium activity near the surface in MBq/cm^2 at various locations. The tritium trapping patterns are consistent with Be and D deposition patterns, as shown in Task Force E studies (see section 3.4).

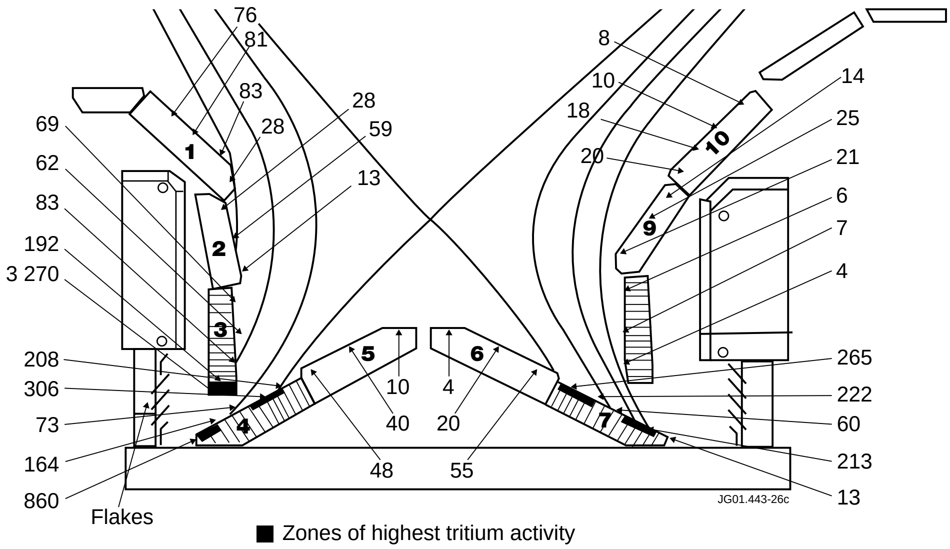


Fig 3.35: Cross-section of the divertor with tritium activity in MBq/cm^2 at various locations

Flakes are also studied. Experimental investigation showed that of the 32 Ci of tritium estimated to be initially contained in a flake sample of 1 g, a total of 11.4 Ci was released after 21 months storage at room temperature.

Tungsten is one of the first wall materials foreseen on ITER. In order to further study the behaviour of such high-Z first wall materials, a set of W-coated divertor tiles have been manufactured/characterised and will be installed in the machine during the 2001 shutdown for measurement of W erosion after plasma operation.

In the field of tritium processing, a permeator/catalyst hybrid unit (PERMCAT) has been built and shipped to JET for installation in the JET Active Gas Handling System (AGHS). After inactive commissioning the PERMCAT will be used for final detritiation of gases during routine operation of the JET machine. This should further improve the already high efficiency of the AGHS and tests technologies foreseen for ITER.

A high priority has also been given to the detritiation of water. Work is going on in view of characterising the best type of catalyst for the isotopic exchange in a Combined Electrolysis Catalytic Exchange (CECE) column and designing a water detritiation plant. Such a plant could be built on JET, both with the intention of easing JET waste management and to support the ITER design.

Other tasks involve collecting operating experience on system component malfunction, remote handling, development of in-vessel inspection techniques and development and validation of activation models. The latter task has been completed. Calculations have been performed to model the neutron transport throughout the JET installation using the MCNP™ code. A very detailed computer model of the tokamak, the torus hall and the associated additional heating and diagnostic systems has been created. This was used to estimate the neutron flux as a function of energy at 32 locations at which "activation foils packs" had been placed during

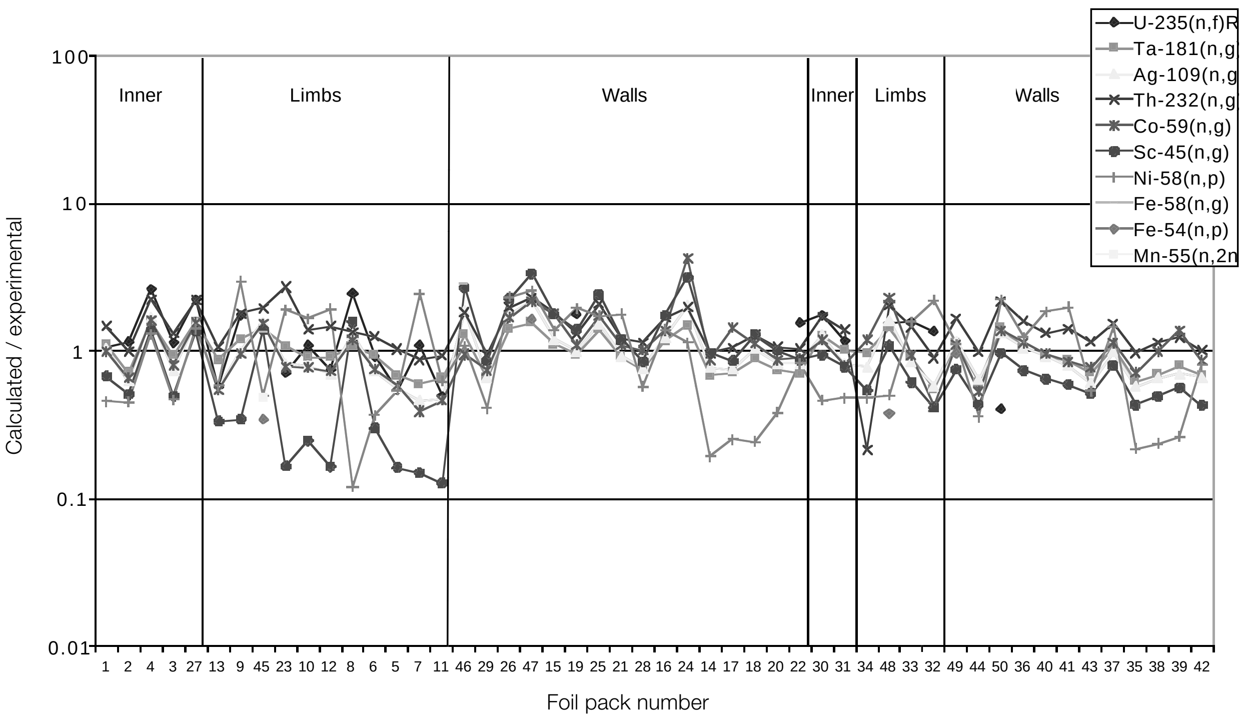


Figure 3.36: Calculated / experimental values following using adjusted spectra.

the 1997 DT experiment. The neutron spectra computed using MCNP were used as input to the inventory code FISPACT that was used to compute the expected levels of activation in the foils. A comparison of calculation and measurement is used to derive a semi-empirical technique for the determination of neutron activation levels throughout the torus hall. The technique can determine the neutron activation level at any point in the torus hall to within a factor of two (see Fig. 3.36). The same model has been used for estimates of the gamma dose inside the vessel which arises from activation of the machine itself. These have been compared with health physics measurements and agree to within the experimental and calculational error bars (~20%).

4. Operation of the Facilities

4.1 Introduction

4.1.1 Hand-over and Preparation for Operation

At the end of 1999 the JET Facilities were put into a passively safe state ready for hand-over from the JET Joint Undertaking to UKAEA to minimise the problems that might arise because the hand-over was during the Christmas holiday period. This approach also minimised the risk from the ‘Millennium computer bug’. Preparations by the JET Joint Undertaking for the hand-over were complete by 23 December 1999 and took place formally at midnight on 31 December, as planned. The hand-over took place without any problems, but the transition period was not completely uneventful because of unforeseeable, and unrelated, failures of two electricity cables feeding the Culham site on 25 December and 6 January. Following hand-over, systems were progressively switched on and a programme of maintenance and upgrade activities started on 24 January in preparation for the first EFDA experimental campaign which was planned to take place in late May 2000. The most significant activities in this period were:

- Upgrading 2 of the 8 sources on one of the Neutral Beam systems.
- Curing minor leaks that had developed during 1999 on two windows on the vacuum vessel.
- Installing new RF matching systems (SLIMPs) on two of the Ion Cyclotron Resonance Heating antennae.

Preparations for the restart of operation of the JET Facilities commenced on 16 March 2000 with the pump-down of the vacuum vessel (Fig. 4.1). The various systems, including machine protection, were then commissioned, leading up to first plasma operation on 3 May. Thereafter a 3-week programme of vacuum, plasma and heating system conditioning started. The EFDA Associate Leader for JET accepted the JET Facilities as ‘ready-for-use’ on 26 May, as originally planned.

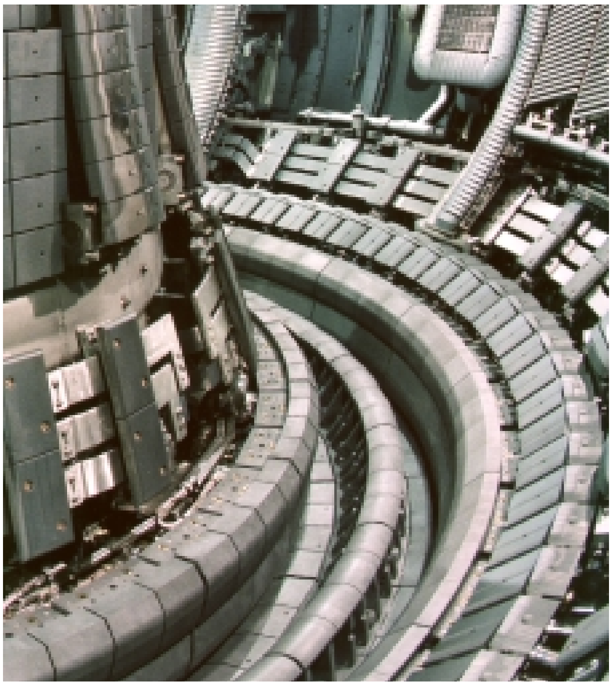


Figure 4.1: Shows an inside view of the torus with the Mark IIGB divertor used during the campaigns C1-C4

4.1.2 Experimental Campaigns during 2000 and 2001

The EFDA arrangements for the use of the JET Facilities represent a significant change compared with those under the JET Joint Undertaking, mainly through the formal separation of the responsibilities for the planning and execution of the experimental programme from the operation of the Facilities. The interfaces between the Operator and the EFDA CSU and Task Forces that this separation has generated have worked well right from the very beginning of the first experimental campaign. Although there is a formal separation of the Operator role from the execution of the EFDA-JET experimental programme, some EFDA secondees help the Operator run the JET Facilities; in 2000/01 28 secondees from 9 EFDA Associates were involved in JET Operations.

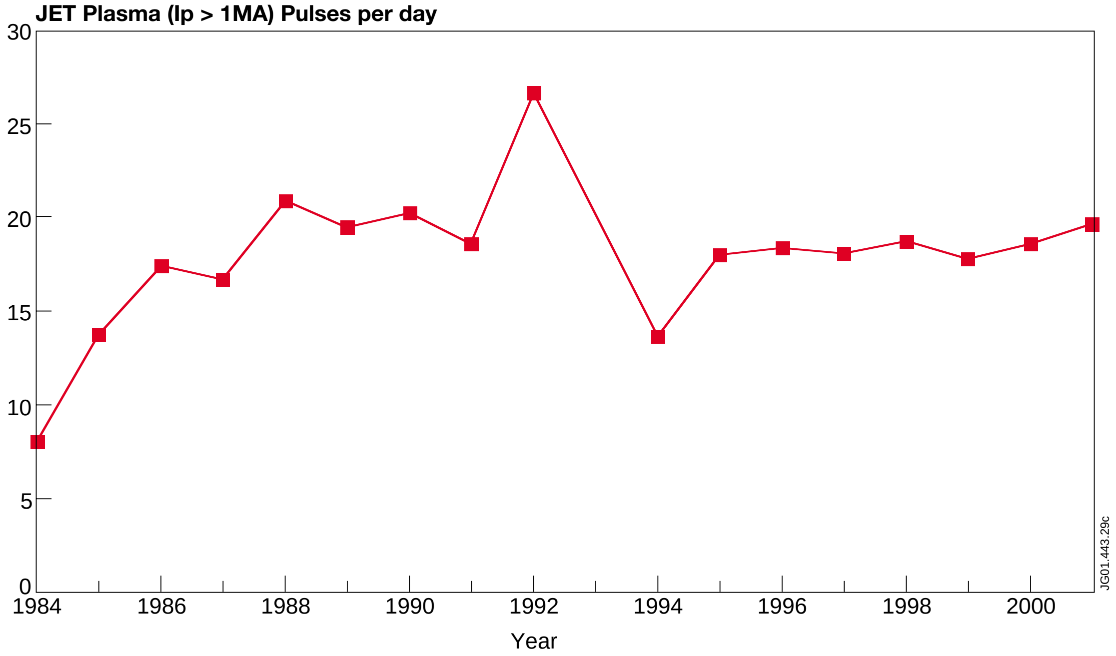


Figure 4.2:The average number of tokamak pulses with plasma current >1MA per operational day achieved in each calendar year on JET. The operation up to the end of 1999 was during the JET Joint Undertaking.

The new arrangements require experiments to be defined well in advance, including when in an experimental campaign they will be done, so that the visits of scientists can be arranged to match the experiments with which they are involved. This means that it is very important that all systems are working well by the start of the campaigns and that any faults occurring during campaigns are fixed as quickly as possible because there is limited flexibility to rearrange the timing and duration of experimental campaigns. In practice all systems have been available for operation at the performance levels required by the EFDA-JET Task Forces on the planned starting dates of the four experimental campaigns in 2000/01. Equipment reliability during the campaigns has also been very good overall but, as is inevitable, there have been some equipment failures. All of the failures have been relatively minor one-off random events with no signs of an underlying cause.

The total number of days of operation lost because of the need to fix faults amounted to 12 days out of a total of 164 days of operation. This is consistent with the operational experience during the JET Joint Undertaking. The number of tokamak pulses per day achieved in the first year of operation of the JET Facilities by UKAEA has also matched expectations based on the experience of the JET Joint Undertaking, as shown by the last two data points in Fig. 4.2. This is the most obvious overall demonstration that the amount of lost time during operational days and the efficiency of Control Room (Fig. 4.3) operations has been maintained at about the same levels as in previous years, despite some loss of expertise that occurred at the end of the JET Joint Undertaking.

4.1.3 Safety Aspects

One of the major, but less visible, tasks during 2000/01 has been to bring the JET safety organisation in line with the UKAEA's safety system. The safety cases for the torus and Active Gas Handling System have been updated and safety cases written for other major items of plant. Around 700 safety related appointments have been made. Prior to the start of the experimental campaigns, all safety-related systems were re-commissioned.

Figure 4.3: Photograph of the JET Control Room

The hazards to personnel associated with both the operation of JET and engineering work during interventions have been effectively controlled. Collective radiation doses to radiation workers on the site have been extremely low. There have been no significant safety related incidents.

The Engineer-in-Charge takes on the responsibility for ensuring that the operation of JET during experimental sessions is done safely and within the design and administrative limits defined by the Operating Instructions. These controls have worked very effectively. Four new Engineers-in-Charge were trained in the period, bringing the complement up to 12 (see fig 4.4).

Figure 4.4: New Engineers in Charge

4.2 Component Systems

4.2.1 Power Supplies

During Campaigns C1-C4 the JET pulsed power supplies remained essentially unchanged and effort concentrated on increasing the availability and reliability of the systems. Improvements to the maintenance of the tokamak power supplies resulted in a 40% increase in the reliability of the Tokamak power supplies and hence a significant reduction of operational downtime to less than 4% during Campaigns C3 and C4, compared with an average of 8% in 1999. The availability of the Neutral Beam power supplies has also been very good, increasing from 85% to 91% despite the programmes placing increasingly higher demands on ageing equipment. The operation of both the Ion Cyclotron Resonant Heating and Lower Hybrid Current Drive power supplies has been exceptional with 100% availability recorded in C3 and 99.5% and 99.3% respectively in C4.

4.2.2 Neutral Beam Heating

Injecting energetic neutral beams is a very flexible way of heating plasmas and was used in nearly every experimental session in Campaigns C1-C4 over a wide range of plasma operating conditions. There are two "Neutral Injector Boxes" (NIBs) on JET, one producing 80keV and the other 140keV beams. During Campaigns C1-C4 Neutral Beams (NB) were used in the highest ever proportion of JET pulses (>80% in C4), and the average NB pulse length was higher than in previous campaigns (>6s in C4). The maximum injected power reached 18MW. Both NIBs were used for an extended period of helium beam operation in C4, surpassing the previous number and total duration of helium beam pulses in any previous campaign by more than a factor 2. Also, deuterium NBs with 1-2% helium "doping" were successfully used for novel plasma diagnostic applications on both NIBs.

One of the eight Positive Ion Neutral Injector (PINI) units on the 80kV NIB experienced a small leak from a water cooling circuit in July 2000, and had to be replaced during August ready for Campaign C2. A residual ion deflection magnet suffered an internal electrical failure during March 2001, resulting in the loss of two PINIs for the final 4 weeks of Campaign C4. It will be replaced by a spare during the 2001 shutdown.

4.2.3 Radio Frequency (RF) Operations

The Ion Cyclotron Resonant Heating (ICRH) system, which operates in the frequency range 23MHz to 57MHz, was used extensively throughout Campaigns C1-C4. All the Task Forces used the ICRH system for different experiments and a major achievement of the system was to couple approximately 15MW of RF heating power to the plasma, the highest for several years. The efficiency and reliability of the ICRH plant throughout the Campaigns C1-C4 was very good.

Specific experiments related to optimising the ICRH system were performed under Task Force H, including the testing and optimisation of the new Sliding Impedance Matching networks (SLIMPs) (see section 3.6.). These devices attempt to provide robust matching of the RF amplifiers to the plasma in order that the ICRH power can continue to be delivered even during sudden expulsions of particles and energy called Edge Localised Modes (ELMs).

4.2.4 Lower Hybrid Heating and Current Drive

The availability and performance of the Lower Hybrid Current Drive (LHCD) system has been good throughout 2000/01. Improvements such as the implementation of individual feedback control of the power of the 24 generators and modification of the protection against high reflected power have increased the reliability of LHCD. Also, the new real time protection on radiation from the grill has successfully minimised damage to the grill and prevented plasma disruptions. More than 4 MW of LHCD power has been coupled in L-mode plasmas, for the first time since 1997. Following successful experiments by Task Force H to improve coupling of the lower hybrid waves to the plasma by means such as optimisation of the plasma shape and injection of deuterated methane (CD_4) near the grill (see Fig. 4.5), more than 3.5 MW of LHCD have been injected in plasmas with both ELMy H-modes and with internal transport barriers. As a result of these improvements requests for LHCD have increased, with pulses having ≥3 MW of LHCD for more than 10s being reliably produced during Campaign C4.

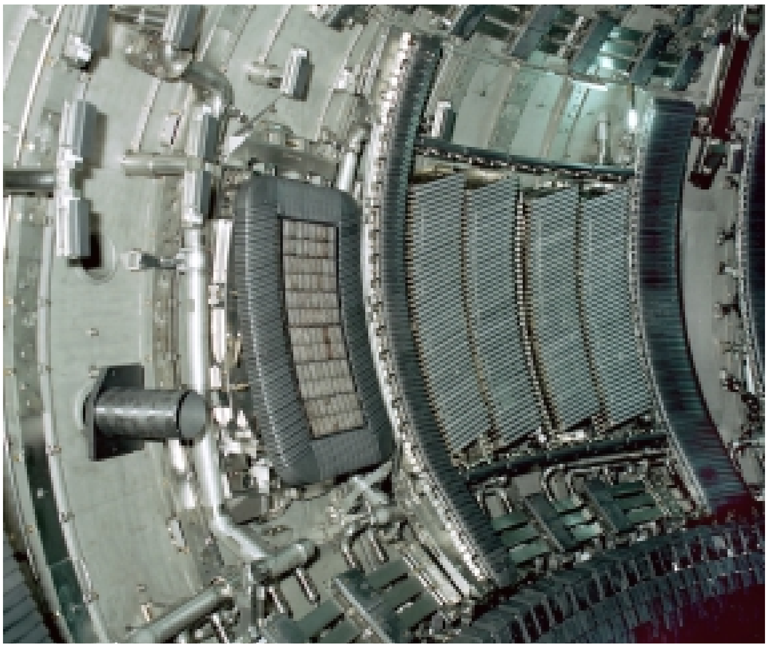


Figure 4.5: ICRH Antennae (Right hand side) and LHCD Coupler (Centre left)

4.2.5 Pellet Injection

JET is equipped with an extruder and centrifuge system designed for delivering 4mm sized pellets at speeds up to several hundred m/s at a repetition rate of 5Hz for studies of plasma refuelling. The pellets can be diverted along one of two tracks, for injection either on the outboard (Low Field Side) side of the plasma, or on the inboard (High Field Side) side. During Campaigns C1-C4, inboard launch, at pellet speeds up to 160m/s, was most commonly used in experiments. During August 2000 a vacuum leak developed on the centrifuge, but as the system can be isolated from the torus it was possible to remove the entire assembly for repair and it was successfully re-instated without disruption to JET operation.

4.2.6 Cryogenics

The experimental campaigns are reliant on the availability of liquid nitrogen and helium to supply the cryopumps in the Pumped Divertor, the NIBs and the LHCD system. Cryogenic liquids are also needed to produce deuterium ice in the pellet injector. This necessitated permanent running of the Cryoplant from the beginning of C1 through to the end of C4 to supply the continuous requirement of liquid nitrogen and liquid helium.

Despite the original parts of the Cryoplant being almost 20 years old, the reliability of the helium liquefiers and distribution system was very good with only two shifts being affected by the loss of helium temperatures on the Pumped Divertor. However, to achieve this high level of performance, repairs and essential maintenance had to be done over-night and at weekends on several occasions.

In Campaign C4 the NIBs were run using helium gas. This required argon frosting of the Cryopumps and daily, rather than weekly, regenerations of the NIB and Pumped Divertor Cryopumps. This arduous duty cycle put severe loading on the helium liquefaction capacity of the plant, but in spite of this, very little operational time was lost during the campaigns.

4.2.7 Vacuum, Gas and Vessel Conditioning

Very good torus vacuum and surface conditions are essential to successful plasma operation. These were achieved using vacuum conditioning techniques such as glow discharge cleaning and beryllium evaporation. There were three occasions when vacuum conditions were affected by leaks. The most disruptive of these was caused by a broken weld on a redundant diagnostic attached directly to the torus. The recovery from this leak was however exceptional: the leak was located, the torus cooled and vented, the weld repaired and the vessel reheated and reconditioned with only six operational days being lost.

The torus was mainly operated at a temperature of 320^{o}C, as has been traditional on JET, but the temperature was lowered to 200^{o} C for the last campaign in 2001 to investigate the effect of vessel temperature on plasma performance in more ITER relevant operating conditions. Operation at the lower temperature proved to be successful. This opens up the possibility of operating at the lower temperature in future, which would reduce some operational risks.

Gas for plasma fuelling and seeding can be introduced into the vessel at 10 different positions. The experimental programmes required a challenging number of different gases to be injected. This included the introduction of gases never previously used on JET, such as silane (SiH_4) and methane isotopes. The systems for the distribution and injection of gas proved to be highly reliable and sufficiently flexible to meet the requirements of the experimental programme.

4.2.8 Active Gas Handling System

The Active Gas Handling System (AGHS) (see fig. 4.6) provides a closed loop for the feeding, recovery and recycling of tritium into the JET torus. No tritium was injected into the machine during Campaigns C1-C4. However, residual tritium remains within the machine from the successful experiments carried out in 1997 and the role of the AGHS has been to remove tritium contamination from the machine exhaust gases prior to their discharge to the environment. It has done this successfully. The only significant problem with the plant was when one of the vacuum pumps failed during August and had to be replaced; this had no impact on the experimental programme. The AGHS also acts as a test bed for tritium handling processes and devices designed for future fusion machines. Work has been done in close collaboration with other European laboratories to install such devices in the AGHS during the recent campaigns.

Figure 4.6: Active Gas Handling System (AGHS)

4.2.9 Diagnostics

Most experiments on JET are complex and require a large number of diagnostic systems to be operating simultaneously. During Campaigns C1-C4 more than 50 different diagnostic sub-systems, using a wide variety of diagnostic techniques, have been used to monitor and control plasma performance. A huge volume of digital data is generated which requires management and storage and large computer systems with analysis programmes requiring input from many diagnostic measurements. There were no major diagnostic plant failures. During JET experiments in 2000 and 2001 new diagnostic techniques were developed which allow the determination of the radial electric field inside the plasma volume (see section 3.7.2.2).

4.2.10 Plasma Control

The configuration of JET is very flexible, which provides challenges as well as benefits: there being about one thousand parameters and waveforms that must be selected to ensure correct operation of a plasma pulse. This situation is rendered practical by CODAS software tools (Level1) which allow these parameters to be prepared in advance for a complete operational session and loaded pulse by pulse (the Pulse Schedule). A recent development by CODAS and the Plasma Operations Group allows each pulse to be built up from up to six ‘Standard Scenarios’ each of which represents a distinct phase of the pulse. There are now thirty such standard scenarios to choose from. On top of these standard scenarios, the plasma formation phase can be selected from a library. Two new scenarios have been added to the library covering low voltage breakdown in helium and a new scheme for breakdown appropriate to “Advanced Tokamak” operation where the reproducibility of the initial plasma current profile is at a premium. These developments have greatly simplified the task of the Session Leader and made it possible for new scenario developments to be exploited by the Task Forces.

The tight geometry of the MarkIIGB divertor emphasises the need for close control of the plasma shape, which can be challenging at high plasma pressure when the plasma can contact the septum of the divertor. This problem was avoided in 2000 by the development of an additional control scheme, called "septum avoidance control" which allows control of both the two "strike points" and the location of the X-point of the magnetic separatrix relative to the septum. As a result it has been possible to extend the duration of the high performance phase of "Advanced Tokamak" regimes to 8 seconds.

Plasma shape is an important parameter for fusion physics because the elongation and triangularity of the plasma cross-section have a strong bearing on plasma stability and confinement. New shapes have been developed by the Plasma Operations Group that match the shape proposed for the reactor-scale ITER device. These new shapes have been exploited by the Task Forces to provide valuable insights into how ITER will behave.

Plasma instabilities can adversely affect the control of the plasma, sometimes leading to Vertical Displacement Events (VDEs) which can put large forces (many hundreds of Tonnes) on the vacuum vessel. This can be particularly challenging for the ITER-like shapes. Important progress has been made during the year, such that only a few such events have occurred. More extensive improvements are in preparation in order to permit safe operation of these plasmas at higher plasma currents.

4.2.11 Control and Data Acquisition Systems (CODAS)

During 2000/01 CODAS concentrated on maintaining and improving the support for Tokamak operations and responding to new requirements as they arose without affecting the experimental programme. A large number of small enhancements were made, ranging from changes to the front-end real-time control systems to the provision of standard plasma scenarios in the high-level (Level1) operations-related software. In parallel, major work was carried out in support of the CSU - including the EFDA-JET web pages; in support of the EFDA Secondees - especially the thin-client desktop solution; in preparation for the small enhancements and the JET-EP; on the IBM replacement project and in many aspects of remote participation.

Figure 4.7: Sun equipment. Space was always a problem during campaigns C1-C4

4.2.12 Real-Time Systems

A number of enhancements have been made to the Real-Time control systems over the past year.

- The incorporation of longer time windows for fast data acquisition into the magnetic diagnostic has improved plasma instability detection and the accuracy of energy calculations.

- The Lower Hybrid heating system has undergone both hardware and software upgrades which has resulted in extremely accurate real time feedback control of the energy injected into the plasma.

- The Electron Temperature Profile diagnostic has been incorporated into the real-time feedback control system (see section 5) and used to control successfully the position of internal transport barriers within the plasma.

More generally, support for faster network communications and more powerful embedded computing systems has been added to the majority of the CODAS standard software libraries.

4.2.13 Central Computing and Data Management

JET used an IBM mainframe as its main storage and analysis computer for a period of almost twenty years. The IBM also supported the main CAD/CAM software (CATIA) used by the JET Drawing Office as well as the data archival system consisting of two Automatic Tape Libraries (ATLs) housing a total of almost 8000 tapes and ~10TBytes of data. These systems were expensive to run and were reaching the end of their useful lives.

The IBM Replacement Project started in January 2000 and consisted of:

- installing enough extra PCs running Linux on the JET Analysis Cluster (JAC) to accommodate the analysis load that was still running on the IBM mainframe;

- replacing the mainframe with a UNIX system for data storage and archival (see fig 4.7);

- increasing the StorageTek ATL storage capacity by moving to newer tape technology - 20GB tapes compared to 800MB- and copying all the legacy data to the new tapes;

- moving the CATIA users to standalone workstations;

- transferring the remaining analysis and predictive codes to the JAC and replacing the IBM terminals in the control room with X-terminals.

Beneficial side effects of the Project are that the number of supported operating systems will be reduced, one of the ATLs can be removed because of the large increase in tape capacity and there are large savings in space occupied and running costs.

The original plan was to switch off the IBM mainframe during the shutdown of 2001 following Campaign C3. However, the change of the starting date of Campaign C4 to the beginning of 2001 required the project to be rescheduled and made it attractive to install in Campaign C2 a smaller, replacement mainframe with new disk technology in advance of the full replacement of the IBM by the Solaris based system. The hardware installation was very successful but unfortunately the reduced CPU power led to temporary performance problems until the inter-shot analysis was moved from the IBM to the Unix systems 2 weeks later.

The project finished in September 2001 after the final validation of some analysis programs, the installation of new ATL technology, the movement of all the data from the mainframe to Unix; the operation of a Solaris-based data server and disk cache; the daily use of 56 Linux based JAC computers; the movement of CATIA users to stand-alone workstations and the movement of all of the analysis and predictive codes to Linux or Solaris computers.

5. Enhancements

Enhancement activities aim at increasing the performance capability of the JET facilities. A first set of such activities was launched early in the 2000-2002 Workplan to allow implementation during the last two years of the Workplan. These are described under the heading “2000-2002 Enhancements”. Studies have also been conducted for enhancements related to the possible use of the JET facilities beyond 2002. These have been conducted mainly in the frame of the JET Enhanced Performance (JET-EP) project and are described under the heading “Longer Term Enhancement Projects”.

5.1 2000-2002 Enhancements

The main objectives of the 2000-2002 Enhancements are:

1. To increase the heating power, in particular by upgrading the Neutral Beam Injector (NBI) on Octant 8 and improving the Ion Cyclotron Resonance Heating (ICRH) and the Lower Hybrid Current Drive (LHCD) coupling capabilities.
2. To improve the JET diagnostics capabilities, in particular the plasma edge and divertor diagnostics in order to provide information for the detailed design of the ITER divertor.
3. To improve the fuelling capabilities, particularly regarding the pellet size and penetration.

5.1.1 Enhancements of Heating Systems

Octant 8 Neutral Beam Upgrade

The upgrade of the Neutral Beam Injection system on Octant 8 aims at approximately doubling the injected power capability of one of the two NBI boxes, namely that situated on the Torus at Octant 8. This constitutes the most significant of all of the 2000-2002 Enhancements. The expected power increase of ~ 7.6 MW (in D^0) results from doubling the current capability (from 30A to 60A) of the 8 Positive Ion Neutral Injectors (PINI). The project started at the end of 1999 in anticipation of the new Agreements and the main procurement contracts (new power supplies and new grids) were placed to industry during 2000. The design of the new power supplies is complete, and the existing power supplies are being reconfigured to feed four new upgraded PINIs which are being installed during 2001 to maintain the power for experimental Campaign C5 (March 2002). Contracts for the procurement of the other sub-systems belonging to this upgrade are being launched, in line with the project plan. The increased power will be available in the last quarter of 2002.

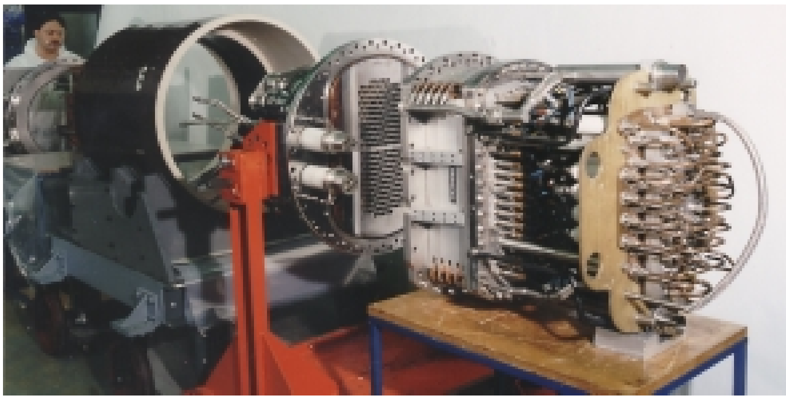


Figure 5.1: New PINI

Ion Cyclotron Resonance Heating - 2nd Harmonic Protection

This project pursues the development of a protection system for the power generator against the generation of second harmonic frequencies. A basic filter was built and tested at low and high power in TEXTOR during 2000. The construction of a prototype filter for JET is under way. It will be installed in two transmission lines and tested in plasma conditions during the 2002 experimental programme.

Ion Cyclotron Resonance Heating - 3DB Coupler

The aim of this project is to provide resilience to Edge Localised Modes (ELMs) which can affect significantly the ICRH coupling capabiliy in the ELMy H-mode, the ITER reference mode of operation. A study is being carried out to assess the work needed and the expected benefits of installing a Hybrid Coupler ELM Reflection Compensation Circuit for the ICRH antennas. This could bring substantially increased power handling capability, up to about a factor of two, in a variety of experimental conditions.

Improved Neutral Beam Injector Neutraliser

The aim of this project is to investigate the reasons for the lower than theoretically expected neutralisation efficiency on positive ion beams. Experiments and studies are being carried out in the NB test bed to characterise the plasma in the neutraliser and identify possible improvements with a view to significantly increasing the power delivered to the plasma. It is estimated that the neutralisation efficiency could be increased by up to ~ 25%.

Lower Hybrid Current Drive Coupling Improvement

A preliminary study is under way to assess the feasibility of two proposals: gas injection through the launcher in order to improve the LHCD coupling by increasing the density in front of the launcher up to 2 to 5 10^{18} m^{-3}; and plasma ionisation techniques in order to improve the coupling by focusing power in front of the LHCD launcher. The result of this study will be used to identify further actions aimed at improving the coupling of LHCD.

5.1.2 Diagnostics Enhancements

Edge LIDAR Thomson Scattering

In order to measure temperature and density of the edge plasma by LIDAR Thomson Scattering with improved sensitivity, a new set of photo-multipliers with GaAs photo-cathodes with increased quantum efficiency, are being procured. Following the testing of a prototype, the new detectors will be available for the 2002 Campaigns.

Reciprocating Probe Head Upgrade

Reciprocating Probe heads are used for measuring edge plasma characteristics. Four new probes will be available in 2002:

1. A new Retarding Field Analyser Probe (measurement of SOL T_i profiles);
2. A new Turbulent Transport Probe which will allow measurement of electron temperature fluctuations and the effect of perturbations due to the probe itself on the transport measurements;
3. Plasma Ion Mass Spectrometers for mass analysis of the scrape-off layer plasma;

4. Collector Probe Heads with rotating sample holders for studying impurity transport.

The design of the new probes is complete and the procurement of components has already started.

Quartz Micro-Balance

A new Quartz-microbalance system has been designed and is being constructed. It will be mounted on the inner divertor during the 2001 shutdown to allow pulse-to-pulse measurements of carbon deposition, an important issue also for the future ITER scenarios.

Reflectometer

This project aims to understand the role of turbulence in the formation of transport barriers. For this purpose, the radial access will be enhanced by including a new set of emitters-receivers in the range 75 –110 GHz (3 correlation channels, one of which will be procured through an international collaboration with Princeton Plasma Physics Laboratory (PPPL)) which will be installed during the 2001 shutdown.

Spectrometer for High Time Resolution Pellet Ablation Measurements

A high time resolution spectrometer is being prepared through an international collaboration (PPPL), in order to measure the properties of the ablation cloud which surrounds injected pellets of frozen deuterium. This will allow physics studies of both the ablation and redistribution processes.

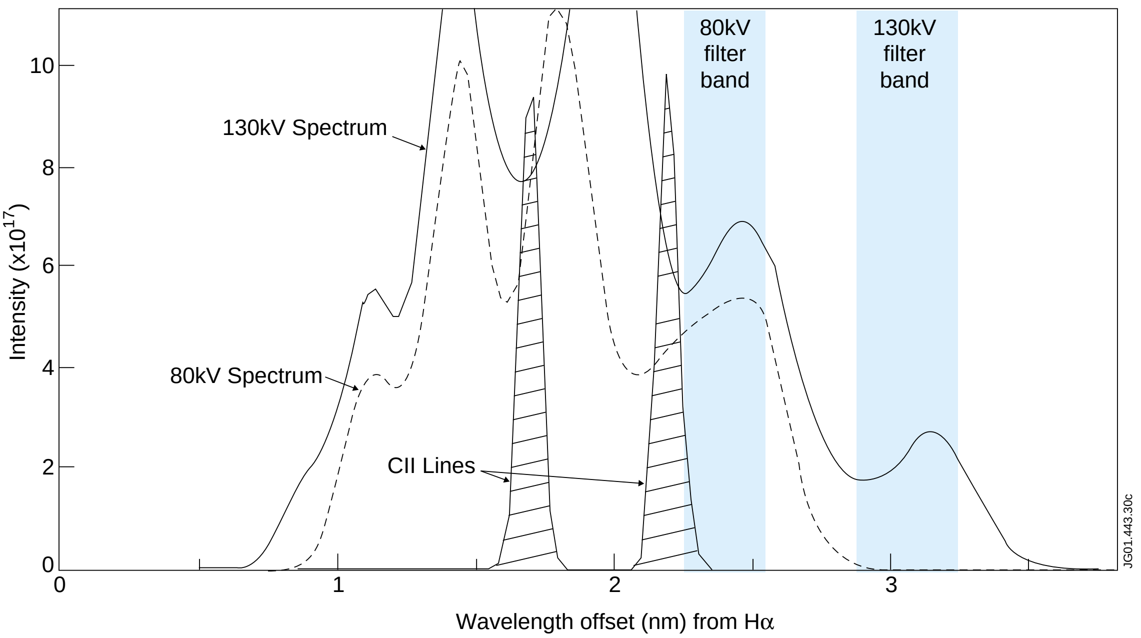


Figure 5.2: The increased doppler shift of the Stark spectrum will give the possibility of: i) having q-profile measurements also during high power phase, because the MSE signal will be clearly discriminated with respect to the signal derived from the other beam sources (with lower energy); ii) the MSE signal measured on edge channels will not be contaminated by carbon lines.

Motional Stark Effect Upgrade

Motional Stark Effect (MSE) diagnostics play a key role in studies of regimes (such as those with ITBs) which require the control of the current profile. This project aims at extending the capability of the MSE measurement of the current profile, by replacing one 80keV/60A PINI by one 130keV/60A PINI on Octant 4 during the 2001 shutdown, and upgrading the optical system. The latter is conducted in the frame of an international collaboration with PPPL. The higher MSE beam energy will allow the MSE signal to be separated from the background radiation originating from other NB sources on Octant 4 and from impurities (fig. 5.2). This, coupled to the improved optics, should provide a dramatic improvement in the MSE performance (increased time resolution; capability to measure current closer to the plasma edge) from the beginning of 2002.

Real Time Control:

A real time monitoring system for sustaining an ITB will be introduced, using electron temperature and density, as well as current profile measurements. This system will progressively be made available during 2002.

ECE Michelson Interferometer:

An ECE Fourier Transform Spectrometer (on loan from FTU Frascati) will improve the time resolution (5 ms) of the absolute electron temperature measurement with 10cm spatial resolution. This system will be available in September 2002.

ECE Heterodyne Radiometer:

The number of radial channels will be doubled from 48 to 96 by using four mixers simultaneously instead of two. This will increase the radial coverage in the measurement of the electron temperature profile to 2.6-3.85m. The simultaneous measurement of the core and edge electron temperatures will be available, as well as low and high field side ITBs with high spatial resolution (~4cm) and temporal resolution (~100μs). This system will be available in September 2002

Li Beam:

The injector will be improved for increased reliability.

HE-Beam:

Remote Control for the helium doping system will be installed on the deuterium neutral beam on Octant 8.

5.1.3 Other Enhancements

Error Field Correction Coils

The main objective of this project is to provide a new set of external coils to produce toroidally asymmetric fields for "error field" and other MHD experiments. The design is complete and manufacture and installation are taking place as part of the 2001 shutdown activities.

D_2 Pellet Upgrade

This project aims at providing a pellet injection capability with a wider range of pellet sizes, inboard or outboard launch, also permitting the pellet size to be varied at constant average fuelling rate. This implies the design, procurement and installation of an additional extruder on the present centrifuge. An extension of the project has been approved to design, procure and install a new quasi-vertical pellet track that would allow higher pellet launch speed for inboard injection. Installation of the new hardware will take place during the 2002 Shutdown and should be operational in September 2002. Significant improvement of the pellet injection capabilities is expected.

Extreme Shape Controller

Studies have been started to investigate the possibility of obtaining extremely shaped plasmas and guaranteeing vertical stabilisation even in the presence of large disturbances, like Edge Localised Modes (ELM), with the existing active circuits and control hardware. If feasible, a control algorithm for the stabilisation and control of highly shaped plasmas will be designed, taking into account the present constraints imposed by the poloidal field system, the diagnostics and the control computer hardware. The proposed controller would be installed during 2002.

5.2 Longer Term Enhancement Projects

5.2.1 Objectives

A significant enhancement of the JET facilities has been studied during 2000-2001, in the frame of the JET Enhanced Performance Project (JET-EP). The project consists of the design, and the possible procurement and installation of new heating systems, new diagnostics, and a new divertor with high triangularity and higher power handling capability.

The following general objectives were proposed and agreed by European Fusion Committees:

- to consolidate the preparation of ITER operating scenarios by increasing the present operating domain of ELMy H-modes (the reference mode of operation of ITER) and further expanding the range of operating scenarios,
- to support design choices in key areas of subsystems of ITER which could be finalised even after the start of construction of ITER.

In more detail, the physics objectives are :

- further studies of the ITER reference scenario, the ELMy H-mode, in preparation of ITER plasma operation. This could be conducted near operating limits, at ITER-like values of H, n/n_{GW}, Z_{eff}, κ, δ, β_N, T_e/T_i, at high field, extended ρ^*, moderate rotation, and with tolerable ELMs, including investigation of the scaling of NTMs and its stabilisation with ECCD.
- establishment and study of integrated optimised shear scenarios suitable for ITER steady state operation. In particular, identify and use triggering processes for barrier formation, implement real-time profile control strategies with the various auxiliary systems (ECCD, ICRF, LHCD, NBI), and demonstrate steady-state potential and performance with a combination of non-inductive current drive schemes.

Significant technological tests in preparation of ITER auxiliary systems would also be provided, in particular:

- the test of an ITER-like ICRH antenna;
- the test of several ITER-relevant ECRH components;
- the development of ITER-relevant and DT diagnostics; and
- plasma wall erosion and co-deposition studies for minimisation of Tritium retention.

The scale of implementation of this project will depend on the Euratom budget during the Sixth Framework Programme (2003-2006) and on the decisions concerning ITER construction and siting. Therefore the activities aim first at conducting all the design and R&D, in preparation for future decisions, while launching the construction of a limited number a components, namely the ITER-like ICRH antenna (Fig 5.3), two gyrotrons and high priority diagnostics which could be operational in the relatively short term.

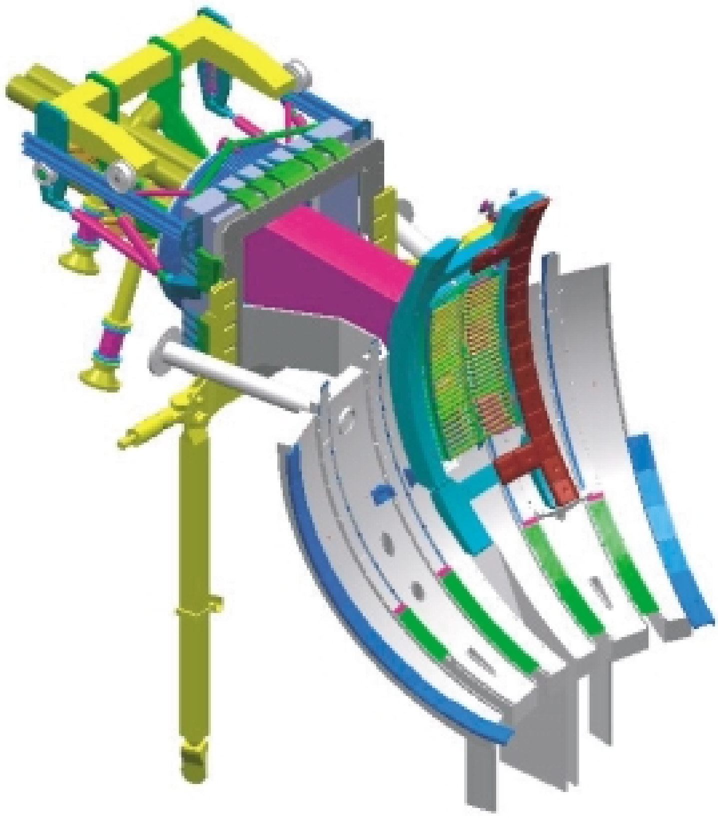


Figure 5.3: ICRH antenna

5.2.2 Scope of the Studies

The systems currently under design are :

- An ITER-like Ion Cyclotron Resonance Heating antenna, capable of coupling 8 MW to the plasma, with modifications of transmission lines and ICRH auxiliaries.
- An Electron Cyclotron Resonance Heating System, composed of six 1MW 113.3GHz Gyrotrons, transmission lines, diamond windows, in vessel launcher with steerable mirror and auxiliaries, capable of delivering about 5 MW into the plasma. This system is compatible with the possible later installation of 170GHz ITER like Gyrotron(s), and includes several ITER relevant technologies.
- A high power divertor (Mark II HP), capable of withstanding up to 50MW of input power into the plasma for up to 10s and compatible with the high triangularity plasma shapes planned in ITER (Fig 5.4).
- A series of new diagnostics, to improve the control of the plasma and the range of data collected during the experiments.

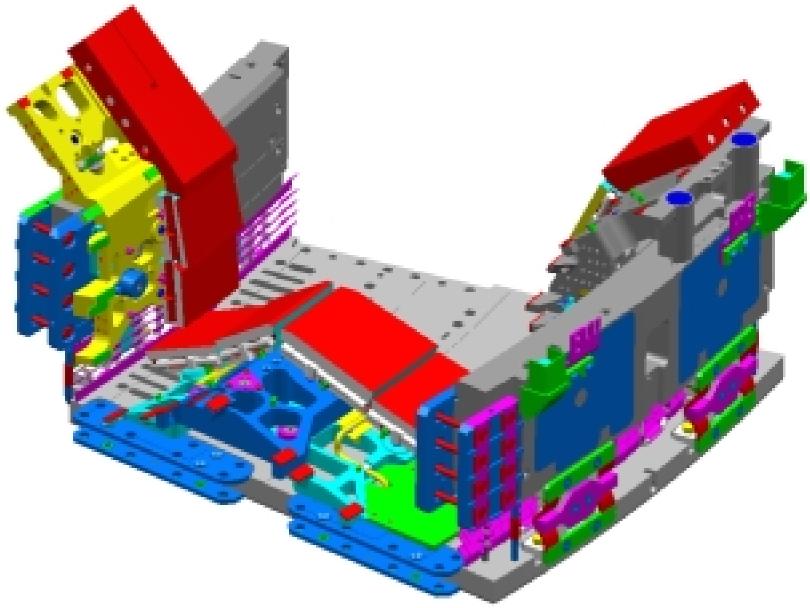


Figure 5.4: MkII HP divertor design

5.2.3 Overview of the Activities

The first semester of 2000 was dedicated to preliminary studies and review meetings. Up to April 2000 a review group (Phase I Ad Hoc Group) was chaired by R. Weynants, ERM/KMS, Brussels. This was followed by a second review group (Phase II type Ad Hoc Group) chaired by C. Alejaldre, CIEMAT, Madrid. This Ad Hoc Group is now in charge of monitoring the project of long term JET Enhancements.

From July 2000 the activities carried out have covered the following areas:

- organisation of the overall coordination and the individual projects (ECRH, ICRH, Divertor): planning, resources, competencies, sharing of responsibilities, project management guidelines, documentation, etc., with the wide involvement of the EFDA Associates and Operator;
- preliminary design of the various sub-systems (gyrotrons, transmission lines, launcher, etc.), identification of the critical points, organisation of R&D activities in support of the design;
- identification of the interfaces between the various sub-systems (including other JET Facilities);
- identifications of the activities to be carried out by the Operator for the installation of the new systems, and for the refurbishment of some critical old/obsolete systems (to ensure an extended period of JET operation during FP6);
- identification of the diagnostics systems: this assessment has been carried out by a Diagnostics Working Group, with experts from the Associates and from the ITER JCT (Joint Central Team). A sub-set of diagnostics has been selected for an immediate implementation, and other diagnostics will be developed in terms of design and R&D, pending decisions concerning procurement and installation;
- the call for tender for the procurement of two gyrotrons (as basic contract) has been issued, with options for procuring four more or six more gyrotrons.

A significant effort is also required from the JET Operator to plan and perform the modifications and adaptations to the JET Facilities, to integrate the new systems, including in-vessel activities. The Operator also provides all the safety and Quality Assessment requirements to the projects. Appropriate Operator involvement is ensured by the nomination of Operator Representatives for each project.

6. International Collaborations

6.1 Introduction

International collaborations are an important part of the scientific work of JET under EFDA and under the JET Implementing Agreement (JIA). They are dealt with either by application of the dispositions of Article 17 (for substantial collaborations that would require accession by other Parties) or by those of Article 18 (collaborations that could be implemented under appropriate existing international agreements).

So far, collaborations have been arranged under Article 18 using IEA (International Energy Agency) Implementing Agreements in the field of fusion, the main one being the IEA Implementing Agreement on Cooperation on the Large Tokamak Facilities.

6.2 IEA Implementing Agreement on Cooperation on the Large Tokamak Facilities

Co-operation between the EU, US and Japan under the IEA Implementing Agreement on Co-operation on the Large Tokamak Facilities (formerly the IEA Implementing Agreement on Co-operation Among the Three Large Tokamak Facilities (JET, JT-60 and TFTR)) has been on-going since 1986 and has been the main Agreement under which collaborations in the framework of the EFDA-JET Workprogrammes have been arranged.

During the period of the report the Executive Committee for this Agreement met twice on 31 May – 1 June 2000 at Culham, UK, with J. Paméla in the chair and on 31 May – 1 June 2001 at Naka, Japan, with H. Ninomiya in the chair. The meetings received reports on the Status and Plans of Each Party, the **Workshops, Personnel Assignments** and the **Task Assignment Programmes** completed during 2000-2001 and considered for the future.

Workshops

Three Workshops were held during the period of the report:

- W43: Transport Barriers at Edge and Core
  (27-30 March 2001; JAERI; T. Takizuka, J. G. Cordey; D. Mikkelsen)
- W45: Fuelling: Core and Edge Density Control
  (30-31 October 2000; PPPL; M. Bell, H. Takenaga, J. Ongena)
- W46: Diagnostics for Burning Plasma Experiments
  (18-21 September 2000; JAERI; T. Sugie, K. Young, J. Sanchez)

Furthermore, three were planned for the period up to June 2002:

- W48: ELMs (June 2002; JET; Matthews, Kamada, Lao)
- W49: Real Time Control of ITB Discharges Approaching Steady State
  (Spring 2002; JT-60; Ushigusa, Stambaugh, Wolf/Hender)
- W50: Electron transport (Spring 2002; US; Synakowski, Garbet/Ryter, Kishimoto)

Personnel Assignments

There were several tens during the period of the Report. These assignments were related to participation in the Experimental Campaigns, analysis of experimental data and provision of diagnostic equipment (e.g. pellet spectrometer, components for MSE sytem).

Task Assignment Programmes

During the year 2000 the work was organised in Task Assignment Programmes in the seven areas indicated in Table 6.1 which also includes the EU, US and Japanese Co-ordinators; Lead Coordinators are **in bold**.

	Area	EU Co-ordinators	US Co-ordinators	Japanese Co-ordinators
I	Research on High-βp and related modes of operation	C. Challis (EU)	**M. Bell** (TFTR)	S. Ishida (JT-60)
II	Disruption studies	O. Sauter (EU)	M. Okabayashi (TFTR)	**R. Yoshino**/ Y. Neyatani (JT-60)
III	Divertor plate technology	**D. Stork** / D. Ciric (EU)		Y. Okumura (JT-60)
IV	Neutral beam current drive research	C. Challis (EU)	L. Grisham (TFTR)	**H. Ninomiya** Y. Kamada (JT-60)
V	Impurity content of radiative discharges	**G. Matthews** / J. Ongena (EU)	K. Hill (TFTR)	N. Asakura (JT-60)
VI	Scaling of access to ITB plasmas	**C. Gormezano** / A. Becoulet (EU)	E. Synakowski (TFTR)	T. Fujita (JT-60)
VII	Remote participation in experiments	V. Schmidt (EU)	**E. Oktay** (USDOE-TFTR) S. Davies (TFTR)	T. Matsuda (JT-60)

Table: 6.1

Table 6.1: EU, US and Japanese Co-ordinators for the seven Areas under the IEA Implementing Agreement on Cooperation amongst the Three Large Tokamak Facilities (up to December 2000)

Since January 2001 Task Assignment Programmes have been reorganised in the five areas indicated in Table 6.2. Table 6.2 also includes the EU, US and Japanese Co-ordinators; Lead Coordinators are **in bold**.

	Area	EU Co-ordinators	US Co-ordinators	Japanese Co-ordinators
I	Transport/ Confinement Studies	J. G. Cordey (UKAEA)	D. Hillis (ORNL)	**Y. Miura (JAERI)**
II	Tokamak Macroscopic Stability	**O. Sauter** (EPFL)	J. Manikam (PPPL)	T. Ozeki (JAERI)
III	Divertor and Plasma Boundary Studies	V. Philipps (FZJ)	**G. Porter** (LLNL)	H. Kubo (JAERI)
IV	Fast Particle and Current Drive Studies	J-M. Noterdaeme (IPP) F. Zonca (ENEA)	**R. Nazikian** (PPPL)	K. Ushigusa (JAERI)
V	Tritium and Remote-Handling Technologies	**R. Laesser** (FZK)	S. Willms (LANL)	T. Konishi (JAERI)

Table: 6.2

Table 6.2: EU, US and Japanese Co-ordinators for the five Areas under the IEA Implementing Agreement on Cooperation on the Large Tokamak Facilities (since January 2002)

6.3 Bilateral Agreement between the EU and the US

Co-operation with entities, which are not part of EFDA, can also take place in the frame of a Bilateral Agreement concluded between the European Communities and other third States. On 14 May 2001 Commissioner Busquin and Secretary Abraham signed a Bilateral Agreement between the EU and the US. The mission for the EU/US collaboration on JET for the next 2-3 years is to resolve mutually-agreed key physics issues by exploiting synergies between the US and the EFDA-JET programmes and contributing to the achievement of the goals of both the EU and the US programmes. The aim is to develop in a coordinated manner understanding, predictive tools, and control tools (diagnostics and actuators) for improving performance and for providing a basis for Next Steps in the five areas listed in Table 6.3. These Areas are slightly different from those used under the IEA Implementing Agreement on Cooperation on the Large Tokamak Facilities (Table 6.2).

On the EU side, this responsibility for coordination resides with the EFDA-JET Task Force Leaders; on the US side, with Key Co-ordinators and Deputy Co-ordinators, whose names are given in Table 6.3; Lead Co-ordinators are shown **in bold**.

To reinforce the scientific co-ordination of the specific topics of mutual benefit to the EU and the US, the Co-ordinators would:

- co-ordinate the European and US teams to perform programmes which, within Europe, had already been agreed by the EFDA-JET Subcommittee and the Associates; and
- report at an annual scientific co-ordination meeting.

Area		EU **Co-ordinator**		US **Key**/Deputy Co-ordinator	
A	Performance-Limiting Edge	A	**TFL S1 (J. Ongena)** in close collaboration with TFLs E (G. F. Matthews) and M (T. Hender)	A	**A. Hubbard** (MIT) / J. Strachan (PPPL)
B	Understanding ITBs and their control	B	**TFL S2 (R. Wolf)**	B	**P. Gohil** (GA) / B. Stratton (PPPL)
C	Understanding NTMs and their control	C	**TFL M (T. Hender)** in collaboration with TFL H (A. Tuccillo)	C	**R. LaHaye** (GA) / C. Hegna (UW)
D	Energetic Particle Driven Modes, and extrapolation to reactor-grade plasmas	D	**TFL H (A. Tuccillo)**, in collaboration with TFLs M (T. Hender) and DT (D. Stork)	D	**R. Nazikian** (PPPL) / B. Breizman (UT)
E	Minimisation of Tritium Inventories, and extrapolation to reactor-grade plasmas	E	**TFL FT (R. Laesser)**, in close collaboration with TFLs E (G.F. Matthews) and DT (D. Stork)	E	**C. Skinner** (PPPL) / S. Willms (LANL)

Table: 6.3

Table 6.3: EU and US Co-ordinators for the Areas under the Bilateral Agreement between the EU and the US

6.4 Participation to Enhancements

While the exchange of staff mentioned in section 6.2 (Personnel Assignements) was mainly focused towards the participation of collaborating institutions in the JET experimental Campaigns, US laboratories have contributed to the enhancement of the facilities mainly through 3 diagnostic projects which shall be operational in 2002 (see section 5.1):

- Reflectometer (PPPL)
- Spectrometer for high time resolution pellet ablation measurements (PPPL)
- Motional Stark Effect Upgrade (PPPL)

In addition PPPL and ORNL are involved in preparing the high power prototype for the ITER-like ICRH antenna project (see section 5.2).

7. Appendices

7.1 The ITER Project

Theoretical and experimental fusion research programmes in Europe and throughout the world have shown impressive progress in efforts to create the conditions for a fusion power plant. The next major step on the path towards confirming fusion as a practical long-term energy source, with acceptable environmental characteristics, is to construct and operate a burning plasma experiment that allows, in a single device, full exploration of the physics issues and testing of key technological features of fusion power stations. ITER would provide the basis for the design of a first demonstration fusion power station that would prove the feasibility of reliable generation of electricity, leading to the development of a prototype commercial power plant.

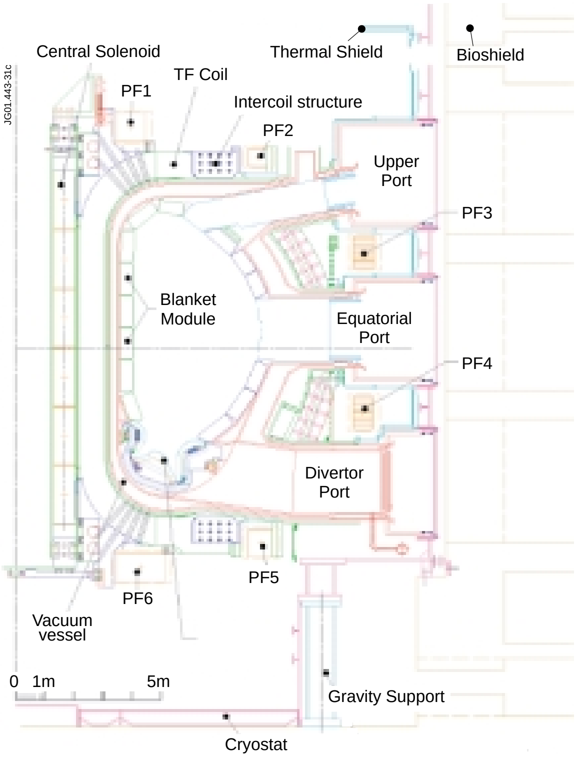
The ITER design was developed within an international collaboration between the fusion programmes of the European Union, Japan and the Russian Federation. This is described in the Final Design Report, issued in July 2001. The physics goals for the device are:

(i) achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary heating power (Q) of at least 10 for a range of operating scenarios and with a duration sufficient to achieve stationary conditions on the timescales characteristic of plasma processes;

(ii) aim at demonstrating steady-state operation using non-inductive current drive with the ratio of fusion power to input power for current drive of at least 5;

(iii) allow the possibility of exploring controlled ignition under favourable confinement conditions.

The technological aims include the demonstration of integrated operation of technologies essential for a fusion reactor, the testing of components for a future reactor and the testing of concepts for a tritium breeding module.



PARAMETER	ITER
κ_{95} κ_X	1.70, 1.85
δ_{95} δ_X	0.33, 0.49
R, a (m)	6.20, 2.0
R/a	3.1
Vol (m^3)	828
B (T)	5.3
I_p (MA)	15.0
t_{burn} (s)	≥300
$<n> / n_{GW}$	0.85
$<n>$ (10^{20} m^{-3})	1.01
$<T_e>$, $<T_i>$ (keV)	8.8, 8.0
$Z_{eff,axis}$	1.69
$n_{He,axis} / n_e$ (%)	4.3
β_N	1.8
β (%)	3.5
P_{fus} (MW)	400
L_{wall} (MW m^{-2})	0.47
Q	10

JG01.443-14c

Figure 7.1: Poloidal cross-section of ITER, with a table of the main device and plasma parameters in the Q=10 inductive reference scenario

7.1.1 The ITER Device

The reference operating scenario for ITER is the ELMy H-mode and the projection of plasma performance to the ITER scale are those established in the ITER Physics Basis, which has been developed from broadly based experimental and modelling activities within the magnetic fusion programmes of the ITER Parties. The design is therefore based on physics constraints on the plasma operating window derived from the ITER Physics Basis, combined with engineering constraints which have been well established in the course of the ITER Engineering Design Activities. Figure 7.1 shows the poloidal cross-section of the device together with the principal tokamak and plasma parameters in the Q=10 reference scenarios.

To provide a capability for long pulse, and ultimately steady-state, operation, ITER utilizes a superconducting toroidal and poloidal field coil system (Fig. 7.2). In addition, actively cooled plasma facing components are required and a mixture of materials is used to satisfy differing requirements for different areas of the first wall: beryllium is used on the main chamber wall to minimize plasma impurity contamination and tritium retention in the vacuum vessel; carbon fibre composite forms the high heat flux areas of the divertor target, as it is resilient to transient heat loads associated with ELMs and disruptions; and tungsten is applied in the remainder of the

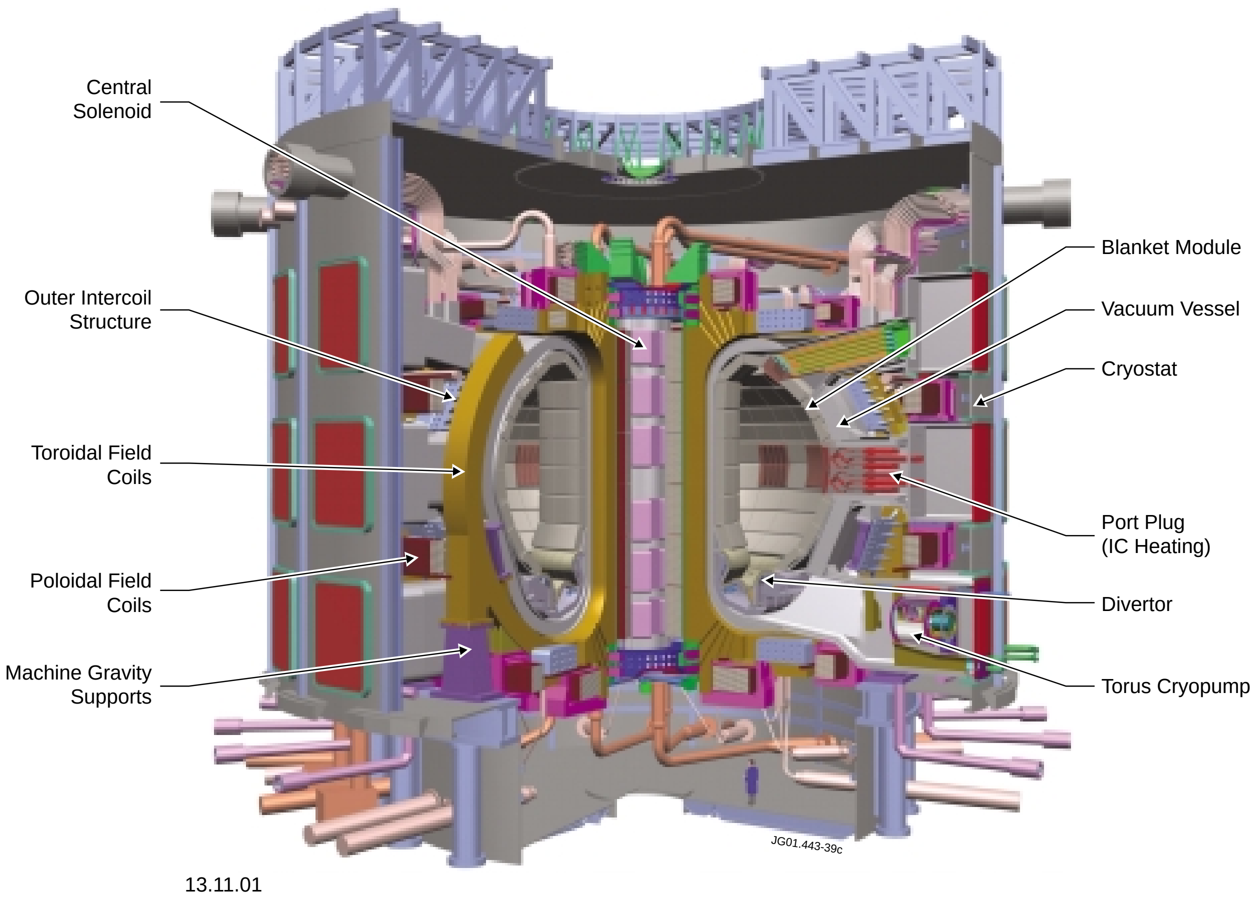


Figure 7.2: Overview of the ITER device showing the main features.

divertor, where robustness to high particle fluxes is required. The design of the divertor, with vertical targets, tight baffling, and a “transparent” dome between the divertor legs, through which particles are exhausted, is based on the results of the extensive experimental studies on divertor geometry performed during the 1990s. Heating and current drive power is provided by a mixture of negative ion based neutral beam injection at 1MeV, ion cyclotron radio frequency heating in the frequency range 35-55MHz and electron cyclotron heating at 170GHz. A comprehensive set of plasma diagnostics, for both plasma control and physics analysis, is also foreseen.

Performance calculations indicate that the ITER design should have a significant margin, not only for Q=10 operation, but for a range of Q values. This is illustrated in Fig. 7.3, which shows the results of a 0-D calculation of the operating window for Q=10, delimited by the boundaries, $P_{loss} = 1.3P_{LH}$, $n = n_{GW}$, and $\beta_N = 2.5$, for operation in the nominal 15MA scenario with $q_{95}=3$. The fusion output power is in the region of 200-700MW (for a confinement enhancement factor relative to the the scaling law $H_{98(y,2)} = 1$), corresponding to an average neutron wall loading, L_{wall}, of 0.23-0.80MWm^{-2}, so that the device retains a significant capability for technology studies, such as tests of tritium breeding blanket modules. Similar calculations confirm, as expected, that for operation at 17MA (i.e. $q_{95}=2.6$) the design can attain Q = 10 for lower values of $H_{98(y,2)}$ (~0.75) and that there is a considerable expansion of the Q=10 operating window.

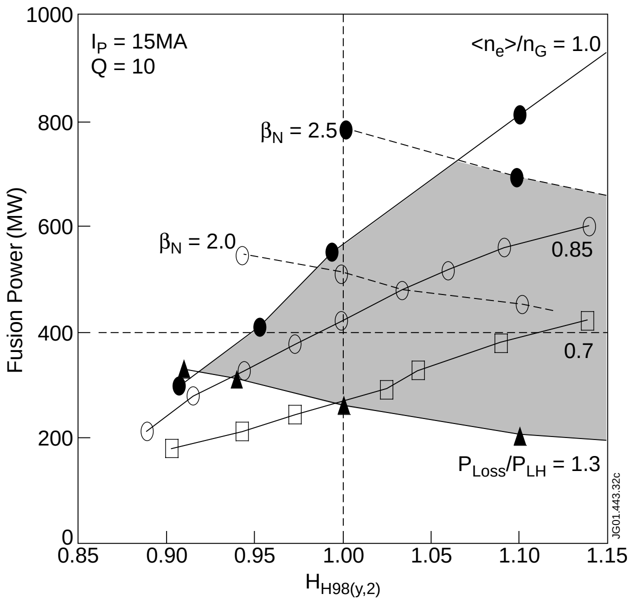


Figure 7.3: Predicted Q=10 operating window (shaded area) for the reference inductive scenario at 15MA in ITER.

Plasma scenarios which might provide the basis for steady-state operation of a tokamak power plant remain the subject of intense research activity and there is gradual progress in this area, but a precise specification of the optimum conditions for steady-state operation in ITER is not yet possible. Nevertheless, the requirements for regimes allowing steady-state operation with Q=5 have been explored numerically and in general this requires an enhanced confinement regime ($H_{98(y,2)}>1$, $\beta_N \geq 4\ell_i$). Less demanding constraints apply to the so-called “hybrid” mode of operation, in which non-inductive currents are exploited to extend the inductive pulse-length to periods of ~2000s while maintaining Q = 5.

7.1.2. ITER Technology Research and Development

The scale and nature of the ITER device, involving, for example, high neutron and heat fluxes in quasi-steady-state operation, distinguish it from any existing toroidal confinement experiment and make significant new demands on fusion technologies. Moreover, the aim of the ITER programme to integrate the study of burning plasmas with the demonstration (at last proof of principle) of many key technologies required for a fusion power plant establishes a requirement for a substantially improved technological capability in several areas. Major R&D studies have therefore been implemented in seven large projects, which have addressed the major technological issues and demonstrated the ability of the ITER Parties to implement and manage large scale projects involving the integration of components manufactured by several of the Parties. The activities have not only confirmed that the required level of performance can be achieved, but also validated manufacturing techniques, provided confidence in the quality assurance applied and supported manufacturing cost estimates for larger cost items.

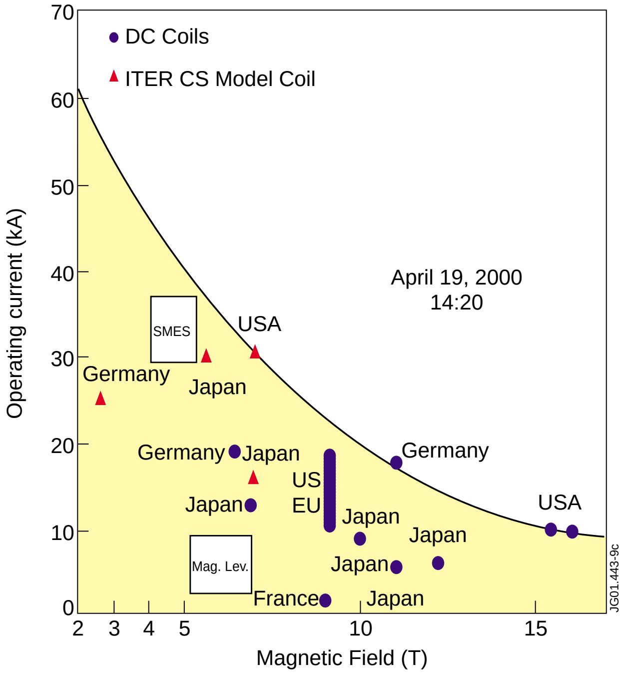


Figure 7.4: Comparison between the parameters demonstrated in the Central Solenoid Model Coil performance tests and the parameters of other major superconducting coils.

Figure 7.5: The Toroidal Field Model Coil ready for installation in the TOSKA test facility (Karlsruhe).

The achievements of the Technology R&D programme are exemplified by the progress in magnet technology which has been demonstrated in the Central Solenoid and Toroidal Field Model Coil projects (CS MC and TF MC). The former project involved the fabrication and testing of an Nb_3Sn superconducting magnet with parameters (diameter of 3.6m and height of 2m) approximately equal to one of the six modules in the ITER central solenoid. This coil, the largest high field pulsed superconducting magnet constructed, has achieved a world record with a maximum field of 13T and a stored energy of 640MJ (Fig. 7.4). It has been safely dumped with a time constant as short as 6s. The TF MC (Fig. 7.5) is an Nb_3Sn coil of smaller overall dimensions (3m×4m) than the ITER TF coil (9m×13.5m), but has a similar cross-section. Fabrication of this coil is also complete and it has achieved its design parameters: a peak field of 9.7T (ie as a single coil) at a cable current world record of 80kA, compared to the ITER operating parameters of 11.8T at 68kA.

7.2 Nuclear Fusion Basics

7.2.1 The Nuclear Fusion Reaction

Nuclear Fusion is the energy-producing process which takes place continuously in the sun and stars. In the core of the sun at temperatures of 10-15 million degrees Celsius, Hydrogen is converted to Helium. A tiny fraction of the energy released in this process is enough to sustain life on earth.

For energy production on earth different fusion reactions are considered. The reaction which would be easiest to achieve occurs between the nuclei of the two heavy forms (isotopes) of Hydrogen, namely Deuterium (D) and Tritium (T). At a later stage reactions involving just Deuterium or Deuterium and Helium (^3He) may be used.

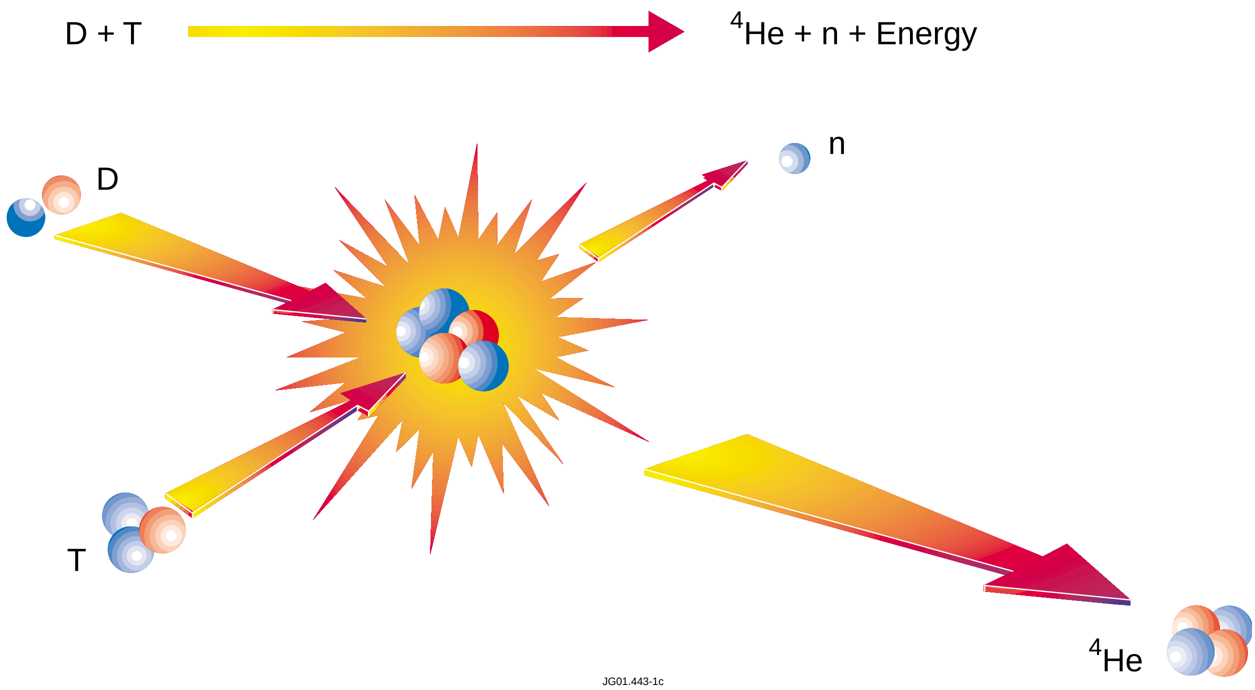


Figure 7.6: The most easily achievable fusion reaction

At the temperatures required for the D-T fusion reaction - over 100 Million deg. C - the fuel has changed its state from gas to plasma. In a plasma, the negatively charged electrons are separated from the atomic nuclei. The stripped atomic nuclei -"ions"- are positively charged. Understanding plasma required major developments in physics. Plasmas are now used widely in industry, especially for semi-conductor manufacture.

7.2.2 Advantages of Energy Generation by Fusion

- Fuels are plentiful.
- Inherently safe since any malfunction results in a rapid shutdown.
- No atmospheric pollution leading to acid rain or “greenhouse” effect.
- No radioactive ash.
- Radioactivity of the reactor structure, caused by the neutrons, is expected to decay rapidly and can be minimised by careful selection of low-activation materials. Provision for geological time-span disposal might be very limited.

Fuels

Deuterium is abundant as it can be extracted from all forms of water. If all the world's electricity were to be provided by fusion power stations, Deuterium supplies would last for millions of years.

Tritium does not occur naturally and will be manufactured from Lithium within the facility.

Lithium, the lightest metal, is plentiful in the earth's crust. If all the world's electricity were to be provided by fusion, known reserves would last for at least 1000 years. Using Lithium from sea water would extend the use of Lithium for hundreds of thousand years.

Although the reaction occurs between Deuterium and Tritium, the consumables are *Deuterium and Lithium: transportation of radioactive products would be very limited*.

Quantities

For example, 10 grams of Deuterium which can be extracted from 500 litres of water and 15 grams of Tritium produced from 30 gram of Lithium would produce enough fuel for the lifetime electricity needs of an average person in an industrialised country.

7.2.3 Plasma Confinement

Since a plasma consists of two types of charged particles, positively charged ions and negatively charged electrons, magnetic fields can be used to isolate the plasma from the vessel walls. In a magnetic field the charged particles readily spiral along the field lines but diffuse only slowly across them. The most promising magnetic confinement systems are toroidal (ring-shaped) and, of these, the most advanced is the Tokamak. JET is the largest Tokamak in the world (see section 1.2).

7.2.4 Conditions for a Fusion Reaction

Temperature

Fusion reactions occur at a sufficient rate only at very high temperatures. Over 100 million deg. C is needed for the Deuterium-Tritium reaction whilst other reactions require even higher temperatures.

The hot plasma must be well isolated from material surfaces in order to avoid cooling the plasma and releasing material atoms (impurities) that would contaminate and further cool the plasma, as well as eventually erode the vessel. In the Tokamak system, the plasma is isolated by magnetic fields.

Energy Confinement

The efficiency of the magnetic isolation is measured by a quantity called the *Energy Confinement Time*. This is the characteristic time-scale for plasma cooling when the source of heat is removed.

Density

The density of fuel ions must be sufficiently large for fusion reactions to take place at the required rate. The fusion power generated is reduced if the fuel is diluted by impurity atoms released from surrounding material surfaces or by the accumulation of Helium "ash" from the fusion reaction. As fuel ions are burnt in the fusion process they must be replaced by new fuel and the Helium ash must be removed.

Numerical values for D-T Reaction

- Plasma temperature: (T) 100-200 million deg. C
- Energy Confinement Time: (τ_E) 1-2 seconds
- Central Plasma Density: (n) 2-3 x 10^{20} particles m^{-3} (approx. 1/1000 gram m^{-3})

Fusion Product

Overall the required conditions are measured by a product of these three values called the Fusion Product.

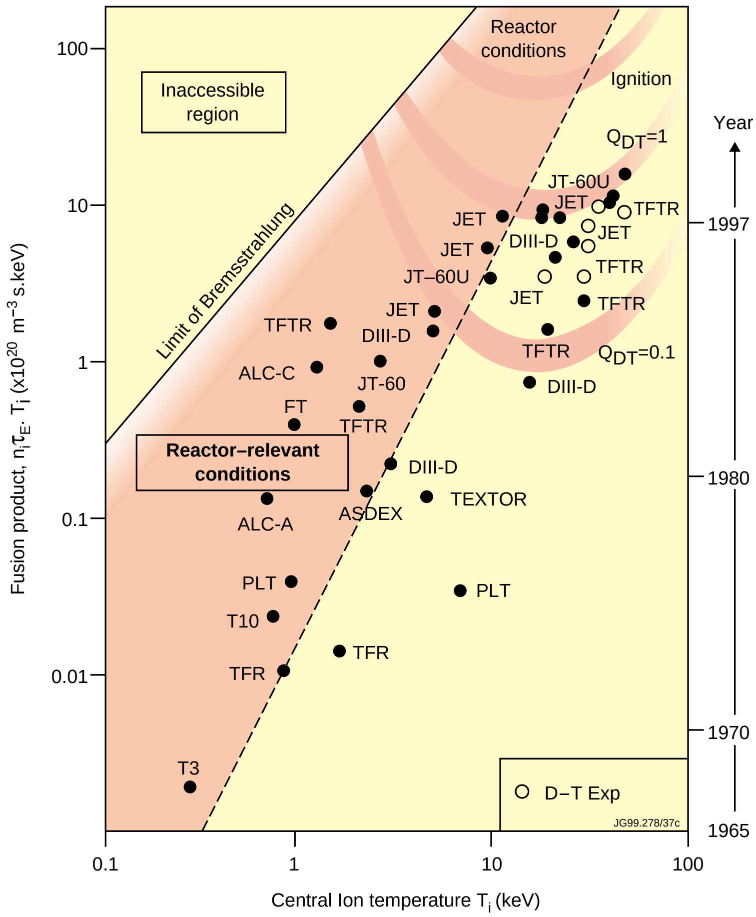


Figure 7.7: Shows that excellent progress has been made by JET towards achieving the required reactor conditions

7.2.5. Heating of Plasma

In JET, the following methods are used:

Ohmic Heating

Currents up to 7 million amperes (7MA) flow in the plasma and deposit a few mega-watts of heating power.

Neutral Beam Heating

Beams of deuterium or tritium ions, accelerated by a potential of up to 140,000 volts, are neutralised and then injected into the plasma. The accelerated beams have to be neutralised in order to penetrate the confining magnetic field. In the plasma, the beams become ionised and the fast ions give up their energy to the plasma by collision. At present, the maximum power available at JET is 21MW.

Radio-Frequency Heating

The plasma ions and electrons rotate around in the magnetic field lines of the tokamak. Energy is given to the plasma at the precise location where injected radio waves resonate with the ion rotation. Eight antennae in the vacuum vessel propagate waves in the frequency range of 25-55 MHz into the core of the plasma to increase the energy of the ions. At JET this method can inject up to 20MW of heating power.

Current Driven by Microwaves

Microwaves at 3.7 GHz have accelerated the plasma electrons to generate a significant fraction of the plasma current. The name of the method, Lower Hybrid Current Drive (LHCD), refers to the particular waves excited in the plasma. Up to 5 MW of LHCD have been coupled to the plasma.

Self Heating of Plasma

The helium nuclei (alpha particles) produced when deuterium and tritium fuse remain within the plasma's magnetic trap. Their energy continues to heat the plasma to keep the fusion reaction going. When the power from the alpha-particles is sufficient to maintain the plasma temperature, the reaction becomes self-heating - a condition referred to as ignition.

7.3 Remote Participation

The staff involved in the EFDA experimental campaigns are at Culham for limited periods. Extensive use of "Remote Participation" (RP) techniques has been necessary to allow their involvement (e.g. in experimental or diagnostics support) before, during and after an experiment when they are at their home laboratory. In particular, JET data need to be analysed routinely by scientists remotely either on JET-based computers or on computers at the EFDA Associate laboratories. Effective collaboration between scientists also requires extensive use of teleconferencing (see Fig.7.8). All these aspects are dealt with in the frame of the Remote Participation activities. It should be pointed out, however, that the control of the equipment remains exclusively from the main control room on the JET site, although session preparation can be undertaken both on and off-site.

The scope of RP activities takes into account that remote users need to have access to:

- JET Data,
- programme information,
- real-time information during JET pulsing,
- discussion with colleagues,
- remotely shared presentations and seminars.

In order to meet the technical challenge of RP for the Campaigns C1-C4, a Co-ordinator (Volker Schmidt, RFX Padova) and Deputy (John How, CEA Cadarache) were appointed; and work was closely co-ordinated with the Operator and the EFDA Associates. The definition of the work required was the result of "Users Group" meetings, plus close daily contact with the users and the members of the Task Forces. The technical needs raised by the Associate laboratories were channelled through local "Technical Contact Persons".

Figure 7.8: Two scientists involved in Telecollaboration

Work was organised along the following topics :

- Remote Computer Access: providing secure, controlled access into the JET computer system.

- Remote Data Access: providing automatic transfer of data from JET into the off-site user's analysis programs.

- Telecollaboration: implementing appropriate communications: dialogues, status broadcasts, tele-meetings and seminars (see Fig.7.8).

- Network Traffic: continuous monitoring, and intervention where necessary.

- Security: both at Culham and throughout the Associates.

- Support: Help Desk and longer-term problem or development areas.

- Visitor facilities at Culham: integration with minimum perturbation.

- Documentation: web site, news sheets, publications and demonstrations.

Campaigns C1-C4 have produced a solid definition of tools and solutions required to meet the immediate needs. Solutions and tool-kits were chosen, where possible, to be platform-independent, free- or share-ware, and non-specialist in order to fit into the widely varying computing environments within the Associates. Most of these techniques have immediately been made available to users as they were introduced, albeit sometimes under ad-hoc or test arrangements.

Although new in the field of "Remote Participation", the European Fusion community is rapidly moving to the forefront of this challenging subject. The future operation of the JET Facilities, as well as the wider application to ITER and other inter-Association collaborations, should lead to a much more routine use of these RP techniques, and the establishment of permanent arrangements and support in the Associations.

7.4 Important Visitors

During the period of this report over 1750 external visitors have viewed the EFDA-JET facilities*. Fig 7.9 shows a breakdown of these visitors in categories.

Some of the most notable visits were by groups from the prospective ITER site candidate countries, France, Canada, Spain and Japan. In particular ITER Canada sent several delegations including a number of Canadian MPs.

Among MEPs who have also visited were Terry Wynn, Chairman of the Budgets Committee, Peter Skinner, a member for the South East Region, Esko Seppanen from Finland and Konstantinos Alyssandrakis from Greece.

As part of the Educational Outreach programme nearly 700 school children visited in small parties or as part of special schools events. One such outstanding event was organised as part of the Campaign to Promote Engineering. A leading Science television personality Johnny Ball gave a series of talks to the children, with the aim of interesting them in a career in Engineering, followed by visits to the facilities where children had an opportunity to try the remote handling system developed for JET.

The media has continued to show a high level of interest in JET and many film crews and journalists have toured the facilities and interviewed staff. These included a party from South Korea, whose visit resulted in a number of television documentaries and coverage in the national financial newspaper.

EFDA JET also staged exhibitions at the Madrid SOFT Conference and the IAEA Conference in Sorrento; demonstrations of remote participation with direct links to JET were organised at the events in collaboration with Padova and Cadarache.

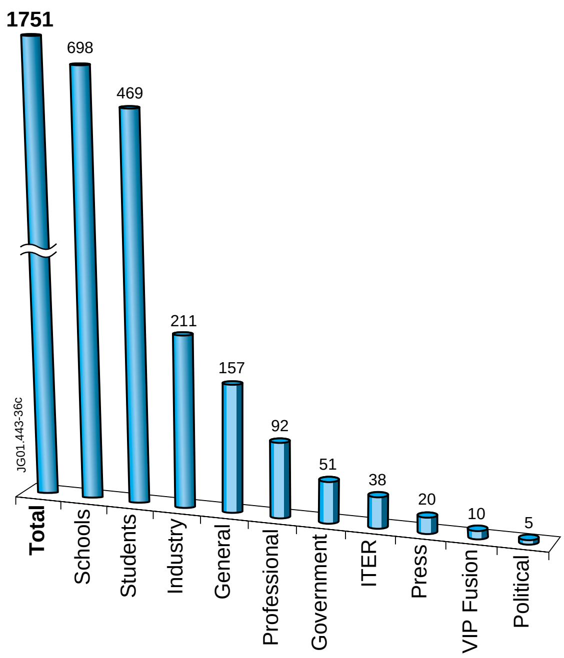


Figure 7.9: Visitors to EFDA-JET January 2000 - March 2001

*The majority of these visits were organised and managed by the UKAEA on behalf of EFDA.

7.5 List of Publications

Publications are the main “output” of all S/T tasks conducted on the JET Facilities. In order to make sure that new results are timely and appropriately advertised, that the quality is ensured and that all contributors are adequately acknowledged a clearance procedure has been set up. This procedure is derived from the EFDA Publication rules which provide the framework. Detailed rules, instructions, templates and tools guide the authors through the procedure.

The following list presents all papers submitted or to be submitted to journals and the contributions to past as well as submissions to future conferences. The highlight so far has certainly been the EPS Conference in June 2001 with over 90 JET-related contributions.

All publications and conference contributions will appear on the EFDA-JET publications web site operated by the IoPP. While still in a draft version, they are systematically displayed on the restricted access internal web site in order to allow wide scientific discussions prior to finalisation.

EFDA-JET Publications List

The following is a list of papers cleared for publication by the EFDA-JET Associate Leader. Note that the papers have not necessarily been published yet, nor been presented at the relevant conference.

Papers Presented at Conferences

Joint Varenna-Lausanne Workshop, Varenna, Italy, 28-31 August 2000

MHD limiting JET optimised shear performance, T C Hender

Transport Task Force Workshop, Varenna Italy, 4-7 September 2000

On the relationship between q-profile, heating and ITBs in JET optimised shear plasmas, Y F Baranov

On the link between ExB sheared flows and rational surfaces in fusion plasmas, C Hidalgo

Symposium on Fusion Technology - 2000, Madrid, Spain, 11-15 September 2000

The error field correction coils on the JET machine, I L Barlow et al

Operational experience with a variety of plasma facing tile assemblies at JET, P K Edwards et al

Experience and lessons from the JET 4.0T assessment, M Gasparotto

Modelling ion source non-uniformity, D J Godden

Remote participation at JET from DRFC Cadarache, J A How et al

Remote participation technical infrastructure for the JET facilities under EFDA, J A How et al

Tritium off-gassing trials on dust and flakes from the JET MKIIA divertor, S J Knipe et al

Neutron Activation Studies on JET, M J Loughlin et al

JET under EFDA: Organisation and recent results, J Pamela

Comparisons of accrued and expected radiation doses to personnel during manual access to the JET vessel, B Patel et al

The audio and visual communications systems for suited engneering activities on JET, R J H Pearce

Vacuum pumping developments on the JET tokamak, R J H Pearce et al

Tritium depth profile in graphite and carbon fibre composite material pre-exposed to tokamak plasmas, R D Penzhorn

Classical and fracture mechanics analysis of the tails of the JET TF coils, V Riccardo et al

CODAS Object Monitoring Service, M R Wheatley et al

9th Int Workshop on Carbon Materials, Munich, 18-19 September 2000

Spectroscopy of hydrocarbon fluxes in the JET divertor, M F Stamp

18th IAEA Fusion Energy Conference, Sorrento, 4-10 October 2000

Direct measurements of damping rates and stability limits for low frequency MHD modes and Alfven eigenmodes in the JET tokamak, A F Fasoli et al

Dynamics of Runaways in JET, R D Gill

Neoclassical Tearing Mode Studies in JET, T C Hender

Pellet Fuelling and ELMy H-mode Physics at JET, L D Horton

Drift-/kinetic Alfven eigenmodes in high performance tokamak plasmas, A Jaun et al

Radiating Edge Plasma Experiments on JET, G P Maddison

Energetic Particle Physics and MHD Stability in JET and START, K G McClements

ITER shaping and elongation experiments in JET, D C McDonald

Overview of recent JET results and future perspectives, J Pamela

Role of Magnetic Configuration and Heating Power in ITB Formation in JET, V Parail

Core and Edge Confinement Studies with Different Heating Methods in JET, F G Rimini

Internal Barrier Discharges in JET and their sensitivity to Edge Conditions, A C C Sips

Combined workshop of the ITER Physics Expert Groups (Pedestal physics, divertor, MHD, disruption control), Garching, 11-13 October 2000

Impact of ELMs on the divertor in AUG and JET, G F Matthews

ITER Energetic Particles, H and CD exp. group, Frascati, 11-13 October 2000

Preliminary results of TF-H(C1-C2), A A Tuccillo

Internat. workshop on heating in tokamaks and transport, Frascati, 11-13 October 2000

JET database on the relationship between q-profile, heating and ITBs, Y F Baranov

42nd Annual Meeting of the Division of Plasma Physics (DPP) of the APS; 10th International Congress on Plasma Physics (ICPP), Quebec , 23-27 October 2000

Tokamak integrated scenarios based on internal transport barriers., A Becoulet

Influence of triangularity and gas fuelling on energy losses in ELMs on JET, M Becoulet et al

Comparison of RF-heated and NBI heated ELMy H-Mode plasmas in JET, R V Budny et al

Neoclassical tearing modes in JET, R Buttery et al

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MHD stability of advanced tokamak scenarios with reversed central current: an explanation of the 'Current Hole', G T A Huysmans et al

Alpha tail production with ion cyclotron resonance heating of He beam ions in JET plasmas, M J Mantsinen et al

Control of neoclassical tearing modes by sawtooth control, O Sauter et al
A dimensionless criterion for characterising internal transport barriers in JET, G Tresset et al

Physicalia Magazine

From JET to ITER: preparing the next step in fusion research, J Pamela et al

Physics Letters A

MHD spectroscopy through detecting toroidal Alfvén eigenmodes and Alfvén wave cascades, S Sharapov et al

Physics of Plasmas

Damping of relativistic electron beams by synchrotron radiation, F Andersson et al

Poloidally asymmetric distribution of impurities in Joint European Torus plasmas, H Chen

Modelling of diamagnetic stabilization of ideal MHD eigenmodes associated with the transport barrier, G T A Huysmans et al

High density high performance ELMy H.mode plasmas in the Joint European Torus (JET), W A Suttrop et al

On the measurement of toroidal rotation for the impurity and the main ion species on the Joint European Torus, D Testa et al

Plasma Physics and Controlled Fusion

Neutral beam stabilization of sawtooth oscillations in JET, C Angioni et al

Local conditions for ion ITB formation in JET, Y F Baranov et al

JET progress towards an advanced mode of ITER operation with current profile control, A Becoulet

ELMs behaviour and edge plasma stability in JET, M Becoulet et al

Microturbulence and flow shear in high-performance JET ITB plasma, R V Budny et al

Influence of the q-profile shape on plasma performance in JET, C D Challis et al

Effect of q-profile modification by LHCD on internal transport barriers in JET, C D Challis

A possible explanation for doubly peaked profiles at the divertor target in JET, A Chankin

Turbulence behaviour during electron heated reversed shear discharges in JET, G Conway et al

Energy Confinement in Steady State ELMy H-modes in JET, J G Cordey

Interpretation of recent power width measurements in JET MkIIGB ELMy H-modes , W Fundamenski et al

The formation and evolution of extreme shear reversal in JET and its influence on local thermal transport, N C Hawkes et al

Neutron emission from JET DT plasmas with RF heating on minority hydrogen, H O I Henriksson et al

Electron heated Internal Transport Barriers in JET, G M D Hogeweij et al

The physics and modelling of macro-instabilities in high performance tokamaks, A Jaun et al

q=1 ITB experiments in JET and comparison with ASDEX-Upgrade, E Joffrin et al

Towards fully non-inductive current drive operation in JET, X Litaudon et al

ELM moderation in high density H-modes on JET and Alcator C-Mod, G P Maddison et al

Analysis of ion cyclotron heating and current drive at $\omega \sim 2\omega_{ch}$ for sawtooth control in JET plasmas, M J Mantsinen et al

The effect of CD4 puffing on the peripheral scrape-off layer in JET, G F Matthews et al

Experimental and simulated VUV spectra from the JET tokamak and the reversed field pinch RFX, M Mattioli et al

Real-time control of internal transport barriers in JET, D Mazon et al

Recent progress on JET towards the ITER reference Mode of operation at high density, J P H E Ongena et al

Reconstruction of the power deposition profiles using JET MKIIGB thermocouple data for ELMy H-mode plasmas, V Riccardo

Edge issues in ITB plasmas in JET, Y Sarazin et al

The role of axisymmetric reconnection events in JET discharges with extreme shear reversal, B Stratton

ITM Formation in terms of w sub (ExB)flow shear and magnetic shear s on JET, T J J Tala

Impact of Different Preheating Methods on q-profile Evolution in JET, T J J Tala et al

Spatial Resolution of the Electron Cyclotron Emission at JET, V Tribaldos

Particle recirculation studies in JET, E Tsitrone et al

Review of Scientific Instruments

Neural networks for real time determination of radiated power in JET, O Barana et al

Ionisation balance in EBIT and tokamak plasmas, N J Peacock

7.6 List of Seminars

From January 2000 till 31 March 2001 41 JET EFDA Seminar meetings were held, occasionally with more than one presentation per meeting. The purpose of the seminars is principally the dissemination of information. Presentations for international meetings are usually rehearsed; visitors from abroad often contribute talks. More recently (notably, in preparation of the EPS 2001 meeting) some seminars have been broadcast from Culham, and in some cases they were broadcast to JET from European laboratories and then re-broadcast. When available, copies of the transparencies are placed on the website http://users.jet.efda.org/ under "seminars".

The following Seminars were presented during the period covered by this report:

- Tuesday 25th January 2000
  1) "Introduction of Seminar Series", J Pamela,
  2) "ICRF induced rotation in tokamaks", R White, PPPL.
- Monday 21st February 2000
  1) "Filamentation of the RTP tokamak plasma", MNA Beurskens,
  2) "Electron thermal transport barriers in RTP", MR de Baar.
- Monday 6th March 2000 "Recent results from DIII-D and future plans" Tony S Taylor, GA.
- Monday 20th March, 2000
  1) "Overview of JET-EP proposal" J Pamela,
  2) "Operation up to 6MA, 4T" Peter Lomas,
  3) "JETTO simulation of JET-EP scenarios" Vasili Parail.
- Monday 3rd April, 2000 "Filamentation of the RTP tokamak plasma" M.N.A. Beurskens, FOM, Netherlands.
- Monday, 10th April, 2000 "MAST and the evolution of the Spherical Tokamak" A. Sykes, UKAEA Fusion.
- Tuesday 2nd May, 2000 "The H-mode: Pedestal stability, scaling, and the L-H transition." Tony S. Taylor, GA.
- Monday 22nd May, 2000 "Report from Task Forces S1 and S2" J. Ongena, A. Becoulet.
- Monday 5th June, 2000 "Task Force H Report" A. Tuccillo.
- Tuesday 6th June, 2000 "ASDEX Upgrade and it's relation to JET and ITER" M. Kaufmann, IPP-Garching, Germany.
- Monday 19th June, 2000 "Recent Progress of JT-60U Experiments toward steady state operation of ITER." Dr. Takeji.
- Thursday 22nd June, 2000 "FIRE, a Next Step Option for Magnetic Fusion" Dale M. Mead (Head, Advanced Fusion Concepts Princeton Plasma Physics Laboratory).
- Tuesday 27th June, 2000 "Toroidal Rotation Without Momentum Input" Ian Hutchinson, MIT.
- Monday 3rd of July, 2000 "Scaling of the Critical Beta for Onset of the m/n=2/1 Neoclassical Tearing Mode in Conventional H-Mode Discharges in DIII-D." Rob La Haye, GA.
- Monday 10th July, 2000, "Tokamak operational behaviour with high and low-Z plasma Facing Materials" Volker Phillips, TFL-E, Julich.

- Tuesday 11th July, 2000, “Reflectometer Measurements of Core Turbulence in the Internal Transport Barrier of JT-60U” Dr. Raffi Nazikian PPPL.
- Monday 24th of July, 2000 “ITER-FEAT” P. Barabaschi, ITER Joint Central Team, Garching.
- Monday 4th September, 2000, HOW Rehearsals of Invited Talks for SOFT
  1) J. Pamela, “JET under EFDA: organisation, recent results and future perspectives”,
  2) A. Kaye, “High Frequency Heating and Current Drive Issues for a ‘Next Step’ Machine”.
- Wednesday 6th September, 2000, “Animation of Drift Ballooning Modes and Zonal Flow Turbulence” J. Candy, General Atomics, San Diego, CA.
- Monday 18th September, 2000, “H-mode & Internal Transport Barrier Studies in DIII-D” Punit Gohil, GA.
- Monday 25th September, 2000, Rehearsals for IAEA.
- Monday 16th and Tuesday 17th October, 2000, Rehearsals for APS.
- Friday, 27th October, 2000, “Overview of MHD in advanced scenarios in large tokamaks” Guido Huysmans DRFC. A “Simulcast” Video Conference from DRFC/CEA in Cadarache, France.
- Tuesday 31st October, 2000, “APS Highlights”.
- Monday 6th November, 2000, “Overview of the Tore Supra CIMES-1 project” A. Bécoulet, CEA-Cadarache.
- Monday 13th November, 2000, “Transport studies with ECRH in ASDEX Upgrade” Francois Ryter, MPI - IPP, Garching.
- Friday 24th November, 2000, “Workshop on Disruption Studies in JET” Various speakers.
- Monday 27th November, 2000,.
  (1) “Mode Conversion and Models for the Damping of Alfven Eigenmodes” A. Jaun, Alfven Laboratory, Royal Institute of Technology, Stockholm,
  (2) “Report from US-Japan workshop on innovative MHD controlling Magnetic Fusion Plasmas” D. Testa, PSFC, MIT, USA
- Monday 4th December, 2000
  (1) “Options for pellet fuelling for JET-EP” P. Lang, IPP-Garching,
  (2) “Pellet injectors for steady state plasma fuelling and diagnostics” I. Viniar, PELIN, Inc. (Canada) - PELIN Laboratory, Ltd. (Russia).
- Monday 15th January, 2001,
  (1) “Preparation of Framework Programme 6”, by J. Pamela
  (2) “RF-induced plasma rotation” T. Hellsten, EFDA-JET CSU, Culham.
- Monday 22nd January, 2001, “On the dynamics of runaway electrons” Per Helander, UKAEA.
- Monday 29th January, 2001, “Temperature and density evolution during a density limit disruption” Francisco Salzedas, CFN-IST, Portugal.
- Monday, 5th of February, 2001, “Transient transport experiments: what do we learn from them?” Paola Mantica, Istituto di Fisica del Plasma CNR-EURATOM, Italy.

- Tuesday, 13th of February, 2001, “Simulated alpha particle self-heating experiments in JET - present status and further prospects” T.T.C. Jones, UKAEA.
- Monday, 19th of February, 2001, “Remote Participation Technical Infrastructure for EFDA” V. Schmidt Technical Co-ordinator for EFDA-JET Remote Participation, Consorzio RFX, Italy.
- Monday, 26th February, 2001, Rehearsals for 9th American Nuclear Society Topical meeting on Robotics and Remote Systems.
- Monday, 5th March, 2001, “Prediction/modelling of the neutron emission from JET discharges” O.N.Jarvis, UKAEA Fusion.
- Monday, 12th March, 2001, “Design criteria for in-vessel components in JET and JET-EP” V. Riccardo, UKAEA.
- Monday, 19th March, 2001,
  (1) “Power deposition measurements in JET MKIIGB L-modes and ELMy H-Modes by IR-thermography” Th. Eich, IPP Jülich,
  (2) “Power Exhaust in JET MkIIGB ELMy H-modes” W.Fundamenski, UKAEA-Fusion.
- Monday, 26th March, 2001, “Tritium in JET First Wall Materials: A Chemists Approach” R.D. Penzhorn, FZK Karlsruhe.
- Friday, 30th March, 2001, Remote Rehearsal of IOP Presentation: “JET Under The European Fusion Development. Organisation, Recent Results and Prospects” Alain Becoulet, Cadarache. The talk was given from CAE Cadarache as a teleconference to the HOW Room.

7.7 Steering Committee and JET Sub-Committee Meetings

EFDA Steering Committee Meetings (Chairman C.Varandas, IST, Portugal)

Meeting No	Venue	Date
11th	Porto,Portugal	29 - 30 March 2001
10th	Leysin, Switzerland	15 December 2000
9th	FZK, Karlsruhe, Germany	23 - 24 October 2000
8th	IPP, Garching, Germany	26 June, 2000
7th	DG Research, Brussels, Belgium	17 April 2000
6th	Dublin City University, Ireland	24 March 2000
5th	Centre A. Borschette, Brussels	14 January 2000
4th	Consorzio RFX, Padova, Italy	27 - 28 October 1999
3rd	Centre A. Borschette, Brussels , Belgium	6 July1999
2nd	CEA, Aix-en-Provence, France	19 May 1999
1st	IPP, Garching, Germany	19 - 20 April 1999

EFDA JET Sub-Committee Meetings (Chairman M.Kaufmann, IPP-Garching, Germany)

Meeting No	Venue	Date
12th	UKAEA, Culham	22 March 2001
11th	UKAEA, Culham	24 January 2001
10th	DG Research, Brussels	29 November 2000
9th	UKAEA, Culham	12 - 13 October 2000
8th	DG Research, Brussels	7 September 2000
7th	CEA, Cadarache	3 - 4 July 2000
6th	UKAEA, Culham	13 April 2000
5th	IPP, Garching	28 - 29 February 2000
4th	UKAEA, Culham	3 December 1999
3rd	ERM/KMS, Brussels	22 October 1999
2nd	UKAEA, Culham	9 September 1999
1st	UKAEA, Culham	13 July 1999

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Contact: Mark Siegrist, EFDA-JET CSU, K1/0/104, Tel: +44 1235 46 44 13, Fax: +44 1235 46 44 15

e-mail: mark.siegrist@jet.efda.org.