# The Pumped Divertor

## THE NEW PHASE OF JET


**B.E. Keen and M.L. Watkins and the JET Team**

*JET Joint Undertaking, Abingdon, Oxford, UK*

**Associate Member of EPS**



**The pumped divertor experiment, in demonstrating before full deuterium-tritium operation an effective method of impurity control, aims to provide essential design data for a Next Step tokamak fusion device.**


The basic principle of the fusion process is the fusing of light nuclei to form heavier ones and the accompanying release of substantial energy. For a fusion reactor, there are several possible fusion reactions, but the one that is easiest to achieve is that between the deuterium and tritium isotopes of hydrogen. This D-T reaction is:

$D + T \rightarrow {}^4He + \text{neutrons} + 17.6$ MeV.

At the temperatures needed for this reaction to occur, the D-T fuel is in the plasma state, comprising a mixture of charged particles (nuclei and electrons), which can be contained by magnetic fields. The most effective magnetic configuration is the toroidal tokamak device, of which the Joint European Torus is the largest in operation.

For a D-T fusion reactor, the triple fusion product of the temperature $T_i$, the density $n_i$ and energy confinement time $\tau_E$ must exceed the value $n_i \tau_E T_i$ of $5 \times 10^{21}$ m$^{-3}$ skeV. This measures the performance of a fusion device and shows how close conditions are to those of a reactor. Typically, for a reactor based on magnetic confinement concepts, the following are required:

- central ion temperature, $T_i$: 10-20 keV
- central ion density, $n_i$: $2.5 \times 10^{20}$m$^{-3}$
- global energy confinement time, $\tau_E$: 1-2 s

During the early 1970's, it was clear that the achievement of near-reactor conditions required much larger experiments, which were likely to be beyond the resources of any individual country. In 1973, it was decided in Europe that a large device, the Joint European Torus (JET), should be built as a joint venture. The formal organization of the Project — the JET Joint Undertaking — was set up near Abingdon, UK, in 1978. The Project Team is drawn from EURATOM and the fourteen member nations — the twelve European Community countries, together with Switzerland and Sweden. By mid-1983, the construction of JET, its power supplies and buildings were completed on schedule and broadly to budget and the programme started.

JET is the largest project in the coordinated programme of EURATOM, whose fusion programme is designed to lead ultimately to the construction of an energy producing reactor. Its strategy is based on the sequential construction of major apparatus such as JET, a Next Step device and a demonstration reactor (which should be a full ignition, high power device), supported by medium sized specialised tokamaks.

The objective of JET is to obtain and study a plasma in conditions and dimensions approaching those needed in a thermonuclear reactor [1]. This involves four main areas of work to study:

– various methods of heating plasmas to the thermonuclear regime;
– the scaling of plasma behaviour as parameters approach the reactor range;
– the interaction of plasma with the vessel walls and how to continuously fuel and exhaust the plasma;
– the production of alpha-particles generated in the fusion of D and T atoms and the consequent heating of plasma by these alpha-particles.

The first and second areas of work have now been well covered and the third area has been well studied in the limiter configuration for which JET was originally designed. However, the highest performance JET discharges have been obtained with a "magnetic limiter", that is, in the so-called X-point configuration with a magnetic separatrix inside the vacuum vessel, with plasma contacting localised areas of wall (the X-point targets) and detached from the limiters (the first points of contact with the plasma) except during the formation of the discharge. The duration of the high performance phase of these discharges can exceed 1.5 s by careful design of the targets and specific operational techniques, but is limited, ultimately, by an unacceptably high influx of impurities. The fourth area of work had been started by earlier studies of energetic particles produced as fusion products or by ion cyclotron resonance heating (ICRH). It has now been addressed further by the first tokamak plasma experiments in D-T mixtures. These results are presented briefly in the following sections.

### Plasma Performance and Impurity Control

JET is in the second half of its experimental programme. The technical design specifications of JET have been achieved in all parameters and exceeded in several cases (see Table 1). The plasma current of 7 MA and the current duration of up to 60 s are world records and are more than twice the values achieved in any other fusion experiment. The neutral beam injection heating system has been brought up to full power ($\approx$ 21 MW) and the ion cyclotron resonance frequency (ICRF) heating power has been increased to $\approx$ 22 MW in the plasma. In combination, these heating systems have provided over 35 MW of power to the plasma.

Since the start of operation in 1983 the study of plasma-wall interactions under such high power conditions and the control of impurities have always been considered as key scientific and technical issues to which particular attention has been paid. Impurity production has been

**Table 1 — JET parameters**

| Parameters | | Design values | Achieved values |
|---|---|---|---|
| plasma major radius, $R_o$ | | 2.96 m | 2.5–3.4 m |
| plasma minor radius | - horizontal | 1.25 m | 0.8–1.2 m |
| | - vertical | 2.1 m | 0.8–2.1 m |
| toroidal field at $R_o$ | | 3.45 T | 3.45 T |
| plasma current | - limiter mode | 4.8 MA | 7.1 MA |
| | - single X-point | not foreseen | 5.1 MA |
| | - double X-point | not foreseen | 4.5 MA |
| neutral beam power | - 80 kV, D | 20 MW | 21 MW |
| | - 140 kV, D | 15 MW | 15 MW |
| ICRH power to plasma | | 15 MW | 22 MW |


**B.E. Keen** leads the Publications Group at the JET Joint Undertaking, Abingdon, Oxon OX14 3EA, UK. He received a Ph.D. in low temperature physics from Oxford in 1962 and then worked at Oxford, Yale University, Sussex Univ., and Culham Laboratory (from 1967) before joining JET.

**M.L. Watkins** is a Scientific Assistant to the Director of JET. After receiving a Ph.D. in plasma physics from Imperial College, London, in 1971 he joined the UKAEA Culham Laboratory where he worked on the JET project before being assigned to JET in 1979.

reduced by both **passive** methods of impurity control (such as the proper choice of plasma facing components — beryllium or beryllium carbide) and **active** methods (such as sweeping the magnetic configuration, and hence the plasma, across the targets at which the plasma-wall interaction is often localised). The resulting significant improvements in plasma performance have brought JET to equivalent "breakeven" conditions and to within a factor of 5 of the fusion triple product needed for a fusion reactor.

Up to 1988, JET operated with a carbon first wall (carbon tiles and wall carbonisation). A fusion triple product of $2.5 \times 10^{20}$ m$^{-3}$ keVs was achieved [2]. The attainment of higher plasma parameters was limited by impurity influxes, mostly carbon and oxygen, from the walls. The impurities diluted the plasma fuel, thereby decreasing the fusion reactivity and increasing the radiative energy losses. Excessively high impurity influxes (called the "carbon bloom") were observed during high power heating and led to a rapid deterioration of plasma parameters and fusion performance [3].

From 1989, JET has operated with a Be first wall. Because of its low atomic number, Be was expected to lead to superior plasma performance, resulting in much reduced radiative losses compared with carbon. It also has the advantage of acting as a getter for oxygen [4]. The experimental campaigns of 1989 and 1990 confirmed these expectations. The chief effect of Be is to improve plasma purity (defined as the ratio of fuel ion to electron densities) and, as a result, to increase plasma performance. A fusion triple product of 8-$9 \times 10^{20}$ m$^{-3}$ keVs was achieved at both high ion temperatures (> 20 keV, the so-called "hot ion mode") and medium temperatures (9 keV), in the parameter range that is more relevant to fusion reactors (see [5] for details of these physics results and fusion relevant parameters).

Towards the end of 1991, the performance of JET plasmas had improved sufficiently to warrant the first tokamak experiments using a D-T fuel mixture [6]. Tritium neutral beams were injected into a deuterium plasma, heated by deuterium neutral beams. This introduced up to 10% of tritium into the machine, although ultimately about 50% tritium will be used in a reactor. As a consequence, a significant amount of power was obtained in JET from controlled nuclear fusion reactions. The peak fusion power generated reached about 1.7 MW in a pulse lasting for 2 s, giving a total energy release of 2 MJ. This was clearly a major step forward in the development of fusion as a new source of energy, and should permit extrapolation to a Next Step device, which should demonstrate ignition in a routine way.

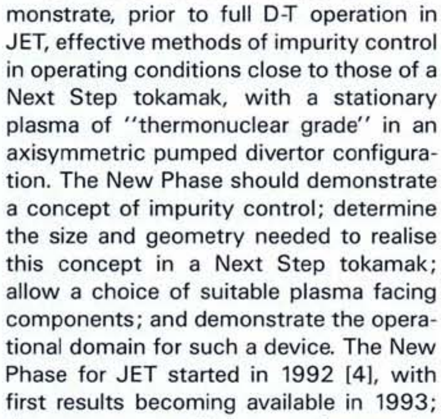
However, as in all high performance discharges, the high power phase is transient, lasting for less than 1 s. It could not be sustained in the steady state: the impurity influx observed with carbon walls also occurs with Be and causes a degradation of plasma parameters. This emphasizes the need for improved methods of impurity control in fusion devices.

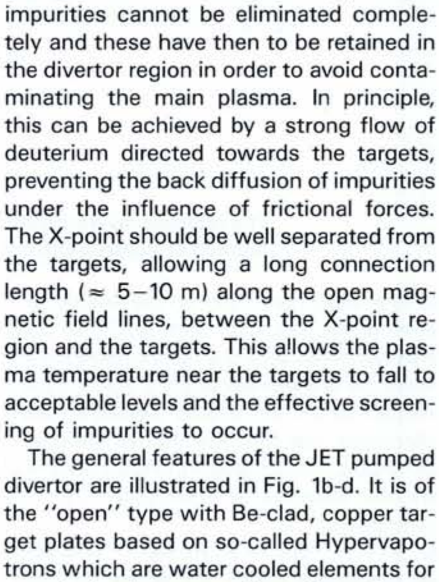
### The New Phase of JET and the Pumped Divertor

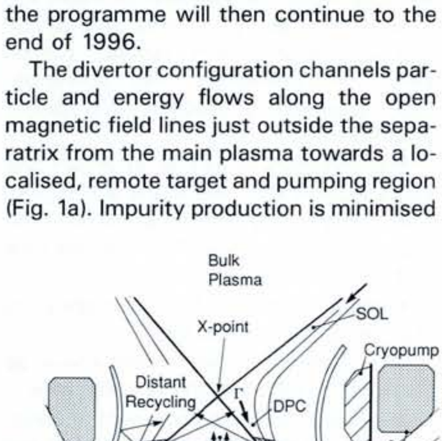
The aim of the New Phase is to demonstrate, prior to full D-T operation in JET, effective methods of impurity control in operating conditions close to those of a Next Step tokamak, with a stationary plasma of "thermonuclear grade" in an axisymmetric pumped divertor configuration. The New Phase should demonstrate a concept of impurity control; determine the size and geometry needed to realise this concept in a Next Step tokamak; allow a choice of suitable plasma facing components; and demonstrate the operational domain for such a device. The New Phase for JET started in 1992 [4], with first results becoming available in 1993; the programme will then continue to the end of 1996.

The divertor configuration channels particle and energy flows along the open magnetic field lines just outside the separatrix from the main plasma towards a localised, remote target and pumping region (Fig. 1a). Impurity production is minimised by the proper selection of target materials and by reducing the plasma temperature at the targets as far as possible. Although the principal source of impurities is well removed from the main plasma, sputtered impurities cannot be eliminated completely and these have then to be retained in the divertor region in order to avoid contaminating the main plasma. In principle, this can be achieved by a strong flow of deuterium directed towards the targets, preventing the back diffusion of impurities under the influence of frictional forces. The X-point should be well separated from the targets, allowing a long connection length ($\approx$ 5–10 m) along the open magnetic field lines, between the X-point region and the targets. This allows the plasma temperature near the targets to fall to acceptable levels and the effective screening of impurities to occur.

The general features of the JET pumped divertor are illustrated in Fig. 1b-d. It is of the "open" type with Be-clad, copper target plates based on so-called Hypervapotrons which are water cooled elements for transferring large quantities of heat from a surface that is subject to a high heat flux. The in-vessel four-coil system allows both



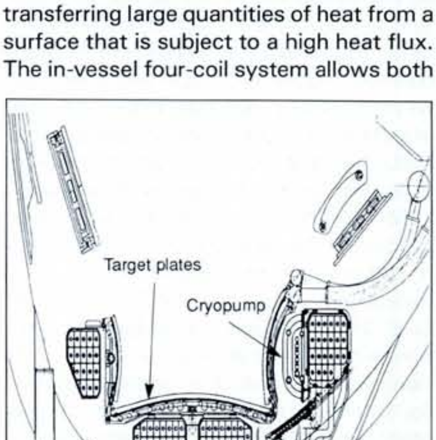



*Fig. 1 — The JET pumped divertor. (a, upper left) A cross-section showing the Hypervapotron targets located inside four magnetic divertor coils which are used to form the magnetic X-point configuration and to adjust the connection length. The scrape-off layer (SOL) and the Divertor Plasma Channel (DPC) are discussed in the text. (b, upper right) A vertical section through the divertor showing in detail the various components. (c, lower left) The layout of the cryopump showing the cryogen pipes placed in front of the divertor coil. (d, lower right) A photograph of one of the 384 U-shaped, beryllium-clad, targets.*

for horizontal sweeping of the plasma along the target to spread the heat load, and for vertical motion of the magnetic X-point to vary the connection length and plasma volume. The possible divertor configurations span the range from a "slim" 5 MA plasma (total divertor coil current of 1.5 MA) with the connection length of about 8.2 m, to a "fat" 6 MA plasma with a connection length 3.1 m.

**A Concept for Impurity Control**

The key concept of the JET pumped divertor, namely the retention of impurities near the targets by inducing a strongly directed flow of plasma particles along the divertor channel towards the targets is now discussed in more detail. Impurities produced at the targets are subject to several forces, as given by the steady-state momentum equation:

$$m_z n_z v_z \frac{dv_z}{ds} = \frac{-dp_z}{ds} + Zen_z E + \frac{m_z n_z (v_i - v_z)}{\tau_{zi}} + n_z \alpha_z \frac{dT_e}{ds} + n_z \beta_z \frac{dT_i}{ds} \quad (1)$$

Here we have considered a single impurity species of charge state $Z$, mass $m_z$, density $n_z$, temperature $T_z$, pressure $p_z$, and flow speed $v_z$ along the field coordinate, $s$. The electric field may be eliminated from this equation by using the electron momentum equation and taking the electron pressure to be nearly constant along $s$. Eq. (1) then becomes:

$$\frac{T_i \, dn_z}{n_z ds} = \frac{m_z (v_i - v_z)}{\tau_{zi}} + (\alpha_z - 0.71Z)\frac{dT_e}{ds} + (\beta_z - 1)\frac{dT_i}{ds} \quad (2)$$

For most cases of interest, $dT_e/ds \ll dT_i/ds$, since they are both determined by classical heat conduction with thermal conductivity $\kappa_{\parallel e} \gg \kappa_{\parallel i}$. The ion thermal force (the last term on the right side of Eq. (1)) is then dominant and is determined principally by the ion heat flow into the scrape-off layer (SOL). The counter-acting frictional force (the first term on the right side of Eq. (2)) depends on the magnitude and spatial distribution of the deuterium particle flux, $nv_i$, and is proportional to $T_i^{-3/2}$. Thus, in principle, the friction force can be set at a level sufficient to overcome the thermal force.

At high scrape-off layer densities, large plasma flows near the targets are established naturally by the ionization of neutrals that recycle at the targets. This ensures impurity control within an effective ionization length at the target, with the plasma flows increasing rapidly on approaching the target. Some neutrals, however, are not recycled locally near the targets, but escape and re-enter the scrape-off layer nearer the X-point after reflection from the torus walls, or transmission across the triangular region formed by the separatrix and the target plates — the "private flux" region. This "distant recycling" then extends the region of significant plasma flow, and the effectiveness of the friction force term, further from the targets. It is also possible, in principle, to increase distant recycling by extracting some of the plasma flow at the targets and directly "recirculating" the flow as neutrals, into the X-point region. Baffles might be needed to facilitate this. At lower scrape-off layer densities, the natural flow from the main plasma is not amplified sufficiently by recycling at the targets to ensure impurity retention. Strong gas puffing or shallow pellet injection into the scrape-off layer is needed. Steady state operation with such "external recirculation" then requires the pumping of an equivalent neutral flux from the divertor chamber and this can impose severe pumping requirements.

Another important feature of the pumped divertor is the formation of a cold and dense target plasma in the divertor channels. The cold dense plasma in front of the targets is expected: to radiate a significant fraction of the input power (thus reducing the heat load on the targets); to reduce the impurity production by shielding the targets; and to reduce the probability of impurities returning to the plasma.

**Technical Aspects**

*Magnetic configuration*

Fig. 2 shows the magnetic configuration which features four divertor coils. The number of coils reflects the experimental nature of the JET pumped divertor programme and the need for operational flexibility. The four coils will allow exploration of a wide range of magnetic configurations. They each carry currents in the same direction. The two, bottom, central

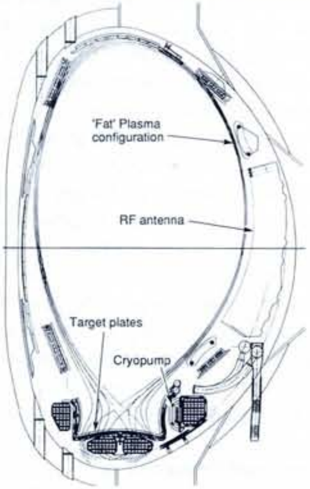



*Fig. 2 — Magnetic field configurations used with the JET pumped divertor. The illustration shows a "fat" configuration (6 MA plasma current, 88 m³ plasma volume, 0.74 MA-turns total divertor current) with a moderate elongation and a short (3.1 m) connection length. Changing the parameters of the X-point plasma to 6 MA/ 75 m³/1.5 MA-turns gives a more elongated "slim" configuration with a long (8.2 m) connection length.*

coils produce the X-point and have been made as flat as possible to increase the volume available to the plasma. The two side coils allow a reduction of the poloidal field in the region between the X-point and the target plates, thus changing the pitch of the magnetic field lines and consequently increasing the connection length. Since all four coils will have independent power supplies, greater flexibility can be achieved in the type of magnetic configuration (see Fig. 2).

The side coils allow the connection length to be adjusted both independently of the plasma current and separately on the inboard and outboard sides of the X-point. The strike zone of the separatrix and scrape-off layer can be swept radially to reduce the power deposition to an acceptable time average value. A total sweep amplitude of 20 cm at a frequency of 4 Hz is possible without significant changes of the connection length both inboard or outboard.

The coils are conventional and use water cooled copper conductors. They will be assembled inside the vessel from preformed one-third turn segments. The coils include 15 to 21 turns and carry typically 0.6 MA-turns. The divertor configuration can be maintained for typically 10 s at 6 MA, and up to one minute at 2 MA.

*Target plates*

The targets feature three elements (see Fig. 1d) in a U-configuration to accommodate the plasma and divertor contours. Horizontal plates at the bottom intersect the heat flux conducted along field lines. Therefore, these bottom targets are subjected to a severe power deposition. Vertical targets on either side can intersect the divertor plasma but receive only a modest heat load. The horizontal and vertical targets are split into 384 radial elements grouped into 48 modules of eight elements each.

In this configuration, the peak power density on the bottom targets would be unacceptably high even for the most advanced heat sinks. Consequently, sweeping the magnetic field across the targets is essential. The load can be accommodated by Hypervapotrons of the type used on the JET neutral beam systems. In contrast to the bottom targets, the side targets receive only a modest heat load from the divertor plasma.

The surface of the Hypervapotrons facing the plasma must be clad with a low-$Z$ material. The choice of beryllium is natural in view of the results already achieved on JET, although it is recognised that Be impurities generated in the divertor plasma will radiate only a negligible fraction of the incident power. However, the choice of material other than Be would entail the risk of impurities migrating back to the plasma and jeopardising the benefits of a Be first wall. Installation of the targets will proceed in two steps. During 1993, Mark I will use solid radiatively cooled carbon or Be blocks as targets. During 1994, Mark II will use water cooled Be-clad Hypervapotrons. This approach was motivated by the desire to start pumped divertor operation with a robust and simple design.

*The cryopump*

In the JET divertor, pumping will be achieved partly by the Be target plates and partly by a cryopump. Pumping by Be surfaces has been observed to diminish and stop during long pulses (> 30 s) and the cryopump is expected to play an essential rôle, particularly for long pulse experiments. A cryopump, rather than a getter or titanium sublimation pump, has been selected because it has no tritium inventory after regeneration, and is not affected by plasma operation or clean up techniques (glow discharge).

A drawback of the cryopump is its sensitivity to thermal loads. Pumping takes place through gaps between the target elements and is seriously restricted by the need to protect the helium cooled surfaces from high temperature gas molecules and thermal radiation. The design uses a liquid nitrogen cooled chevron baffle and water cooled structures.

Nuclear heating is expected to be the most severe heat load. In the case of operation at breakeven with a neutron production of $10^{19}$ $s^{-1}$, power of 4.5 kW is expected to be absorbed by the helium and the stainless steel helium conduits. To minimise nuclear heating, the conduits have thin walls and are slightly corrugated for increased strength. The heat capacity of the helium content of the pump ($\approx$ 40 litres) is about 50 kJ for a 1 K temperature rise and should limit the pulse duration only if operation is close to breakeven.

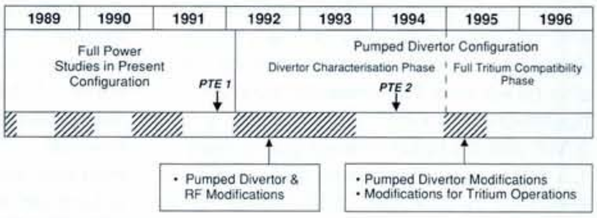
The pump has a nominal pumping speed for deuterium at 300 K of $5 \times 10^5$ $\ell s^{-1}$ and has the thermal capacity to cope with up to $10^{23}$ particles per second. The actual pumping efficiency depends crucially on the parameters of the divertor plasma, which should ideally be cold and dense, and on the neutral gas pressure in the vicinity of the target plates. The pump is split into quadrants in the toroidal direction and will have two cryo-supplies each common to two quadrants. The details are shown in Fig. 1.

*Fuelling*

Pellet fuelling is expected to play a key rôle in controlling the plasma density profile and impurity and alpha-particle concentrations in the plasma. Fast pellets at velocities up to 4 $kms^{-1}$ are planned to fuel the plasma centre and flush impurities towards the edge. Fast pneumatic guns, under development at JET, should be available for divertor operation.

Low velocity pellets (up to 500 $ms^{-1}$) launched by a centrifugal injector can fuel the outermost layers of the plasma and enhance the flow of plasma towards the targets. This type of gun ought to have the capacity to deliver long strings of small pellets (27 $mm^3$, 40 $s^{-1}$ for 10–30 s) and is being built at JET. Gas puffing will also be used if required, to enhance recycling in the vicinity of the targets. Twenty-four gas nozzles distributed along the torus will inject gas inside the private flux regions.

**Table 2 — JET programme schedule 1989-1996**




*PTE:* Preliminary Tritium Experiment

*Other components*

Eight ICRH antennae will be used to heat the plasma. Each antennae is designed to be moved radially and tilted to match the plasma shape and maximise power coupling. Twelve discrete poloidal rail limiters are also provided for plasma start up and to protect the ICRH antennae. These limiters are large vertical structures attached to the outboard wall of the vacuum vessel and similarly to the antennae: they can be moved radially and tilted. These limiters carry radiation cooled Be tiles on their front face. The inboard wall protections are carbon fibre composite tiles mounted on rigid vertical beams attached to the inboard wall reinforcing rings.

*Diagnostics*

In view of the experimental nature of the pumped divertor, an extensive range of diagnostics is planned. Many of these diagnostics are an integral part of the target assemblies. The main measurement goals are the local magnetic geometry near the targets (flux loops and magnetic probes), the plasma temperature and density in the divertor (Langmuir probes, LIDAR Thomson scattering, microwave systems including interferometry, reflectometry, and electron cyclotron absorption), impurity behaviour in the divertor (VUV and visible spectroscopy), the radiated power in the divertor channels (bolometer array), and the neutral gas pressure near the targets (pressure probes).

The targets and cooling pipes will be fitted with thermocouples to give data on temperature distributions and total incident power. Thermographic observation of the target plates using CCD cameras with infra-red filters is also foreseen.

**Experimental Programme**

Installation started early in 1992 and should take about 18 months. The programme with the pumped divertor is split into two periods (see Table 2). The first operation period will use the Mark I targets (radiation cooled) and will focus on establishing reliable operation, defining parameter space and identifying the optimum magnetic configuration.

A short shutdown will then allow for replacement of Mark I by Mark II targets (water cooled) and for the installation of the ICRH antennae and the outboard limiters in the positions matching the chosen plasma configuration. Other minor modifications may also be carried out at this stage. The second period of operation will be devoted to the study of impurity control at high power and during longer pulses.

**Conclusions**

Individually, each of the plasma parameters $n_i$, $T_i$ and $\tau_E$ required for a fusion reactor have been achieved in JET in single discharges. The triple fusion product of these parameters has now reached equivalent breakeven conditions and is within a factor five of that required in a fusion reactor. The performance improved sufficiently to warrant the first tokamak experiments using a D-T mixture in late-1991. However, these good results were obtained only transiently, and were limited by impurity influxes due to local overheating of protection tiles: they emphasize the importance of controlling dilution in a reactor. The divertor concept must be developed further to demonstrate effective methods of impurity control in an axisymmetric pumped divertor configuration.

The JET pumped divertor is an experiment to study impurity control scenarios and techniques in conditions relevant to the Next Step. For such an experiment, operational flexibility is essential. The design features four divertor coils which will permit experiments involving a variety of magnetic configurations. The targets are the most critical design issue and a two-step approach will be followed for their installation and testing. It is expected that the JET pumped divertor experiment will yield essential data for the design of the divertor of a Next Step device.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