IMPLICATIONS FOR ITER

JET Divertor Results

M. Keilhacker, M.L. Watkins

JET Joint Undertaking, Abingdon

Results obtained by JET in two areas of divertor research during recent experimental campaigns are of particular importance for the International Thermonuclear Experimental Reactor (ITER).

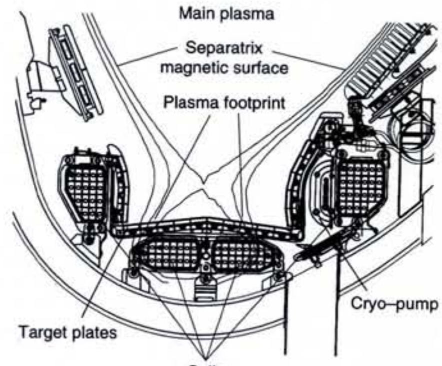


Fig. 1 — The JET pumped divertor showing the cryopump, the divertor coils which generate the magnetic configuration, the magnetic separatrix and the plasma footprint on the target plates.

Progress achieved in the Joint European Torus (JET) has been instrumental in defining the parameters of the Next Step tokamak on the route towards a prototype fusion reactor. Currently, the Next Step is identified as ITER, which is in the phase of its Engineering Design Activities (EDA) under the quadripartite ITER-EDA agreement between Euratom, Japan, the Russian Federation and the USA. By virtue of JET's size, geometry, divertor configuration, and proximity in working conditions to ITER, its results are crucial for validating the ITER divertor design by the end of the ITER-EDA.

After extensive modifications, JET resumed operations at the beginning of 1994 with a pumped divertor. During the ensuing 1994/95 experimental campaign, the power handling, heating (up to 32 MW of neutral beam and radiofrequency heating power), pumping and high current capabilities (up to 6 MA) of the new JET have been fully demonstrated and an extensive programme in support of ITER has been carried out.

The Divertor Concept

The JET divertor configuration (Fig. 1) channels particle and energy flows along open magnetic field lines just outside the separatrix from the main plasma to a localised, remote target and pumping region. The divertor must fulfil three main functions: (i) exhaust plasma power at acceptable erosion of the divertor target plates; (ii) control main plasma purity; and (iii) exhaust helium "ash" and provide density control. For ITER, successful divertor operation must also be compatible with the main plasma having high energy confinement (H-mode).

Erosion of the divertor target plates can be reduced by increasing the plasma density and decreasing the plasma temperature at the target plates. This is achieved in the "high recycling" divertor regime in which rapid recycling of plasma particles takes place near the target plates. However, in this regime, the exhausted plasma power conducted to the targets is not reduced and can only be accommodated by:

- inclining the target plates so as to project a larger surface area to the conducted heat flux (but not sufficient for reactor power levels); and/or
- "sweeping" the plasma "footprint" across the targets (as in JET, but not possible in a reactor).

At the beginning of the ITER Engineering Design Activity, it was realised that the high recycling divertor would result in target heat loads higher than those which could be accommodated without target melting or vapourization. This led to the proposal for reducing the conducted power to the targets by atomic physics processes (charge-exchanged neutral losses and hydrogen and impurity radiation) in the divertor channel. The resulting detached, highly radiating divertor plasma distributes the exhaust power over the divertor sidewalls, which have a sufficiently large area to reduce the heat load to acceptable values. The power losses can be enhanced by injecting seed impurities which then have to be retained in the divertor by generating plasma flows. This requires sufficient pumping and the relatively free recirculation of plasma within the divertor. At the same time, the divertor conditions required must not affect adversely the performance of the main plasma and this requires a low leakage of neutrals from the divertor to the main plasma. Such "closure" of the divertor can be achieved by introducing baffle structures at the entrance to the divertor or by maintaining a sufficiently dense plasma to attenuate neutrals within the divertor (plasma "plugging"). Computer calculations show that the geometry of the divertor is important in providing the necessary degree of closure, and this requires experimental testing of several different divertor configurations.

At JET, the investigation of detached, highly radiating divertor plasmas began in 1992, and has been pursued vigorously with the Mark I divertor which formed the basis of the 1994/95 Experimental Campaign. The Mark I divertor had inertially cooled carbon fibre composite (CFC) divertor targets and relied on sweeping to distribute the conducted power over a large enough target area for adequate power handling. It was a broad divertor which required the leakage path for neutrals from the divertor to the main plasma to be plugged by expanding the magnetic configuration, and hence the plasma, to fill much of the divertor.

Detached Radiative Divertor Plasmas

The divertor cryopump in the new JET allows good density control. By varying the location of the plasma footprint relative to the outer corner of the divertor (where the conductance to the pump is largest), the pumping rate can be varied by a factor of about two. When used in conjunction with varying amounts of neutral beam injection and gas fuelling, it is possible to vary the plasma density in steady-state by a factor

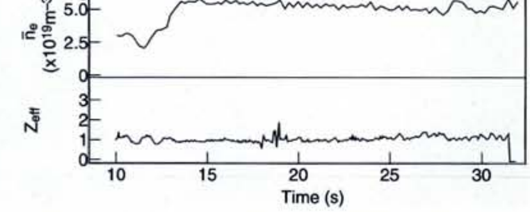


Fig. 2 — H-mode discharge in which pumping and ELMs help maintain steady-state plasma conditions for 20 s. From the top: total input power (P_{tot}), radiated power (P_{RAD}), stored plasma energy (W_{dia}), electron density ($\bar{n}_e$), and effective ionic charge (Z_{eff}).

M. Keilhacker was appointed Director of the JET Joint Undertaking, Abingdon, OX14 3EA, UK, in 1993 having served as the Associate and Deputy Director. After receiving a Ph.D. in physics from the TU Munich in 1959 he joined the Max-Planck-Institut für Plasmaphysik, Garching, where he became Project Leader of the ASDEX tokamak in 1974.

M.L. Watkins is the Scientific Assistant to the JET Director. 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.

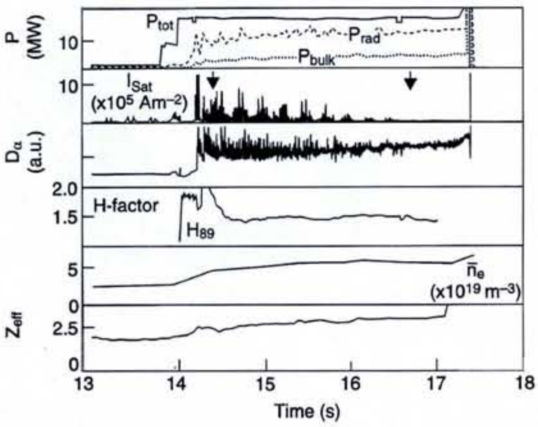


Fig. 3 — H-mode discharge in which a detached radiative divertor plasma is established by nitrogen seeding. From the top: total input power (P_{tot}), total radiated power (P_{rad}) and power radiated from main plasma (P_{bulk}), particle flow to the targets (I_{sat}), recycling from the targets (D_α), confinement enhancement factor (H), electron density ($\bar{n}_e$), and the effective ionic charge (Z_{eff}).

greater than 1.5. As shown in Fig. 2, the cryopump makes possible long, clean, stationary H-mode discharges with plasma density, impurity concentrations, radiated power losses and stored plasma energy remaining constant for up to 20 s (about 40 energy confinement times). The fine structure in the traces is due to magnetohydrodynamic activity in the edge plasma (Edge Localised Modes, or ELMs) which help in establishing stable conditions.

Detached divertor plasmas are produced as the plasma density is increased by deuterium fuelling. The particle flux at the target first increases (high recycling regime), then levels off ("rollover" regime), and finally decreases (detached regime). At a plasma density of about 5×10^{19} m^{-3}, the radiated power is about 70% of the input power, the plasma temperature in front of the targets is below 5 eV and both the heat flux and plasma pressure at the target are low. A significant pressure drop (more than a factor of 10 near the separatrix) develops between midplane and target. This is a signature of detachment.

Energy confinement in the main plasma is, however, low (L-mode) and, so far, it has not been possible to produce such detached divertor plasmas with H-mode confinement by using deuterium fuelling alone. This difficulty had been predicted theoretically for both JET and ITER. At the high powers required to maintain the H-mode, the plasma density cannot be increased sufficiently to reduce the divertor temperature to that needed for detachment (below 5 eV). H-mode detached plasmas can, however, be produced by introducing an impurity into the divertor to enhance radiation. Neon, nitrogen, and argon seeds have been injected into JET, either pre-programmed or feedback-controlled. It has been seen that nitrogen radiates more in the divertor, while neon and argon require lower influxes. Preliminary indications are that argon produces the lowest impurity contamination in the plasma core for a given radiated power fraction.

In the case of a nitrogen seeded discharge (Fig. 3), the radiated power fraction rises to more than 85%, with two-thirds of this emanating, as desired, from the divertor. Energy confinement in the main plasma is about 50% better than L-mode and is typical of detached H-mode discharges in JET. Impurity concentrations, as measured by the effective ionic charge Z_{eff} which rises to about 2.5 with nitrogen seeding (unity in a completely pure hydrogenic plasma), are somewhat lower than those found with neon seeds. The fine structure on the signals for the particle flows to the targets (I_{sat}) and recycling from the targets (D_α) is due to ELMs. Tomographic reconstructions of the radiation pattern (Fig. 4) in the lower half of the divertor shows that, when the plasma is partially detached, the radiation is well distributed throughout most of the divertor volume. In this case, there is still some power flux to the target, but it is quite low. When the plasma is fully detached, the radiating volume has moved away from the targets and is located at the entrance to the divertor, with some fraction of the radiation emanating from inside the separatrix. The motion of the radiating volume is smooth, but occurs over a fairly small range of total radiated power. Similar behaviour is predicted using the JET multispecies computer simulation codes.

Beryllium Target Tile Assessment

For the last months of the 1994/95 experimental campaign, the CFC divertor target tiles were replaced by beryllium tiles of similar geometry to compare CFC and beryllium as a plasma facing material. The campaign was concluded by exposing the beryllium target to significantly higher heat fluxes, with the explicit aim of studying the behaviour of molten and damaged beryllium target tiles.

Under normal operating conditions, the power handling capabilities of the CFC and beryllium target were found to be comparable. Long pulse steady-state H-modes with similar characteristics could be established on either carbon or beryllium. Carbon concentrations in the plasma were lower with the beryllium target, although impurity radiation was still dominated by carbon. This may be due to impurities from the carbon poloidal limiters and wall protection tiles. The H-mode power threshold, including its scaling with toroidal magnetic field and density, was the same with CFC or beryllium tiles. Behaviour at high density was the same, with little margin between detachment and the maximum operating density. The density range for detachment was also very similar and plasma fuelling and exhaust rates differed very little. H-mode confinement was lost when the radiated power fraction reached ≈ 50% with intrinsic impurities, and detached H-modes (radiated power fraction ≈ 80%) could only be achieved with impurity seeding.

Gross melting of the target tiles was generally avoided by limiting the input energy and by "sweeping". Superficial melt damage was caused, however, by some giant ELMs in high performance, low density H-mode discharges. Each ELM deposited about 1 MJ of plasma energy onto the target in about 20 ms. This is much shorter than the sweep period of 0.25 s and the heat is therefore deposited locally.

JET has also tested the hypothesis put forward by ITER that a beryllium target would "self-protect" itself against excessive heat fluxes during abnormal events by evaporated beryllium from solid and slightly molten targets leading to high radiation from the plasma and reduced heat fluxes. JET experiments were nominally at the ITER reference heat fluxes of 25 MW/m^2, but only a moderate degree of self-protection of the beryllium target was found. Despite significant surface melting, the radiated power rarely exceeded 50% of the input power although in some low density discharges it did increase to ≈ 70%, but only after several seconds, which is too long for effective self-protection. The target had melted to a depth of 2-3 mm near the plasma footprint on the outer target.

Following these experiments, operation on the melted beryllium was attempted.

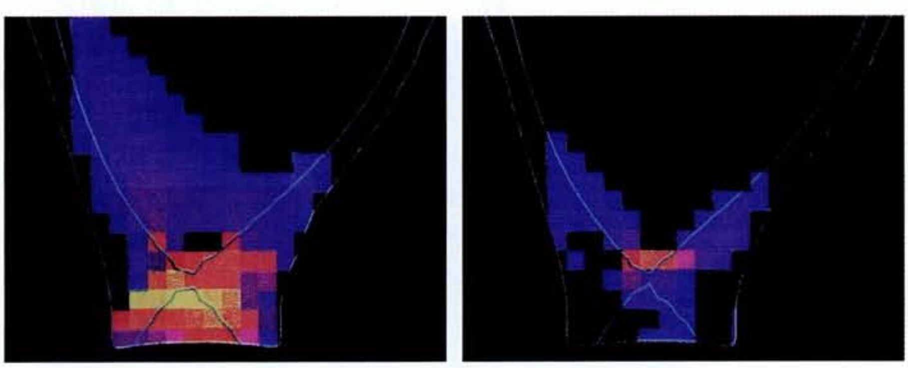
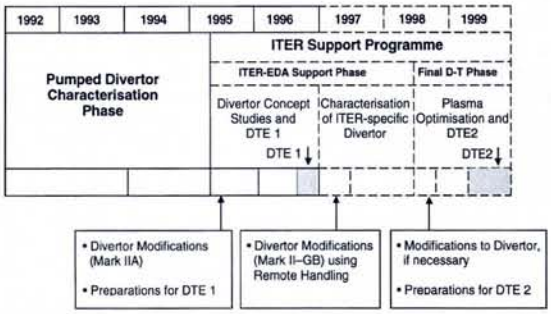


Fig. 4 — Tomographic reconstruction of the radiation pattern in lhe lower half of the divertor during partially (at 14.4 s: left) and fully (at 16.7 s: right) detached phases of the discharge of Fig. 3 showing the radiating zone moving smoothly from the target to the X-point region. The images are colour-coded blue through to yellow corresponding to increasing radiation density (temperature).

JET EXTENSION

ITER Support Phase



A European Commission proposal to introduce a three-year ITER-Engineering Design Activity (EDA) Support Phase into the JET Programme is awaiting the opinion of the European Parliament and approval by the European Union Council of Ministers.

This ITER-EDA Support Phase aims to provide further data of direct relevance to ITER's detailed design and to demonstrate key ITER technologies such as remote handling and tritium handling. In following on from the 1992-95 Pumped Divertor Characterisation Phase, it begins with the installation of an improved divertor (Mark IIA) to allow the study of ITER divertor concepts. Deuterium-Tritium Experiments (DTE 1) generating up to 2×10^{20} neutrons would then assess whether the favourable D-T confinement shown recently on the TFTR tokamak in the USA is likely to apply to ITER. More accurate scalings of the performance and heating of ITER could then be made. The experiments would also constitute the first large-scale tests of tritium processing technology with an operating tokamak.

Using remote handling, the divertor assembly would then be changed to the Mark II – Gas Box design to test a specific divertor concept for ITER using high-power deuterium plasmas. To capitalise on performance improvements, an extended series of D-T experiments (DTE 2) with an overall maximum neutron budget of 5×10^{21} would follow.

Low-density H-modes showed degraded performance, but the ITER-related ELMy H-mode discharges and nitrogen seeded radiative divertor discharges were essentially unchanged. Furthermore, in high power swept operation, a record 180 MJ of input energy was deposited on the target, causing additional melting.

Implications for ITER

The results for detached, radiative divertor plasmas are quite promising for ITER. The achieved radiated power fractions of 80-85% are sufficient to prevent target damage, overheating and erosion. Energy confinement in the main plasma would be just sufficient for ignition in ITER. Detachment is accompanied by a transition from large isolated ELMs to more benign "grassy" ELMs which should not cause target damage. There is somewhat more of a problem with main plasma purity from contamination by the seed impurity. The observed impurity concentrations would be barely acceptable in ITER with argon, and are too high with either neon or nitrogen. The scaling from the present JET experiments to ITER has, however, yet to be established. In JET, improvements are expected (see insert) with the more closed geometries of the Mark IIA divertor (start of operation in early l996) and the Mark II-Gas Box divertor (middle of 1997).

For operation with beryllium divertor target tiles, gross melting damage to the tiles was avoided by careful operation, although some superficial damage was produced by giant ELMs. Significant tile damage was inflicted during a controlled melt experiment when ITER reference heat fluxes of 25 MW/m^2 were applied and only a moderate degree of self-protection of the beryllium target was found. Following melting, the effect of a damaged beryllium target on the operating regimes relevant to ITER was found to be small. All in all, beryllium should remain one of the candidate plasma facing materials for ITER. The final choice will therefore depend largely on other considerations such as the retention of tritium in the materials.

PLASMA PHYSICS AND CONTROLLED FUSION CONFERENCE

Focussing on Tokamak Research

Most of this year's *22nd European Conference on Controlled Fusion and Plasma Physics* (Bournemouth, 3-7 July 1995) dealt with new experimental and theoretical results from magnetic confinement fusion research. Summaries of the invited papers give a "flavour" of the scope of the meeting and the significance of the latest results.

Topics at the *22nd European Conference on Controlled Fusion and Plasma Physics* covered: tokamaks; stellarators; alternative magnetic confinement schemes; magnetic confinement theory and modelling; plasma-edge physics; plasma heating; current drive and profile control; diagnostics; and basic collisionless plasma.

• ***Tokamaks***

The worldwide tokamak programme is focused on the needs of ITER, a joint effort between the European Union, Japan, Russia and the USA to design and plan to build a device providing an ignited fusion plasma for pulse lengths of typically 1000 s duration. An overview of the present status of **ITER** was given by G. Janeschitz (see page 110). A key problem is the energy and particle exhaust. One solution could possible come from a concept in which a large fraction of power is radiated from the plasma periphery, preferably from the divertor region, to large wall areas of the vacuum vessel. Under these conditions, the plasma may "detach" from the divertor target plates which then show a strongly decreased convective power fluxes.

M. Keilhacker presented for the **JET** Team the results obtained with the new pumped divertor in JET, the world's largest tokamak device, and analysed the implications for ITER (see page 105). G. Matthews gave detailed results on the divertor detachment in JET with deuterium gas puffing. At the very high levels needed for detachment, the confinement usually degraded from ELMy H-mode to L-mode. With neon and nitrogen injection, the appearance of ELMs changed. This was associated with a decrease of confinement. With neon, 50% of the radiation occurred from the bulk plasma, whereas nitrogen showed a factor of 2-3 more radiation from the divertor. T.T.C Jones reported on scenarios for high performance at JET. The most promising one is the hot ion H-mode requiring low hydrogen recycling to obtain peaked density profiles together with central beam deposition. High edge shear and triangularity are important for long, ELM-free periods.

• ***Divertor physics***

S.L. Allen discussed recent results from divertor investigations at San Diego's **DIII-D** (the largest divertor tokamak in the USA), notably gas-puffing experiments with deuterium and neon. With deuterium, a stable radiating zone in the divertor was found, resulting in a partially detached divertor plasma. For neon, radiation occurs from inside the confined plasma which suffered from enhanced values of effective ionic charge; the divertor heat flux was reduced by factors of 3-5. Active particle control by a cryopump installed in the divertor region has demonstrated density control, efficient helium exhaust, and reduction of the particle inventory in the wall.

J. Neuhauser described studies on the compatibility of enhanced energy confinement and complete divertor detachment. Using feedback control of injected neon and deuterium, the **ASDEX**-Upgrade team succeeded in establishing a stable radiative plasma boundary, reducing the divertor heat flux to very low values, with frequent small ELMs, and H-mode factors around 1.7 (CDH-mode: completely detached H-mode).

The latest results from the large **JT60-U** tokamak in Japan were presented by M. Itami. With deuterium and neon injection, 5 MW of a total of 25 MW injected power were radiated at a toroidal field of 2 T maintaining ELMy H-mode confinement, whereas a loss of confinement was observed at higher toroidal fields. Other parts of the presentation related to studies of high confinement modes.

A general problem for detached divertor regimes is the shrinkage of operation space. At the high densities needed, the plasmas tend to fall out of the high confinement regimes. Seeding by impurities is often associated with increased impurity influx causing a deterioration of the core plasma. Future work on the divertor tokamaks will address the effects of divertor geometry for the optimization of energy and particle exhaust. Plans for modifications of the present geometries were presented for all devices.

• ***Confinement, heating and control***

M. Tokar presented experiments from TEXTOR and detailed modelling efforts on