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ABSTRACT
The JET programme is strongly focused on preparations for ITER construction and exploitation. To this end, a major programme of machine enhancements has recently been completed, including a new ITER-like Wall, in which the plasma-facing armour in the main vacuum chamber is beryllium while that in the divertor is tungsten – the same combination of plasma-facing materials foreseen for ITER. The goal of the initial experimental campaigns is to fully characterise operation with the new wall, concentrating in particular on plasma-material interactions, and to make direct comparisons of plasma performance with the previous, carbon wall. This is being done in a progressive manner, with the input power and plasma performance being increased in combination with the commissioning of a comprehensive new real-time protection system. Progress achieved during the first set of experimental campaigns with the new wall, which took place from September 2011 to July 2012, is reported.

1. INTRODUCTION
In 2006, a JET Programme in Support of ITER [1] was launched with the goal of exploiting JET's unique capability to handle tritium and beryllium in a coherent approach along three main axes:
– Experimentation with an ITER-like Wall;
– Development of plasma configurations and parameters towards the most ITER-relevant conditions; and
– Integrated experimentation in deuterium-tritium.

The scientific objectives of the ILW experiment were and are to:
– Demonstrate sufficiently low fuel retention;
– Demonstrate ITER-relevant tritium retention mitigation;
– Show impact of beryllium migration on divertor erosion and core tungsten density;
– Effect of transients (ELMs and disruptions) on ILW;
– Develop control strategies for detecting and limiting damage;
– Study melt layer behaviour in ELM and disruption energy losses and implications for subsequent plasma operation;
– Develop integrated ITER compatible scenarios for an all-metal machine including impurity seeding; and
– Investigate special heating system-related effects.

Following a brief description of the ITER-like Wall in Section 2, progress towards meeting these scientific objectives will be described. In Section 3, a review of operational experience with the wall will be given. In Section 4, progress to-date of integrating ITER plasma scenarios with the new Be/W wall is given. The paper concludes with a section summarising the results and describing future plans for JET.

2. JET'S ITER-LIKE WALL
The new plasma facing components in JET are the same combination of beryllium in the main plasma chamber and tungsten in the divertor as foreseen for the active phase of ITER operation. An

annotated photograph of the ITER-like Wall is shown in figure 1. The wall is composed of almost 3000 installable items, compromising more than 15,000 tiles and approximately 2 tonnes each of beryllium and tungsten. A combination of solid tiles (beryllium and tungsten) and coatings has been used for the wall, which is inertially-cooled, in contrast to the wall to be used in ITER. Virtually all of the components of the ITER-like Wall were installed using JET's remote handling capabilities [2,3]. During the 18 month shutdown, manned access to vessel was used for only three short periods of approximately two weeks each. Upgrades to the remote handling equipment as well as careful attention to the remote handling interfaces throughout the entire design and procurement process resulted in a substantial improvement in installation productivity. Indeed, the main issues faced during the installation of the wall were a result of discrepancies between the JET configuration model and the state of the vessel unveiled during the shutdown itself [4], rather than difficulties encountered with design and fabrication process itself.

3. OPERATIONAL EXPERIENCE WITH THE ILW

3.1 POWER HANDLING

The exploitation of the ITER-like Wall (ILW) is being done in a progressive manner, with the input power being increased in combination with the commissioning of a comprehensive new real-time protection system. Nevertheless, during the first period of experimentation, which ran from August 2011 to July 2012, it has been possible to demonstrate that the ILW meets or exceeds its design power- and energy-handling capabilities [4,5]. Indeed, record levels of neutral beam power (up to 25.9 MW) have been injected into JET plasmas during this period, with no damage to the wall. Some damage to the solid beryllium limiters has occurred, however, during transient such as disruptions and runaway electron beam events. More detail on the causes and consequences of these events is given in [4].

3.2 REAL-TIME PROTECTION OF THE ILW

In order to safeguard the investment in the ILW, a new real-time protection system has been implemented [6-8]. The system has been integrated into the existing JET protection architecture with the main new components being a Vessel Temperature Map (VTM) system that uses the data from an expanded suite of first wall temperature measurements [9-13] to build a model of the temperatures of all critical in-vessel components and a Real-Time Protection Sequencer (RTPS) that takes input from VTM and other pre-existing protection diagnostics and uses it to generate selectable responses to overheating of particular components. The system has been progressively implemented during the 2011/12 campaigns and is now essentially fully operational. Remaining work in the project during 2012/13 involves completing the set of diagnostic measurements, extending the advanced functionality to the JET Lower Hybrid Current Drive system and commissioning more advanced responses/stops as required by the experimental programme.

3.3 FUEL RETENTION

The primary reason for not using carbon as the plasma-facing armour in ITER is its propensity to trap and retain the hydrogen isotopes used as the reactor fuel. In a plant using deuterium-tritium mixtures, such retention rates are limited by safety requirements on the maximum in-vessel tritium inventory. In ITER, the corresponding fuel retention rate must be less than 10^{20} atoms per second if the build-up of tritium is not to cause unwanted delays for tritium removal between divertor cassette changes [14].

Measurements of the deuterium fuel retained in the JET ILW have been made in variety of plasma conditions and compared to similar measurements made with previous carbon wall [15,16]. As shown in figure 2, the measured retention is more than an order of magnitude lower with the ILW, consistent with predictions made before the wall was installed and with the model which is being applied also to ITER.

3.4 IMPURITY MIGRATION

The 2011/12 JET experimental programme began with a dedicated campaign to study the migration of beryllium inside the tokamak and, in particular, to study the influence of beryllium migration on divertor tungsten erosion. The experiments were carried out with Ohmic plasmas in order to minimise the amount of pre-campaign plasma commissioning and thus provide the experiment with as near pristine wall conditions as possible.

The long-term evolution of the beryllium influx from the divertor, measured during identical reference pulses, shows a fast rise for the first ~100 pulses of the dedicated migration experiment, a slower rise during subsequent low power, L-mode experiments covering roughly 1000 pulse and a subsequently more or less constant value [17]. Modelling of the beryllium migration from the main chamber to the divertor using the WALLDYN code reproduces the observed trends during the initial migration experiment only if it is assumed that the initial plasma limiter commissioning (corresponding to approximately 540 plasma seconds) was sufficient to transport beryllium onto the tungsten divertor surface [18] (see figure 3).

3.5 IMPURITY EROSION AND CONTENT

The erosion of the new wall components by plasma and neutral bombardment has been carefully documented using an expanded range of spectroscopic diagnostics. The erosion of main chamber beryllium was tested in dedicated limiter pulses and is consistent with physical sputtering by deuterium and, especially at low densities, with self-sputtering [19]. Modelling is underway [20] to benchmark main chamber (beryllium) erosion calculations in divertor configurations with the goal of improving predictions for the lifetime of the ITER first wall.

The erosion of the tungsten from the new divertor has also been the subject of specific experiments and dedicated modelling. The measured influxes are consistent with the expected combination of physical sputtering by deuterium and plasma impurities [21-23] with beryllium being the dominant intrinsic impurity. The tungsten core concentration can be kept low with the use of sufficient gas fuelling but limits low fuelling rate, low density H-mode operation due to central accumulation.

The carbon content of JET plasmas has been reduced by more than a factor of ten with the installation of the ILW [17]. There has been a corresponding decrease of the plasma effective charge, with Z_{eff} now typically 1.2-1.4 for both L-mode and H-mode discharges as compared to 1.8-2.5 in similar discharges with the carbon wall.

3.6 ICRH INTERACTION WITH THE ILW

The interaction of the radio frequency waves used for Ion Cyclotron Resonance Heating (IRCH) with metal plasma-facing components is of concern in JET and ITER both due to the possibility of local overheating of components and to the potential for additional sputtering of tungsten, thus leading to unacceptable tungsten concentrations in the confined plasma.

Dedicated experiments with the ILW [24] have been used to quantify the ICRH-induced power load on surrounding PFCs. Due to the lack of deposited layers on the wall and in contrast to previous experiments with the JET carbon wall, a simple thermal model can be used to deduce local ICRH heat loads from infra-red thermography. There is good agreement between modelling and the measured dynamic temperature behaviour of the in-vessel components during ICRH. This gives confidence that the deduced maximum power loads (up to $4.5MW/m^2$) are realistic and that the models can be used to guarantee safe operation up to the wall's design limit (typically $6MW/m^2$).

With regard to sputtering of tungsten, comparison experiments using ICRH- and NBI-heated L-mode plasmas [25] have shown significantly higher tungsten concentration and radiated power in discharges heated by ICRH. The levels of tungsten reached are, however, tolerable. Neither the plasma energy nor the temperature decreases significantly, except at very high minority hydrogen concentrations. On the other hand, it must be said that the utility of ICRH to widen the available operating space, as observed in AUG [26], has not yet been demonstrated with the ILW.

The source of the increased tungsten has not yet been identified. Tungsten sputtering of some divertor areas magnetically connected to the antennas could contribute to the plasma impurity increase with ICRH [27]. On the other hand, some experiments indicate a substantial contribution from some remote areas covered with tungsten in the main chamber. In particular, ICRH heated limiter plasmas also exhibit higher tungsten content with ICRH and a preliminary experiment using the JET beryllium evaporators resulted in substantially reduced impurity content during ICRH. Further experiments are planned for the 2013 experimental campaigns.

3.7 DISRUPTIONS

The dynamics of disruptions are very different with the ILW due to the higher plasma purity and thus lower radiation losses during the disruption [28]. Indeed, the lower radiation and higher temperatures routinely lead to disruption heat loads [29] and forces [30] that would be unsafe at even moderate levels of plasma current in JET. For this reason, the use of massive gas injection as a disruption mitigation tool is now mandatory for JET experiments at or above 2.5 MA [31]. With the mitigation, the forces and power loads resulting from disruptions are returned to the level observed with the carbon wall (see figure 4). In parallel to this mitigation, significant improvements have been made in the detection of disruptions at JET [32,33], which will allow an earlier response to impending disruptions when activated in closed loop.

Two operational issues associated with disruptions and the ITER-like Wall are also worth noting.

The generation of dust due to disruptions is a significant safety concern for ITER and for that reason a qualitative measurement of dust using the laser light from Thomson Scattering scattered from particles immediately after disruptions [34]. Since the installation of the ILW, this system has measured an order of magnitude less ‘dust’ than was seen during operation with the carbon wall. The robustness of plasma operation to disruptions has also been greatly improved with the ILW. Whereas roughly a quarter of plasma attempts failed after a disruption with the carbon wall, this phenomenon is completely absent with the ILW [28].

4. PLASMA SCENARIOS WITH THE ILW

4.1 CONVENTIONAL H-MODE

H-mode scenarios were developed progressively during the 2011/12 campaigns as the new protection systems were commissioned and the enhanced neutral beam injection system [35] was brought into service. By the end of operation in July 2012, operation at 3.5MA had been re-established and record levels of NBI power had been accommodated. An example of a typical ELMy H-mode with the ILW is given in figure 5. The resulting scenario incorporates measures such as sophisticated stops now possible with the new real-time protection, the routine use of the disruption mitigation valve for massive gas injection and sweeping of the divertor strike points to spread the deposited power and energy. In addition, it was necessary to use high levels of gas fuelling (a few times 10^{22} D/s) to increase the ELM frequency and thus maintain the stability of the regime against impurity accumulation.

The confinement of H-modes with the ILW is typically about 20% lower than that achieved previously. The origin of the confinement loss appears to be in the edge transport barrier [36, 37] with the effect propagating to the core due to the profile stiffness generated by turbulent transport. Interestingly, the variations are different in low and high triangularity configurations. For low triangularity discharges such as those used in the experiments at higher plasma current, the main difference appears to be a trade-off of higher edge density for lower edge temperature when using gas fuelling to control impurities. Indeed, there are already examples of low triangularity, high power H-modes with confinement enhancement factors of unity, as measured against the ITER reference scaling [38]. For high triangularity configurations, the edge and core confinement with the ILW are lower due to a lower edge temperature even at the same edge density.

It is fair to say that the exploration of conventional H-mode scenarios with the ITER-like Wall has only just begun and that further optimisation is expected in future campaigns. Nevertheless, understanding and improving the edge transport barrier performance will be a priority task with the resumption of experiments on JET in 2013.

4.2 HYBRID SCENARIOS

To so-called hybrid scenario is intermediated between the conventional H-mode and fully steady-state regimes of operation in that it relies on self-driven current both to increase the potential pulse duration and to improve the plasma confinement due to profile effects. Hybrid scenarios are foreseen in ITER as either a scenario for long pulse operation and technology testing or so as to provide margin in obtaining the Q=10 mission.

The goals of hybrid experiments in JET in 2011/12 were to re-establish the regime with the ILW, to push to higher plasma current taking advantage of increased available input power and to investigate impurity seeding as a means of mitigation the first wall power loads.

In contrast to conventional operation, it has been possible in high triangularity hybrid modes to reproduce the performance achieved with the carbon wall, at least transiently. The duration of the high performance phase is typically limited by onset of strong MHD with subsequent impurity accumulation in the plasma core. Attempts to prolong the duration of the regimes have been successful albeit at the price of somewhat reduced performance. As with conventional H-modes, the experimental time available so far has not allowed anything but an initial attempt at regime optimisation and further experiments are being prepared for 2013.

4.3 IMPURITY SEEDING

The addition of impurities to enhance radiative losses is a crucial part of the ITER strategy for steady state power handling. With the installation of the ITER-like Wall and the upgrade of the additional heating systems to a capability of ~40 MW for 20 s, this tool has also become necessary in JET as a means of achieving maximum performance.

Experiments with impurity seeding of H-mode discharges during 2011/12 have concentrated on nitrogen fuelling (in combination with deuterium fuelling) and high triangularity discharges, as this corresponds to the largest database in the old, carbon wall. In addition to increasing the total radiated power, addition of nitrogen has been found to improve the plasma confinement. This is thought to be primarily an edge effect as the pressure at the top of the edge transport barrier increases to values similar to those achieved with the carbon wall. With increasing nitrogen fuelling, the total radiated power saturates at about 60% of the input power and the confinement begins to degrade. Tests at low triangularity have not shown the same increase in confinement with the addition of nitrogen, suggesting that the effect of nitrogen on confinement is related to the MHD stability of the plasma edge.

Hybrid discharges at JET require very high input power in order to achieve the required plasma pressure and operate best at low to moderate density. This combination results in rapid heating of the divertor tiles and makes impurity seeding mandatory if the regime is to be brought to maximum performance. To date, nitrogen fuelling has been tested only in a very few hybrid discharges. The result is a dramatic decrease in the divertor tile temperature but at the cost of a reduction in confinement. The effect, however, is reversible as there is sufficient pumping to reduce the nitrogen level on the time scale of the applied heating pulse. It is thus planned to employ real-time control of the nitrogen fuelling in the 2013 experimental campaigns.

5. SUMMARY AND FUTURE PLANS

The JET programme is strongly focused on preparations for ITER construction and exploitation. To this end, a major programme of machine enhancements has recently been completed, including a new ITER-like Wall, in which the plasma-facing armour in the main vacuum chamber is beryllium while that in the divertor is tungsten – the same combination of plasma-facing materials foreseen for ITER.

The first set of experimental campaigns with the new wall has just been completed. Already, results with the new wall are providing information that will influence the operating regimes and plans for ITER. In this regard, the expected strong (order of magnitude) reduction of hydrogen retention has been confirmed. Low levels of carbon in the plasma have also led to significant changes in the JET operating scenarios. Plasma initiation is now much easier than with the carbon wall and little, if any, de-conditioning is observed after disruptions or operation with extrinsic impurities. On the other hand, radiation losses are lower during the disruptions, leading to relatively higher thermal loads on first wall components. Disruption mitigation using massive gas injection is being routinely used, with which it is possible to fully recover the good results obtained with carbon walls.

Confinement enhancement factors ($H_{98(y,2)}$) of up to unity have been obtained in conventional H-mode discharges. The regime has been extended to 3.5 MA and record levels of neutral beam input power without damaging the new wall. For these plasmas, operation above a certain ELM frequency is important in order to avoid accumulation of high-Z impurities in the plasma core and the typically observed confinement is ~20% lower than both the $H_{98(y,2)}$ scaling and values obtained with the carbon wall. The use of nitrogen as an extrinsic impurity for radiation and divertor power load control has begun. Initial experiments have shown increases in global plasma confinement similar to those observed on ASDEX Upgrade [39], which appear to be the consequence of improvement in the edge confinement and stability.

Having demonstrated the compatibility of the new wall with operating at moderate power, a primary goal of the future JET programme will be to probe the present operating limits at high input power and plasma beta. This programme will take advantage of the recent upgrade to the JET neutral beam injection system. Already in the present campaigns, $\beta_N \sim 3$ has been achieved in the ITER hybrid regime of operation, showing that high performance operation is compatible with the ITER-like Wall. In both conventional and hybrid regimes of operation, the incorporation of impurity seeding to reduce power and energy loads on the divertor is seen as a main ingredient to be added to the present recipes for determining the maximum plasma performance compatible with the wall. As foreseen in the original proposal for this enhancement of JET [1], preparations are underway for a final demonstration of optimised regimes of operation in deuterium-tritium plasmas.

Finally, as part of the R&D required to support the ITER IO's recent proposal to begin operation with a full tungsten divertor, a dedicated tungsten melt experiment has been designed in collaboration between the ITER and JET Teams. A divertor module with modified lamellae so as to deliberately expose part of the divertor to the full parallel heat flux is being installed during the present JET shutdown. Experiments to make deliberate, shallow melts of this region of the divertor are planned for mid-2013, in time for the results and first analysis to be available as input to the final decision on the ITER day-one divertor armour material.

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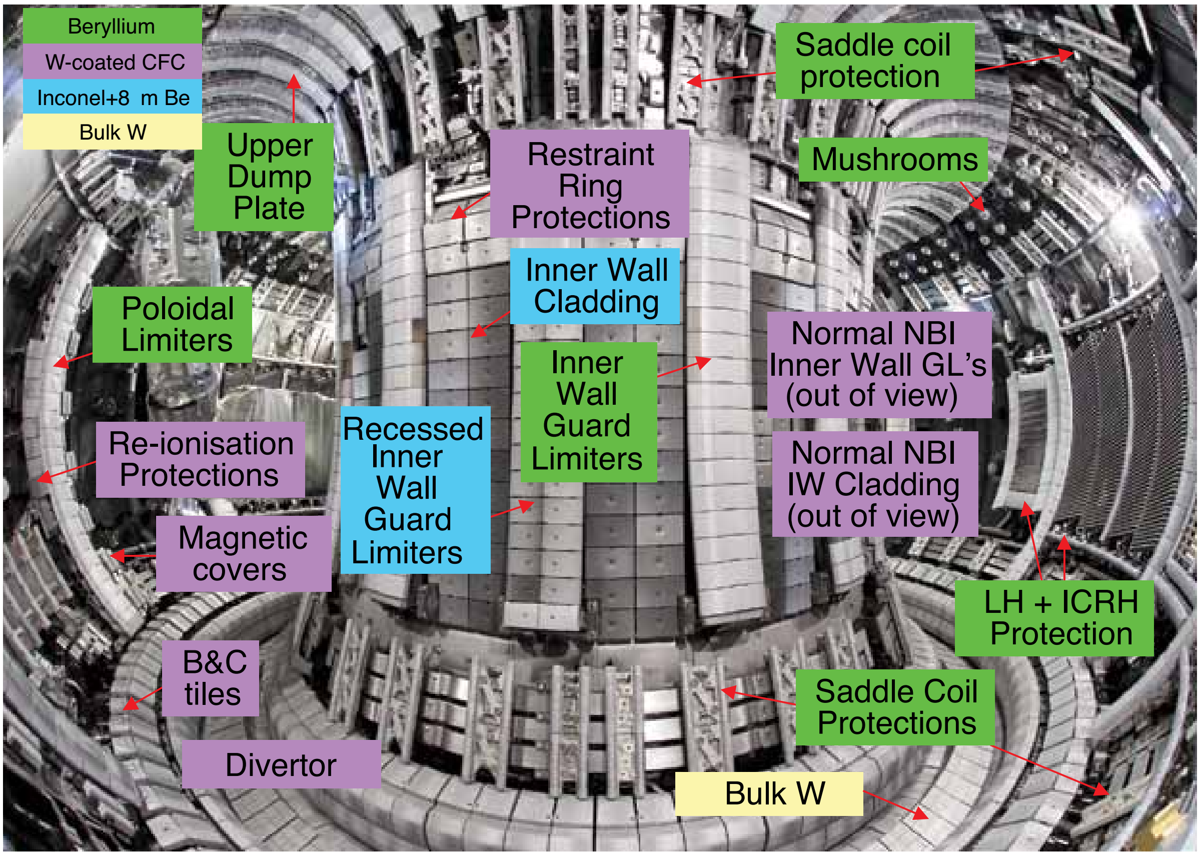

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Figure 1: The JET ITER-like Wall, showing the distribution of solid beryllium, beryllium-coated inconel, solid tungsten and tungsten-coated carbon fibre composite material.

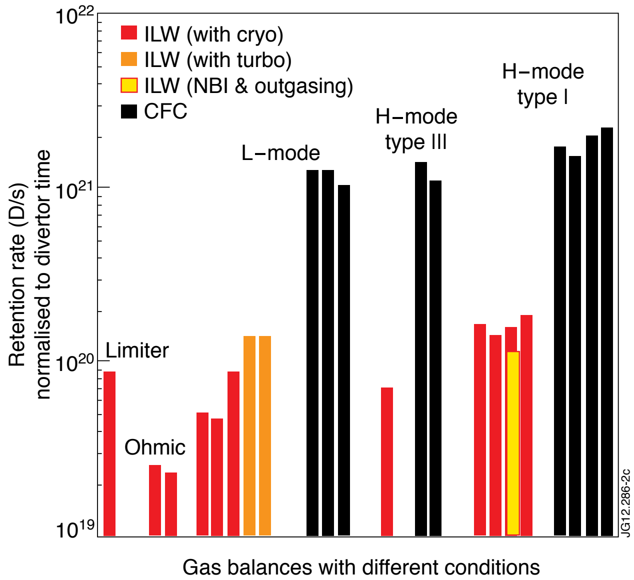


Figure 2: Retention rates of deuterium in the ITER-like Wall for different plasma operating conditions [16].

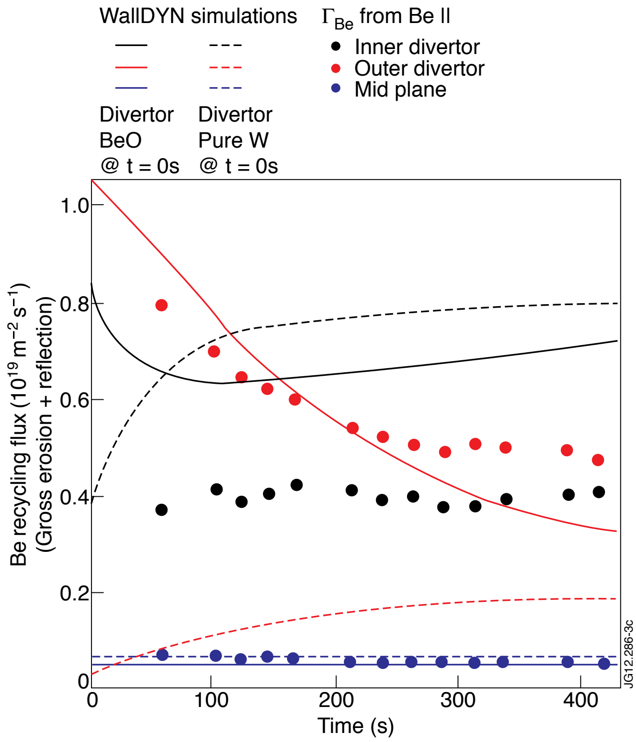
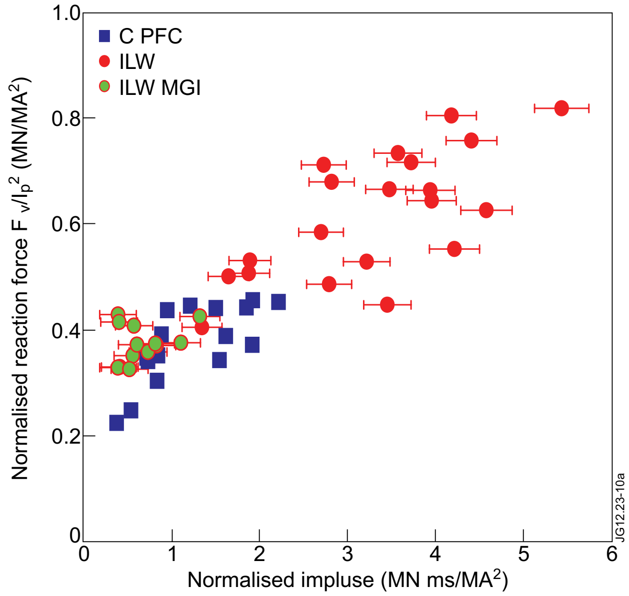


Figure 3: Measured (circles) and calculated (curves) beryllium recycling flux from the inner divertor (black), the outer divertor (red) and main chamber (blue) during the divertor phase of the migration experiment carried out in the initial ITER-like Wall campaign [18]. The experimental trends can only be reproduced when an initial BeO layer is assumed on the divertor (solid curves) as opposed to assuming pure W (dashed curves).

(a)



(b)

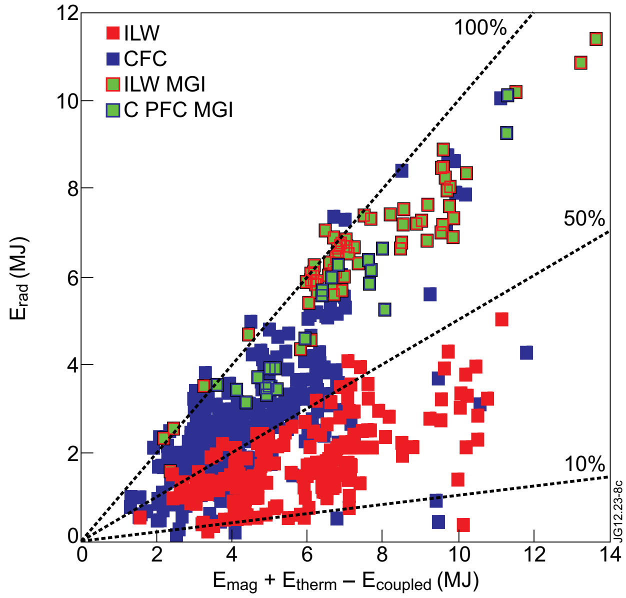


Figure 4: (a) Energy radiated from the plasma during disruptions with the JET ITER-like Wall (red), the old JET carbon wall (blue) and for disruptions mitigated using massive gas injection in both cases (green) as a function of the available energy. (b) Reaction force on the JET vacuum vessel during mitigated and unmitigated disruptions versus the normalised impulse generated by the plasma [28].

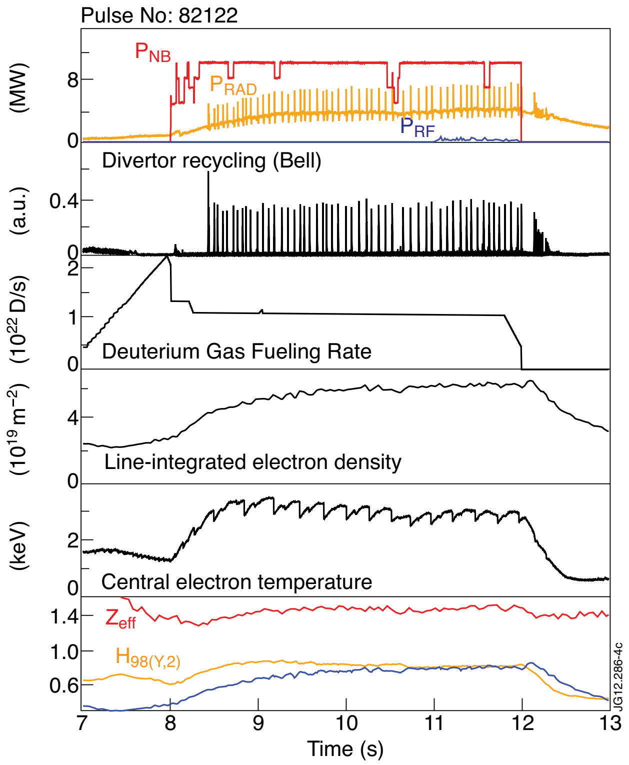


Figure 5: Time traces from a conventional, Type I ELM H-mode discharge with the JET ITER-like Wall. Shown are: Input neutral beam and ICRH power and total radiated power; Outer divertor Be II recycling emission; Deuterium gas fuelling rate; Line-integrated electron density; Central electron temperature; and Plasma effective charge, confinement enhancement factor and Greenwald fraction.