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Beryllium as a Plasma Facing Material for Near-Term Fusion Devices

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Abstract

This paper reviews the properties and features of beryllium that are of primary relevance for plasma protection applications in near-term magnetic fusion devices (i.e., plasma-wall interactions, thermal and mechanical properties, fabricability and ease of joining, chemical reactivity, etc.) together with the most recent knowledge on performance and operation in existing fusion machines (e.g., Joint European Torus). Special attention is given to beryllium's erosion and deposition and formation of mixed-materials, hydrogen retention and release characteristics that play an important role in plasma performance, component lifetime and operational safety. The status of the available techniques presently considered for joining the beryllium armour to the heat sink material of copper-alloys for the fabrication of beryllium-clad actively cooled ITER first-wall components is briefly discussed together with the results of the performance and durability heat flux tests conducted in the frame of the ITER first-wall qualification programme. The effects of neutron irradiation on the degradation of the properties of beryllium itself and of the joints, is also analyzed. Based on the information available from existing fusion machines, the expected behaviour of beryllium in ITER and in future reactors is critically discussed and an attempt is made to quantify the main effects that are still at issue in the design and operation of ITER. It should be noted that an earlier version of this paper has appeared elsewhere few years ago [1]. However, several parts have been updated to take into account very recent results of design and R&D work - primarily carried out in support of the ITER Programme. This includes for example the design of a new ITER Be first-wall, recent experience from JET operation with a new Be-wall, etc.

Keywords: Beryllium, ITER, JET, erosion, thermonuclear fusion, fusion devices, plasma-facing-components, plasma-wall-interactions, plasma contamination, neutron irradiation.

[§] on assignment to EUROfusion Consortium

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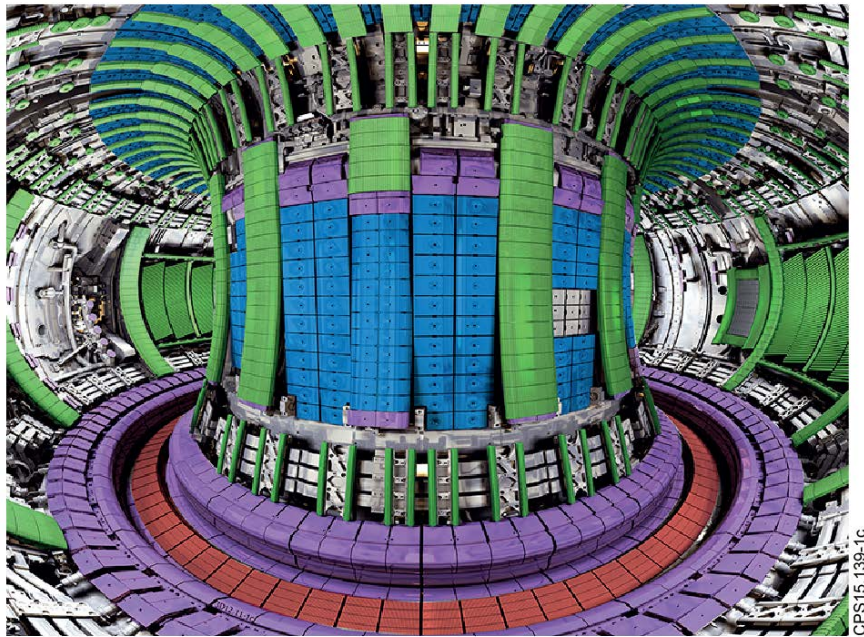
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6.2 Design of the beryllium ITER-like Wall at JET

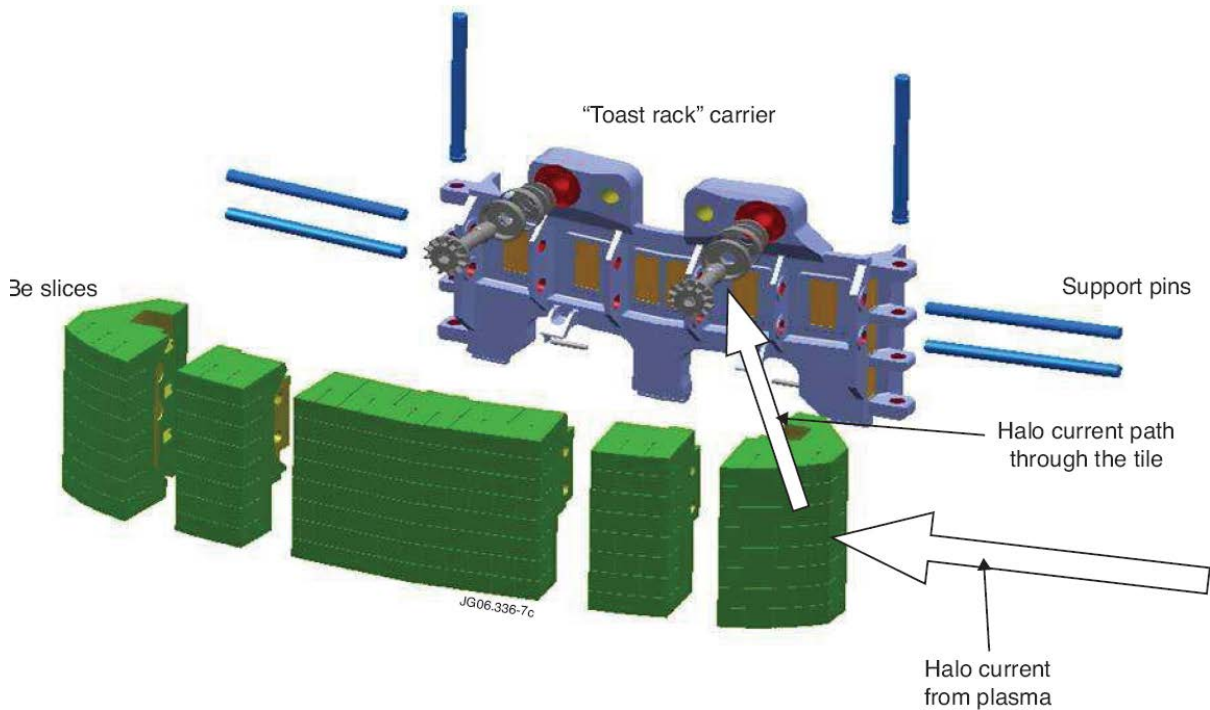
The aim of the ITER-like Wall project at JET [2] was to replace the pre-existing carbon fibre composite tiles with the plasma facing material combination now selected for ITER [3]. A key aim of project was to verify the prediction that the beryllium main chamber wall and tungsten divertor would deliver at least an order of magnitude reduction in tritium retention in ITER compared to an all carbon wall [J.Roth, et al., *Plasma Phys. Control. Fusion* 50 (2008) 103001 (20pp)]. The ITER-like Wall was almost entirely installed using remote handling and was completed in 2011 [4]. The strategy for the first phase of exploitation was focused on operation within the limits set by the new wall materials so that the impact on plasma scenarios and plasma-wall interactions could be studied [5]. The actual campaigns delivered all that was hoped plus some surprising new results [6]. Following an intervention to remove long term samples for surface analysis [7], the next phase of operation with JET ITER-like Wall started in 2013 and was focused on exploring the path to increased performance of the scenarios through increases in plasma current, heating power and energy confinement. Specific issues of major concern for ITER such as tungsten melting, disruptions and runaway electron mitigation [8] were also studied and the results have proved very influential [9]. Equally important to the scientific mission has been the opportunity to develop fully integrated scenarios and control schemes for protecting the wall [10]. The project is therefore providing essential information for interpreting material behaviour in ITER and a sound technical basis for guiding the development of ITER scenarios.

The design layout, the main engineering challenges and the operational limits of the JET ITER-like Wall are discussed elsewhere (see for example [11, 12,13, 14]). Fig. 20 shows the material layout implemented in JET. In addition to inertially cooled bulk Be tiles, recessed areas between the inner wall limiters and upper dump plate tiles were clad with Inconel with an 8 μ m coating of Be [15]. It must be noted that the existing JET wall relies on a series of discrete poloidal limiters whereas at the moment ITER relies on a plasma conforming wall. The electrical resistivity of Be $\approx 0.08 \mu\Omega m$ is more than a hundred times lower than that of CFC ($\approx 10 \mu\Omega m$). In order to keep the eddy current loads developed in the beryllium tiles within the loads developed by the CFC tiles and for which the tile supports, which had to be reused, had been designed, these tiles had to be sliced [13]. The chosen design has vertical slices with a large central block and 1 to 3 slices on each side, depending on the toroidal extent of the tile assembly, supported on a carrier via pins (see Fig. 21). The design is defined by the balance between conflicting requirements of eddy currents (avoidance of large low resistance loops) and power handling (minimum number of vertical cuts to be shadowed). The problems associated with the design of the JET beryllium tiles (power handling capacity and disruption induced eddy currents) are discussed in detail elsewhere. See for example [14], where the engineering limits of the W-coated CFC tiles and the bulk-W tile assemblies are also discussed and references to other papers dealing with these are listed.



- Bulk Be PFCs
- Be-coated inconel PFCs
- Bulk W
- W-coated CFC PFCs

Fig. 20 Materials configuration of the JET ITER-like Wall. The W-coated CFC tiles in the main chamber are recessed with respect to the bulk Be limiters.



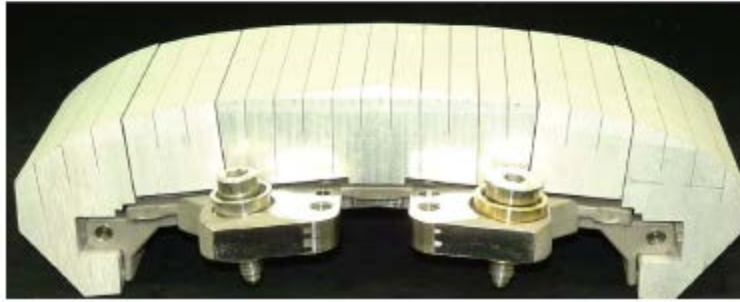


Fig. 21 Inner wall guard limiter tile (exploded view, top, and prototype, bottom). The five castellated Be slices have inter-slice and outer slice internal toroidal edges ski-slope shadowed. The slices are held on an Inconel carrier by pins which allow bowing under thermal load. The RH bolts are designed to be shadowed by the next installed tile. (Reproduced with permission from [14].)

6.3 Recent experience from JET operation with a full Be- Wall

In this section we summarise the experience of JET operation with its ITER-like Wall. Our focus is on those aspects where the beryllium wall played a dominant role. At the time of writing, there have been two main JET operational phases for the ITER-like Wall with interleaving shutdowns for removal of long term samples for surface analysis. The strategies for these campaigns and main outcomes, which cover all aspects from impact on energy confinement to issues related to tungsten, are described in [6,16,17,18]. Some of the physics outcomes such as reduced pedestal pressure in typical ELMy H-modes were not expected. Therefore pedestal studies have been carried out at JET with the main objective of addressing the physics responsible for the decrease in H-mode pedestal confinement observed in the initial phase of JET-ILW operation [C.F. Maggi et al, submitted for publication in the Proceedings of the 25th IAEA Fusion Energy Conference, St Petersburg, Russia “Pedestal Confinement and Stability in JET-ILW ELMy H-mode Scenarios”]. The effects of neutral recycling, plasma beta, plasma triangularity and nitrogen seeding on pedestal confinement and stability have been compared with predictions made on the basis of the Peeling-Ballooning paradigm. It was found that low neutral recycling, achieved either by low deuterium gas injection rates or by divertor configurations with optimum pumping, and high-beta are necessary conditions for good pedestal (and core) confinement. Under such conditions the pedestals are consistent with the Peeling-Ballooning paradigm. In contrast, under conditions characterised by high neutral recycling, additional physics is required in the pedestal model to explain the onset of the ELM instability. The physics mechanism leading to an increase in electron temperature in the pedestal with nitrogen seeding in high triangularity JET ITER-like Wall H-mode is not yet understood. The changes in the JET wall composition from CFC to beryllium/tungsten suggests the importance of the role of neutral recycling, low-Z impurities and scrape-off-layer physics in pedestal stability are ingredients that are not currently included in pedestal models. These aspects need to be addressed before a full predictive capability of the pedestal height can be achieved. Although not yet understood, the physics is probably not a direct consequence of the Be wall but rather due to the absence of carbon and possibly also the constraints imposed by the tungsten divertor leading to higher gas fuelling. For this reason we do not review these results here.

JET is now working towards a campaign utilising the 50/50 DT fuel mix required by ITER. Extrapolation of current discharge data to a 50/50 DT fuel mix predicts that about 15MW of fusion power could be produced. This target could be achieved in stationary conditions for about 5s, rather than transiently as in the 1997 JET DT experiment, corresponding to a total produced fusion energy of 75MJ, at 3.5MA/3.45T with 39MW of auxiliary power [F. Romanelli and on behalf of JET Contributors 2015 *Nucl. Fusion* **55** 104001 doi:10.1088/0029-5515/55/10/104001]. The high accounting accuracies possible in closed cycle operation with tritium will allow the most accurate global determination of total long term retained tritium in JET yet with the Be/W dominated environment of the ITER-like Wall.

6.3.1 Conditioning, breakdown and residual carbon

After installation of the JET ITER-like Wall (JET-ILW) the vessel was baked to 320°C, glow discharge cleaning in deuterium was carried out for about 100 hours and the temperature reduced to 200°C for first plasma [19]. Breakdown was achieved at the first attempt and the plasma current rose to about 1MA and lasted 15s. When JET had an all carbon wall, re-establishing reliable breakdown and building up the plasma current to the mega amp level required for routine operation took weeks including repeated cycles of conditioning. Subsequent operation with the JET-ILW was also notable for an almost total absence of non-sustained breakdowns (NSBs) attributable to machine deconditioning rather than technical issues [20]. NSBs in the carbon wall phase of JET occurred 8.8% of the time and were mainly attributable to failure of the burn-through phase due to the ohmic input power being less than the radiated power. With the JET ITER-like wall there is much better control of the density during breakdown due to lower out-gassing and low carbon impurity levels associated with the beryllium wall. The few failures at burn-through that did occur have resulted from residual nitrogen levels following impurity seeding experiments. With the carbon wall, regular beryllium evaporations and glow discharge cleaning (GDC) were used to maintain good conditions for plasma breakdown but with the ITER-like wall these conditioning methods were not required (see Fig 22).

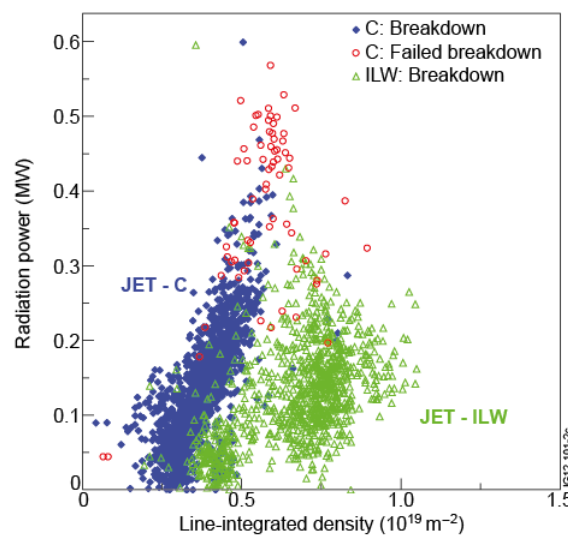


Fig. 22 Low radiated power and better density control have resulted in much more reliable breakdown with the ITER-like Wall compared to the carbon wall (JET-C). Figure reproduced from [6].

6.3.2 Plasma purity and material migration

A strong reduction in carbon content by a factor ~ 20 and $Z_{\text{eff}} \sim 1.2$ was observed from first operation with the ITER-like Wall and oxygen impurities were also very low to the extent that machine conditioning after first plasma was not required [21]. Overall beryllium deposition rates in the divertor, measured by surface analysis, with the carbon wall were more than an order magnitude higher than are now observed with the ITER-like Wall [7]. Even more dramatic was the change in the shadowed inner divertor corner leading to the pump duct had about two orders of magnitude less deposition with the ITER-like Wall. These reductions are due to the fact that carbon chemical sputtering can occur at very low hydrogen isotope impact energies leading to a large main chamber source even in diverted plasmas. The current understanding of the beryllium migration in diverted plasma configurations with the ITER-like Wall can be described as follows [22]: neutral Be atoms are eroded from the main chamber walls by physical sputtering and BeD molecules are produced by chemically assisted physical sputtering (CAPS) in the same areas. Both, Be and BeD, enter the plasma, dissociate in the case of the molecules, then ionise and are transported by the strong plasma flows towards the inner divertor where significant deposition of Be occurs on the tungsten surfaces of the inner divertor. The contribution from CAPS has been assessed in limiter plasmas [23] where it was deduced from the surface temperature dependence of the effective Be sputtering yield (Fig. 23).

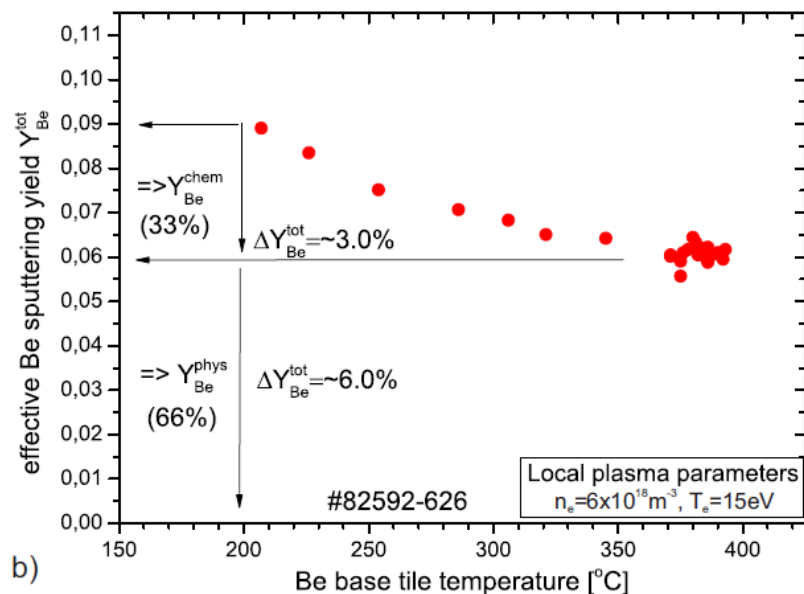


Figure 23 Be sputtering yield deduced from repeated identical limiter discharges with JET ITER-like Wall in which the surface temperature ratcheted up. Details can be found in [23].

Post-mortem analysis has revealed that the majority of all deposition is found on top of the apron of the inner divertor away from the strike point. The described migration process in the main chamber probably occurs in several steps as suggested by the diagonal form of the redistribution matrix in WallDYN simulations [21]. The observed net deposition location for Be depends on the local balance of erosion and deposition flux. Good agreement has been obtained between the deposition pattern as measured by surface analysis [7] and the WallDYN code results for the JET ITER-like Wall. The comparison of the model results for fuel retention with JET gas balance experiments [24] is also good and details of both aspects can be found in [25]. Further transport of Be from the inner divertor apron is strongly hindered because local plasma conditions do not provide enough energetic deuterons to re-

erode the deposited Be layer. This layer therefore builds up and incorporates the majority of the retained fuel. This behaviour is very different to the carbon wall phase of JET where chemical erosion and multiple step transport of ten or more re-erosion cycles occurred, thus leading to a covering of the whole inner divertor by carbon layers which were especially thick in areas shadowed from the plasma.

6.3.3 Fuel retention

The prediction of a strong reduction in tritium retention if ITER were built with a beryllium wall as opposed to the more traditional carbon wall has been one of the strongest arguments in its favour. However, until the ITER-like wall was installed in JET, the predicted reduction of over an order of magnitude [26] had not been proven in a tokamak. The retention rates for JET carbon wall and ITER-like Wall measured by gas balances for different plasma regimes are shown in figure 24 [24]. The retention rate is found to be 10-20 times lower than with an all carbon wall.

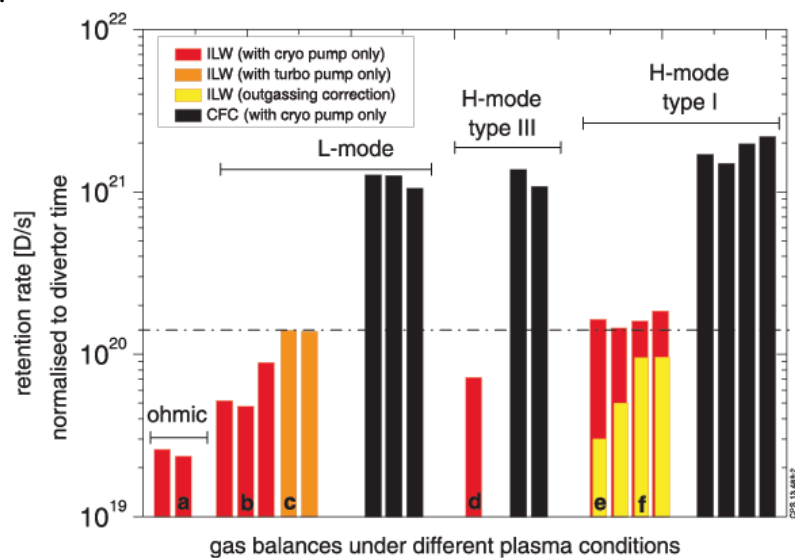


Figure 24. Measured D retention rates (logarithmic scale) for different plasma and confinement conditions in JET with the ILW and related JET-C references. Global gas balances are performed in the JET-C case solely with cryogenic pumping (black), and in the JET-ILW case with either the use of turbo-molecular (orange) or cryogenic pumping (red). The longer inter-shot outgassing period after H-mode discharges with the JET-ILW in comparison with JET-C leads to a reduction in the retention rate (yellow) owing to pumping of neutrals by the NBI cryogenic system. Reproduced from [24].

The long term deuterium outgassing behaviour of carbon and Be/W walls is very similar and varies as $t^{-0.7}$ for the ITER-like wall [24] and $t^{-0.8}$ for the carbon wall [27]. This dependency has been tracked for timescales of around 100 hours but must eventually slow down since the integral is infinite. In practical terms this means that the only way to determine the true long term retention rate, which is the primary concern for ITER, is by surface analysis of tiles removed from the vessel.

Using surface analysis on tiles from the beryllium wall and tungsten divertor, highest retention values were measured from regions with highest deposition. Lowest retention was obtained in erosion zones or areas with thin impurity layers where implantation might be the main retention mechanism. During the 2010–2012 experimental campaign, which was the

first for the ITER-like Wall, approximately 1.7×10^{26} D atoms were puffed into JET. Post-mortem analyses have mapped the distribution of the 3.9×10^{23} retained D atoms, which corresponds to 0.24% or 1.31g of retained D. This data can be converted into retention rates for main wall and divertor by assuming that the amount of deuterium retained in each area is proportional to the time spent in limiter or divertor configuration. Results are summarised in table 5 [28]. The global retention rate measured in this way is a factor 18 lower than for the carbon wall. Taking the main wall on its own however, the difference is only a factor 4. The retention rate measured by gas balance, figure 24 is an order of magnitude higher than has been deduced by surface analysis and the difference is attributed to outgassing.

Table 5 Fuel retention rates for JET carbon wall and ITER-like Wall determined by post-mortem analysis of samples from the first ITER-like campaign [28].

Retention rates:	Global (D/s)	Main wall (D/s)	Divertor (D/s)
JET-ILW 2010–2012	5.7×10^{18}	4.9×10^{18}	6.1×10^{18}
JET-C 2007-2009	100×10^{18}	19×10^{18}	130×10^{18}

Material migration and associated fuel retention in the JET ITER-like Wall have been simulated using the WallDYN code and a good match to the surface analysis data and gas balance data was obtained [29]. This has provided the basis for the extrapolations to ITER [30] shown in figure 25.

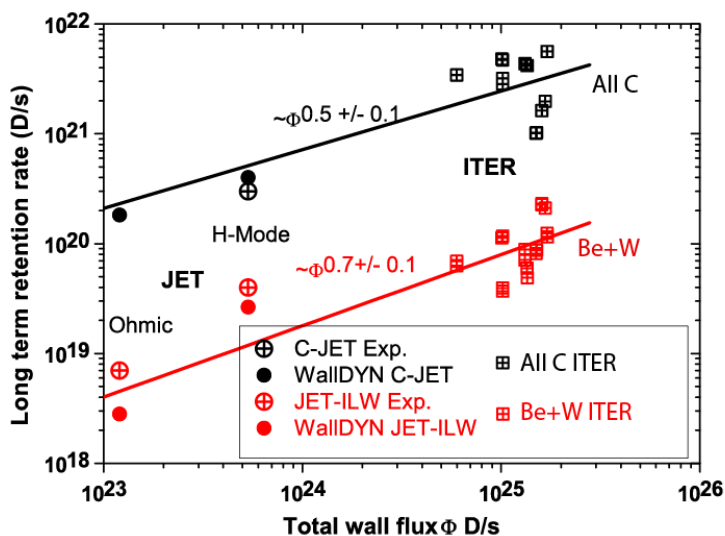


Figure 25: Long term retention rates simulated by WallDYN for all carbon or Be wall + W divertor for ITER and JET. Comparison is made to JET experimental data. Based on figure 4 from reference [30].

1.1.4 Disruptions and melt behaviour

Disruptions in tokamaks are defined by an event in which there is a rapid loss of plasma thermal energy followed by a quench of the plasma current with the potential to deliver high heat loads and high electromagnetic loads to the PFCs. Installation of the JET ITER-like Wall (ILW) had a dramatic effect on disruption behaviour [31]. As already discussed in section 6.3.1 the beryllium wall eliminated deconditioning by disruptions which with the carbon wall was a common cause of non-sustained breakdowns. Natural disruptions were also very different due to the beryllium main chamber wall with slow current quench and consequently slow loss of magnetic energy as shown in figure 26. With the carbon wall between 50% and

100% of the total available magnetic energy and plasma thermal energy was naturally radiated leading to low plasma temperatures, high resistivity and so fast current decay. However, with the new beryllium wall, natural disruptions only radiate 10%-50% of the available energy and so plasma temperature stays high, current quench times can be 100ms or more, figure 26. The negative consequences of this behaviour with JET ITER-like Wall were increased energy load on the beryllium PFCs and higher impulse on the vessel leading to higher dynamic forces. In JET, disruptions usually move upwards and inwards thus hitting the so called upper dump plate tiles which are an evenly distributed array of beryllium ribs protecting the top of the machine. The main wall in JET is typically baked at 200°C and the energy density Q (MJm^{-2}) required to raise the beryllium tiles to melting point is $\sim 20t^{0.5}$ where t is the event duration in seconds. Although disruptions with the beryllium wall are naturally slow, there are sometimes fast events due to loss of vertical stabilisation leading to a so called vertical displacement event (VDE) with current quench timescale of $\sim 10\text{ms}$. Due to the radiated fraction remaining low and the quench duration being short, VDEs have caused melting of the upper dump plate tiles, as shown in figure 27 from Ref. [32]. It is thought that dozens of such events contributed to the melt damage seen in this picture prior to tighter restrictions on test VDEs and unmitigated disruptions.

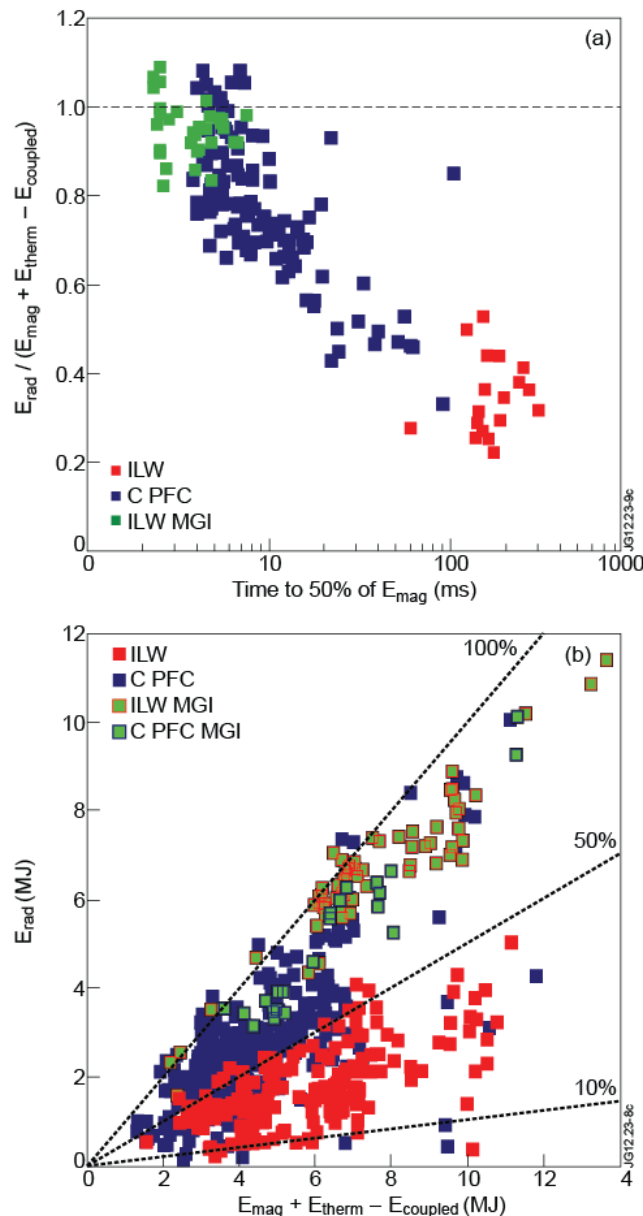


Figure 26 (from reference [31]): (a) Fraction of energy not coupled back into the toroidal conductors, that is radiated during the current quench versus the time it takes to reduce the magnetic energy ($\propto I_p^2$) to 50%. (b) Total energy radiated during the current quench phase versus the part of the total energy that is not coupled back into the toroidal conductors. In both graphs blue represents unmitigated disruptions with C PFCs, and red those with the ILW. In green those mitigated by MGI, which in (b) are differentiated between those with C PFCs (blue edge), and ILW (red edge).

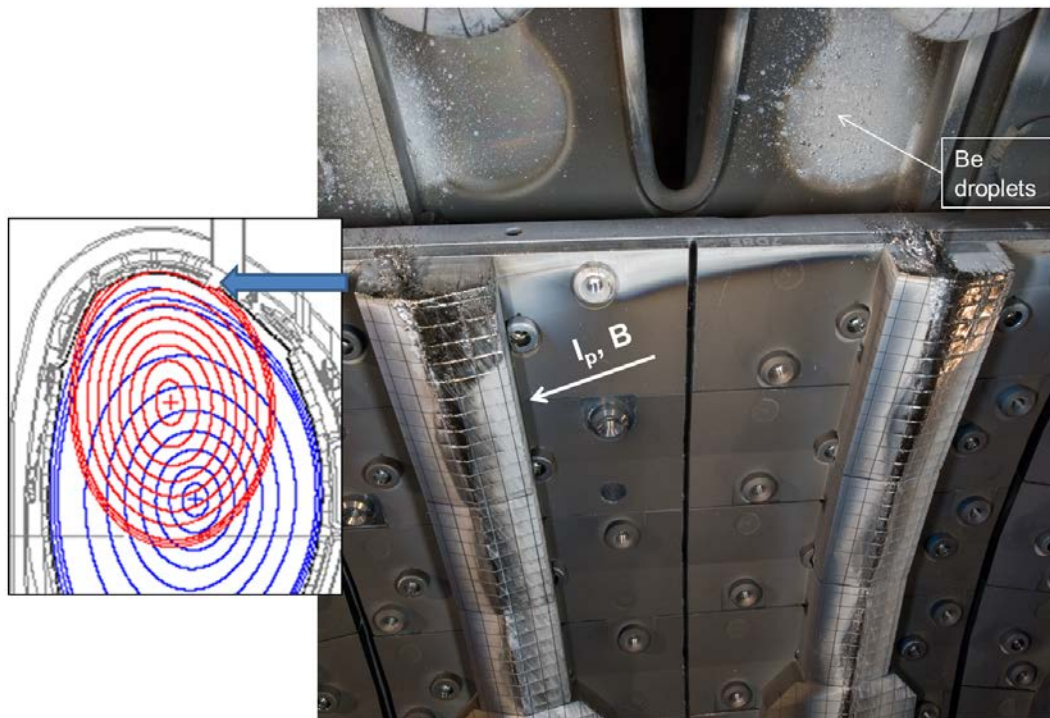


Figure 27 (from reference [32]): JET plasmas usually move upwards and inwards during disruptions and can cause melt damage to the beryllium upper dump plate tiles, which are a series of 64 ribs, particularly near the outer ends. The stress relieving castellations on the Be tiles are 12mm square. Electromagnetic forces drive the beryllium melt layer along the tile surfaces and up the end towards the top of the machine.

The melt layers produced by disruptions are relatively thin ($\ll 1\text{mm}$) and electromagnetic forces ($\mathbf{j} \times \mathbf{B}$) drive the layer along the surface and up the nearly vertical tile end [32]. The source of the current is thought to be the halo current associated with the disruption. Perpendicular current flowing from the melt layer surface to the tile beneath drives it along the surface. Detachment of the layer from the surface would create a parallel current and the resulting $\mathbf{j} \times \mathbf{B}$ force pushes the layer back towards the surface. Melt motion dominated by electromagnetic forces has been seen in the case of tungsten but in this case the source of the current is thought to be thermal electron emission [33]. For beryllium, the lower melting point means that even at boiling point the current density due to thermal electron emission would only be just sufficient to counteract the gravitational force [34].

In order to avoid such damage JET now requires massive gas injection (MGI) to be used to mitigate disruptions [35] in X-point plasmas above 2MA or 5 MJ total energy (poloidal magnetic plus kinetic). MGI with a mixture of 10% Ar in D₂ restores the high radiated fraction and rapid current quench as shown in figure 26 (b).

1.1.5 Runaway electron generation

During disruptions it is possible for relativistic runaway electrons (REs) to be accelerated to energies of up to 20MeV by the high electric fields associated with the current quench. The experience in JET with a carbon wall was that REs were observed in spontaneous disruption

(albeit rarely), but since installation of the ITER-like wall RE generation has only been achieved by deliberately injecting argon (or similarly high-Z impurities) via MGI. The current carried by such REs can exceed half the initial plasma current. REs are a concern for ITER because spontaneous RE generation is thought more likely in disruptions due to current scaling and availability of hot tail electrons. Also, experiments in JET have shown that while RE generation can be mitigated using MGI in the very early phases of RE formation, once the beam is accelerated MGI using $>2000 Pa m^3$ of Kr or Xe had no noticeable benefit [36]. At some point in the RE current plateau, an MHD instability occurs which dumps the REs onto the PFCs within a few ms. When REs are finally lost, the physics of the interaction with the Be tiles is governed by the range of relativistic electrons in solid beryllium which is $\sim 3mm/MeV$ [32]. For typical RE energies in JET ranges up to 6cm are possible and lead to volumetric heating on very short timescales not compatible with thermal diffusion from the surface. This results in deep melting as shown in figure 28. In this case a 1MA current plateau containing REs with average energy of around 12.9MeV was created. The impact on the inner wall limiter generated a number of melt spots and molten droplets were ejected. In contrast to other beryllium melt phenomena which involve surface heating, the melt pattern is toroidally very localized (one rib of tiles but not the next one) and does not exhibit the usual asymmetries you would associated with $\mathbf{j} \times \mathbf{B}$ driven melt layers (suggesting that electromagnetic forces do not play such a strong role). This may also be because of the electron range in beryllium so the electrons are travelling parallel to \mathbf{B} and so not generating much perpendicular current in the liquid layer. Modelling of this type of event is possible using codes such as ENDEP and MEMOS which can simulate the heating and melting of the beryllium by REs [37].

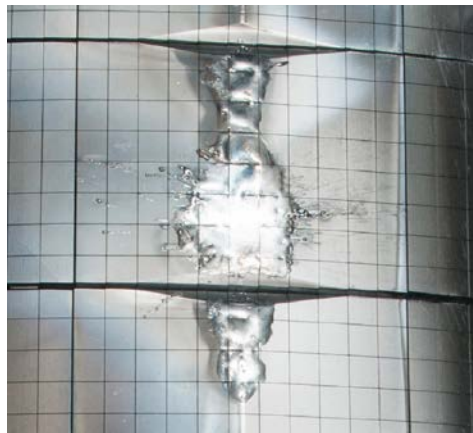


Figure 28 (from reference [32]): In-vessel image of melt damage due runaway electrons from pulse #86801 in which REs hit the tops of the inner wall limiters about 60ms after they are created. The RE plateau was $\sim 1MA$. The castellations are 12mm square.

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