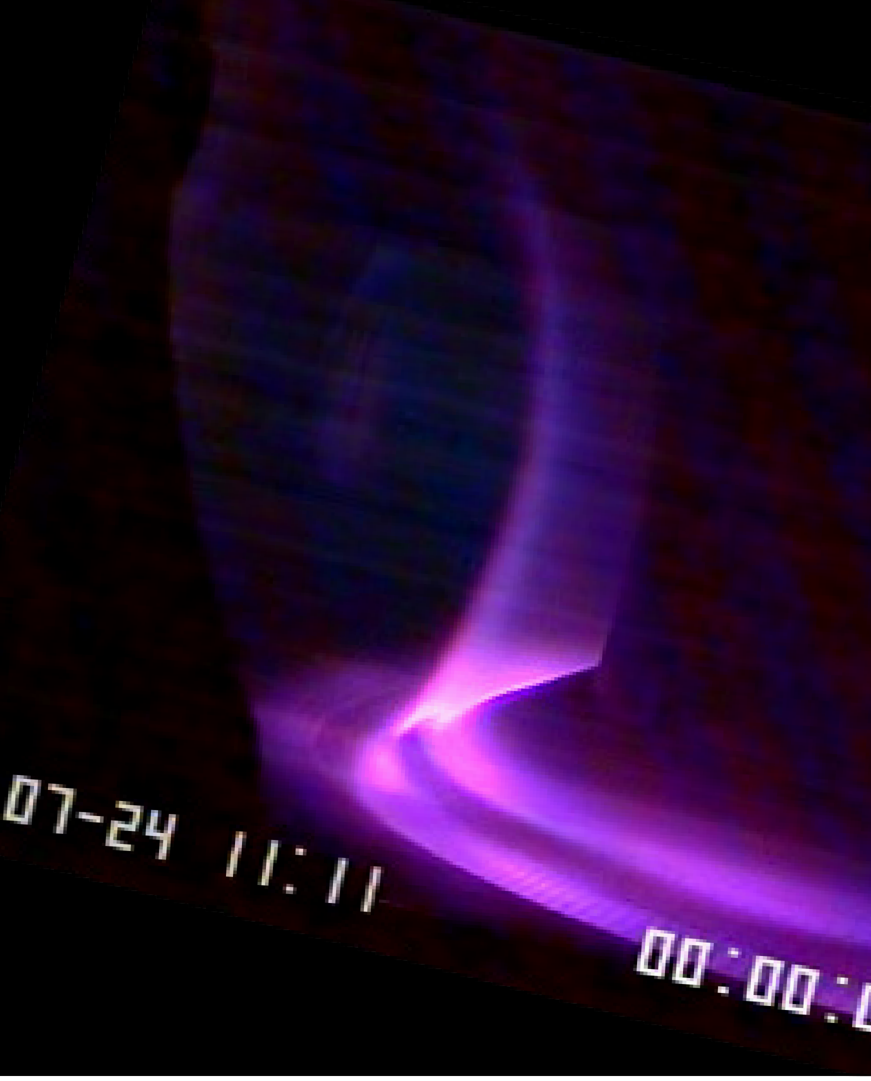
Institute for Magnetic Fusion Research

WEST

Newsletter N°25 - July 2019



Plasma is back for C4

Last runs before the installation of the full actively cooled divertor.

C4 was launched in July after the replacement of the cryogenic plant main compressor. During the commissioning phase, the **5 Radiofrequency antennas** have been successfully used on plasma for the first time while **1MA ohmic plasma** was achieved. The experiments have now started with notably the objective to rise the heating power to 10 MW (Lower Hybrid+Ion Cyclotron Heating) and enter into H-mode operation.

A four week helium phase will conclude the campaign from mid-September to mid-October. The effect of the helium on the tungsten plasma facing components will be closely examined in view of the non-nuclear phase of ITER operation. After this campaign, the startup lower divertor target will be removed to be replaced by the full actively cooled ITER-like divertor, which is expected to be in service from 2020 on for WEST long pulse/high fluence operation phase (WEST phase 2).

6th Governing Board meeting

For the sixth time, WEST international partners from Europe, China, Japan, Korea, USA and the ITER Organization gathered on May 17 at the Chateau de Cadarache.

The first part of the meeting was devoted to the overview of the C3 experimental campaign. The main scientific highlights were presented, showing good progress in terms of coupled power (5 MW) and pulse length (30 s achieved), although H mode operation was not reached yet. The results obtained ensued a significant number of publications, including invited and oral contributions in major fusion conferences. The plans for the coming C4 campaign were also presented.

The second part of the meeting was dedicated to the longer-term plans, for the second phase of WEST operation with its full actively cooled ITER like tungsten divertor. CEA indicated that the manufacturing of the divertor elements series is on going. The Governing Board members presented their proposals for participating in phase 2. In particular, three new diagnostics from US laboratories are in preparation while two new research proposals are under evaluation by the DOE. Finally, the ITER Organization outlined the R&D needs that WEST could address and pointed that an IO-CEA agreement has been signed to put in place a joint team for ITER divertor tests in WEST..

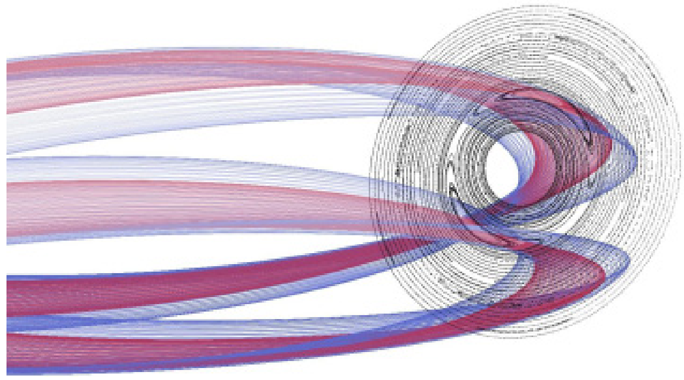
WEST Science — Magnetic stability

Macroscopic deformations of the magnetic configuration can spontaneously develop in a tokamak plasma; they determine its domain of existence.

The realization of a magnetically confined plasma in a toroidal device relies from the beginning on the understanding of its MagnetoHydroDynamic (MHD) properties. The most elementary consequence of it is the need for real-time control to maintain the hot plasma ring position in the toroidal vacuum chamber. At this level of description, the plasma organizes itself on nested magnetic surfaces of constant pressure. Nevertheless, non-axisymmetric deformations of these surfaces can develop on extremely short time scales if the pressure gradient or the plasma current density are too high. These conditions determine the limit of the operational domain of a tokamak. Any attempt to cross this boundary can result in a rapid loss of the plasma called disruption.

Inside the operational domain, MHD instabilities can still develop, but they become controllable. Magnetic reconnection tears the magnetic surfaces and forms islands, that can either saturate or relax periodically, like the so-called sawteeth in the plasma core. At the plasma edge, the pressure gradient can be large when turbulent transport vanishes spontaneously during the so-called H-mode, and periodic relaxations that are triggered there (the ELMs) need to be controlled in order to preserve the plasma facing materials. ELMs effects on WEST will be thoroughly scrutinized.

Finally, the interaction between fast particles and plasma waves can result in another family of MHD instability. This one is particularly relevant in the context of a burning plasma, where α-particles at 3.5 MeV will result from fusion reactions. Several techniques exist for the control of all these instabilities. Assessing their relevance for a fusion reactor is a key motivation for experimental and theoretical research in this area.



Three-dimensional structure of the magnetic surfaces of a tokamak plasma in the presence of magnetic island, superimposed on the Poincaré section (in black). Simulation with JOREK code.
More information (in French) in "Sciences en Fusion #2" (http://irfm.cea.fr/SciencesEnFusion/)

Each monoblock has a story to tell

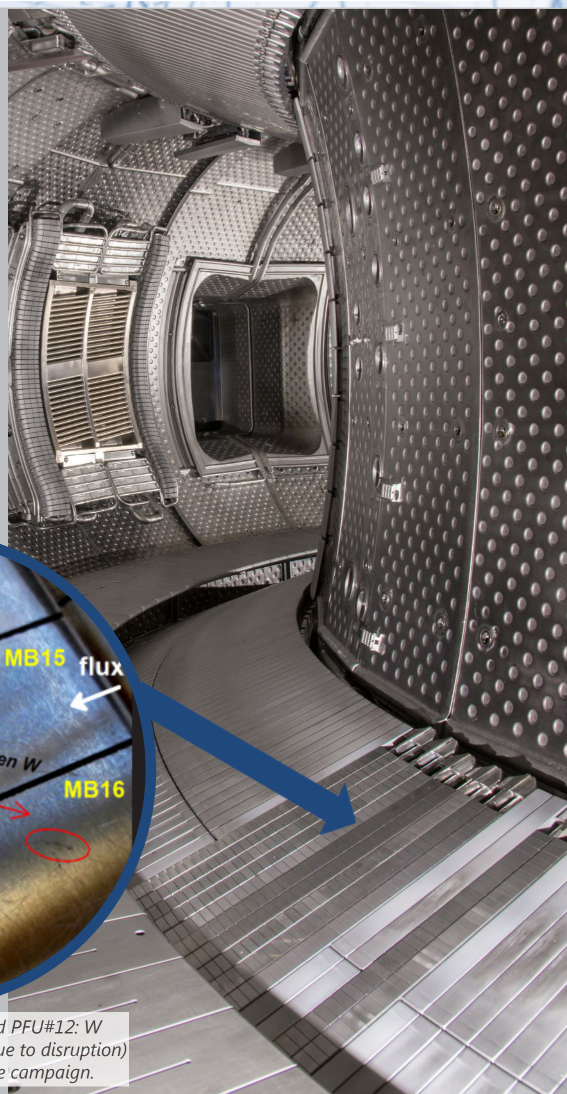
ITER divertor prototypes were exposed in WEST during the last experimental campaign (C3). While no component failure was observed, a wide range of damages was evidenced, in particular «optical» hot spots, which are predicted to occur in ITER.

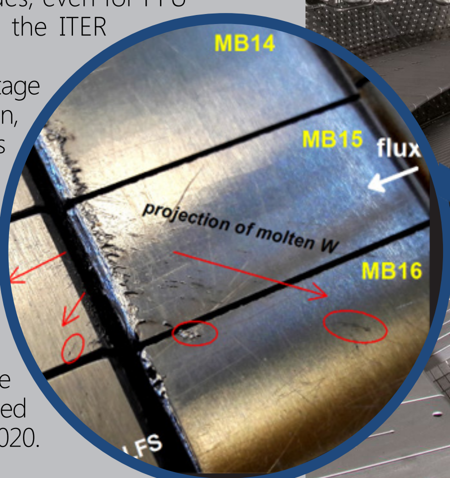
One of the west key missions is to assess the performance of the iter divertor in a tokamak environment and its impact on plasma operation. For the C3 experimental campaign, the 12 iter like actively cooled tungsten plasma facing units (PFU), provided by F4E, JADA and ASIPP, endured a substantial exposure time and energy (2.30 hours of plasma, 5 GJ of energy injected). They featured various manufacturing options, in particular in terms of monoblock geometry, with sharp or chamfered edges. PFU with significant radial misalignment were exposed (up to 0.8 mm of radial misalignment, to be compared with iter tolerances of 0.3 mm). In the course of plasma operations, the components were loaded with moderate steady state heat fluxes (L mode, peak parallel heat flux in the range 20-50 MW/m^2 at the outer strike point) combined with higher transient heat fluxes during plasma disruptions.

Local melting was observed on the top and side surface of the most misaligned PFU (misalignment of 0.8 mm, with sharp edges), but did not evolve during the campaign. This is presumably due to a single transient event. Damages (local melting and network of cracks) were observed on misaligned PFU side (both for sharp and chamfered edges), even far from the peak steady state heat flux area.

These «optical» hot spots (due to misalignment of toroidal gaps between PFU), which were predicted to occur in simulations for ITER, were also evidenced on the PFU sides, even for PFU aligned within the ITER tolerances.

At this early stage of operation, these damages do not affect plasma performances... to be continued in ongoing C4 and more extensively with the full actively cooled divertor from 2020.





Non-chamfered & misaligned PFU#12: W melting identified early in C3 (due to disruption) but did not evolve during the campaign.

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