Progress **report** 2020-23

Institut de Recherche sur la Fusion par confinement Magnétique

cea irfm

Progress report
2020-23

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PREAMBLE

This report summarizes the activities of IRFM (Institute for Magnetic Fusion Research), a CEA institute, for the period 2020-2023. The institute has a staff of 300 people, including 220 permanent employees, approximately 30 post-graduate and post-doctoral students, around 20 people on temporary contracts for specific projects and about 20 apprentices, all involved in two main research areas:

- Fusion plasma physics, addressed via theory and experiments on magnetic confinement fusion facilities, primarily the WEST tokamak, but also JET, JT-60SA and most other major facilities in Europe and worldwide, as well as numerical simulation of fusion plasmas, at various fidelity scales, with support for scientific calculation.
- The technology of fusion devices, addressed via operation of the WEST research facility and its various systems (cryomagnetic system, wave-based plasma heating system, vacuum and cooling system, command-control, etc.) and several specific technical platforms, as well as systems engineering for fusion, enabling the institute to develop, produce and qualify actively cooled plasma-facing components and to develop diagnostics for ITER.

July 2023: Some IRFM employees in front of the WEST tokamak

The period concerned covers the two years that were significantly impacted by the COVID19 pandemic, which required adaptation and flexibility from all our staff to maintain the ambitious objectives of the institute and obtain the results presented herein. Particular examples include the manufacture of elements for the ITER-grade divertor in 2020-2021, its qualification and installation at the end of 2021, the first plasma pulses in WEST with this new component in 2022, then the official inauguration of this new phase for WEST on June 6, 2023.

This report is not exhaustive, but aims to illustrate the activities of IRFM through a selection of key events during the period 2020-2023. After this preamble, the second part reviews the main upgrades of the WEST tokamak, some of which are then described in more detail in the third chapter on key events in "fusion plasma physics" and the fourth chapter on "fusion technologies." Chapter five summarizes the training and communication actions of IRFM. The institute's main collaborations are presented in the sixth section. A final chapter lists the various awards obtained during the period in question.

The results presented reflect the scope and value of the scientific activities deployed for the domestication of fusion energy. Interested readers can find details of the results published in the scientific literature by the institute and its partners for the period considered. This report does not address future perspectives. However, we will mention those offered by the new WEST configuration, which paves the way to very long-duration plasma pulses (a thousand seconds), improvement and innovation projects enabling WEST to be ever more pertinent in the preparation of the future operation of ITER, studies that are being launched on high critical temperature and very high field superconductive magnets, an increase in activities to improve our knowledge of tritium, the fuel of future fusion plants, and R&D actions related to the industrial sector to overcome certain scientific and technological obstacles to help accelerate the development of fusion energy. The contents of the next IRFM activity report are already looking exciting!

Finally, I would like to end this preamble by expressing my sincere thanks to all the staff of IRFM and its partners for their dedication to the success of the institute's activities.

I hope you enjoy reading this report!

Jérôme BUCALOSSI

Head of IRFM

WEST
DEVICE OPERATION

Years 2020-2023 saw the installation and commissioning of the WEST divertor with tungsten monoblock technology over its entire surface; this is the technology that will be deployed in ITER. This new configuration is the nominal configuration of WEST, with active cooling of all its in-vessel components. This configuration offers the capacity of extending the plasma pulses to almost continuous conditions (several minutes). The first plasmas were produced in December 2022 and the first experiment campaign was carried out in early 2023, supervised by EUROfusion. This first campaign exposed the divertor to plasma pulses up to 100 s long, achieving plasma fluence similar to that expected in an ITER plasma by the end of the campaign. Original observations, such as the growth of tungsten films, from this first exploration provided valuable information for the future operation of the ITER divertor. The first results, and those of the other studies carried out in 2023, suggest promising perspectives for the upcoming campaigns.

Implementation of the new ITER-grade technology (Figure 1) was disturbed by several events. Initially scheduled for September 2020, the new divertor was finally exposed to its first plasma in December 2022.

The manufacture and qualification of the 456 elements to equip the divertor were interrupted several times in 2020 because of the restrictions imposed during the Covid pandemic, initially at the manufacturing site in China, then in Cadarache during the acceptance and qualification phases. The installation of the elements on the 12 support plates making up the divertor, which was started at the same time as element acceptance, was interrupted to conduct the C5 experimental campaign in 2020. The first two completed sectors were installed and connected to the cooling circuit for this campaign, while the rest of the divertor was equipped with the inertial sectors that have been used since WEST was first started at the end of 2016. The challenge of this campaign was to perform the last experiments scheduled in the EUROfusion program for the period 2014-2020. The campaign was also an opportunity to replace the tungsten-coated tiles in the equatorial region of the limiters with boron nitride (BN) tiles to improve evaluation of the impact of the limiters equipped with high atomic number (high Z) materials compared with that of low Z material limiters in the plasma start-up (Figure 1). Campaign C5 thus concludes the first operating phase of WEST, with its initial bottom divertor equipped with inertial components made of tungsten. This first phase laid the foundations for producing plasma with a tungsten wall and divertor and performing the first tests on ITER-technology element prototypes. Some of the results are presented herein.

Figure 1: Inside WEST with the new ITER-grade bottom divertor and boron nitride tiles in the equatorial regions of the limiters

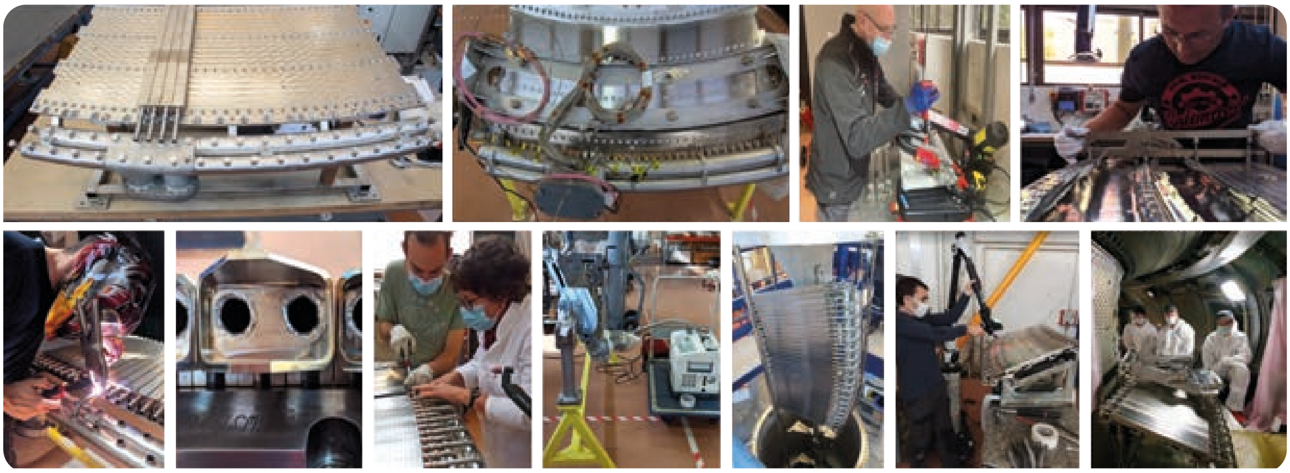


Figure 2: Assembly, welding, qualification and installation of the sectors of the new ITER-grade divertor and in-situ diagnostics

Figure 3: Elements of the actively cooled divertor in WEST. Each Plasma-Facing Unit (PFU) is equipped with 35 tungsten monoblocks. The WEST divertor has 456 PFUs. The luminosity differences reproduce the variable shapes, bevels and angles of the monoblocks.

Figure 4: Loading of the 20,000l liquid helium tank for factory repair

The last elements of the new divertor were delivered to IRFM in October 2020. The new divertor installation project restarted in February 2021, and was completed in July 2021. The procedures and tools implemented throughout the process (Figure 2) enabled the objectives to be fulfilled in terms of pre-characterization of the elements before exposure, and in terms of alignment of these elements on the divertor sectors in the machine (Figure 3). The alignment tolerances indicated for ITER, which are critical due to the grazing incidence of the field lines (just a few degrees), i.e., 0.3 mm between two adjacent elements, were respected.

At the end of April 2021, during the restart phase of the cryogenic system, conducted at the same time as the machine work, a leak was found on the internal nitrogen circuit of the 20,000l helium cryogenic tank, preventing magnet cooling. The repair work required disconnection of the tank (Figure 4), dispatch to the factory for diagnostic and repair, before return to Cadarache, reconnection to the cryogenic lines and reconditioning of the facility. This repair process took 9 months, and magnet cooling was resumed at the end of January 2022.

The plasma restart sequence was interrupted in 2022 by a series of water leaks that occurred during the vacuum vessel baking phases. No water leaks had interrupted operations previously. These micro-leaks affected the internal stainless steel protective panels (February 2022, August 2022), already in use on Tore Supra, and the bottom divertor baffle (May 2022), which had not been filled with water during the first operating phase. The two leaks on the internal protective panels appeared around the weld seams made porous by the thermal cycles and time. These weld seams were repaired. For the baffle, the leaks appeared at the CuCrZr-stainless steel junction between the component and the cooling circuit. This type of leak had already been observed several times during the qualification phases in a vacuum chamber, and the leaky components had been replaced each time with spare parts. It was therefore decided, after metallurgical analysis of the faulty components, not to fill the baffle with water for the first campaign with the new divertor, but to develop a more reliable, more permanent solution for the baffle's 288 junctions of this type.

In October 2022, repairs related to an incident on a cryogenic satellite valve required heating of the magnet, postponing the plasma start for two months. In December 2022, the first plasmas were finally produced with the new divertor (Figure 3). The conditioning sequences, concluded by a boronization phase, proved particularly effective as plasma pulses stable for several seconds on the divertor were created from the very first attempts (Figure 5). Campaign C6, which ended late 2022, was limited to systems commissioning. Campaign C7 started in early January 2023. It was mainly dedicated to the experiments within EUROfusion that could not be performed in 2021 and 2022, along with those scheduled for 2023.

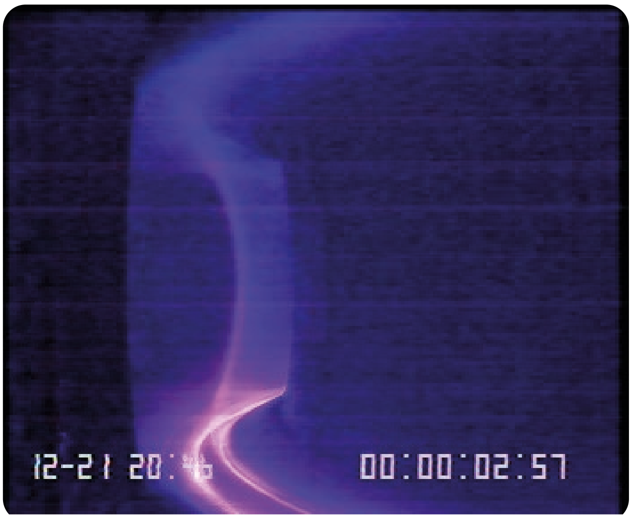


Figure 5: One of the first plasma pluses obtained with the new ITER-grade divertor in December 2022

After the technical incidents of 2021/mid-2022 involving key components of a fusion facility equipped with a superconducting magnet system and actively cooled components, campaign C7 was a huge success both technically and scientifically. All the scheduled experiments were carried out. One particular achievement was the production of plasma pulses lasting 100 s. The last three weeks were devoted to a "high-fluence" plasma experiment, which consisted in a series of identical one-minute discharges to achieve the fluence expected in an ITER plasma. The three weeks required to reproduce the fluence of a single ITER plasma illustrate the gap between today's tokamaks and ITER, and, to an even greater extent, future fusion plants. This campaign was the first chapter of ITER-grade divertor loading and study of its response to high-fluence plasmas.

The 2023 shutdown was an opportunity to repair the CuCrZr/stainless steel junctions of the baffle, replace several elements of the bottom divertor with elements manufactured by F4E, supplied by ITER Organization, install the ECRH antenna for the commissioning of the gyrotrons in 2024, install a specific internal protective panel for the new Thomson scattering diagnostic, and perform sectorization work on the cooling circuit to facilitate the detection of water leaks. The baffle repair work, started after several months of R&D efforts and qualification of the repair process, was completed quickly and, at the end of 2023, all the actively cooled components had been filled with water and were in their nominal configuration for the start of C8. Another leak was identified during the baking phase in the external limiter in a stainless steel nozzle; these stainless steel nozzles were all replaced. After this intervention, campaign C8 was limited to the restarting of the systems and preparation of C9, scheduled to begin in early 2024.

The old plasma-facing components protecting the internal vessel, used in Tore Supra, clearly demonstrated their limits. In fact, their extraction capacities were revised in consideration of the radiated power of the tungsten and it appeared that they were not all compatible with an injected power of 10MW lasting 1,000 s (10 GJ). The improvements required have been identified and several solutions are being investigated.

In spite of these numerous incidents, initial operation of the new divertor is very promising, as indicated by table 1, with a sum total of 5.8 hours of plasma pulses and an injected energy of approximately 44 GJ. The lower hybrid power injection system (3.7GHz) worked with a high level of reliability at 5 MW. The ionic cyclotronic frequency system, more difficult to implement in a tungsten environment, is still in the power ramp-up phase. At the beginning of 2024, the duration record has already been smashed with more than 6 minutes and 1.15GJ energy injected in a single plasma pulse. The first gyrotron 105 GHz is being tested in Karlsruhe and 2024 is expected to produce a plethora of results.

Campaign	Plasma number	Ip max. (kA)	Duration max. (s)	Total (s)	LH (pulse)	LH max. (MW)	Total LH W (MJ)	IC (pulse)	IC max. (MW)	Total IC W (MJ)	Boronization
C1-C4	3234	1004	55	21551	1445	5.4	17165.5	546	5.7	1247	16
C5 Nov20-Jan21	655	806	23	4630	280	5.4	2625	145	3.7	182	2
Phase 1	**3889**	**1004**	**55**	**26181**	**3170**	**5.4**	**19790.5**	**691**	**5.7**	**1429**	**18**
C6 Dec 22	47	807	14.8	295	0	0	0	0	0	0	1
C7 Jan23-Avr23	1113	741	101.25	19665	722	5	42981	104	4	507	1
C8 Dec 23	151	514	16.42	1083	75	3.8	496	4	0.1	0.03	1
Phase 2	**1311**	**807**	**101.25**	**21043**	**797**	**5**	**43477**	**108**	**4**	**507.03**	**4**

PLASMA **PHYSICS**

1. Introduction

CEA-IRFM develops and applies simulation codes to understand the phenomena involved in experiments and uses this understanding to propose optimized plasma scenarios. The topics addressed are broad and often overlap: turbulence, transport and confinement, plasma-wall interaction and magnetohydrodynamics. The results of these simulations have been published in prestigious international journals.

CEA-IRFM is committed to an initiative to apply the principles of FAIR data[1] within the magnetic confinement fusion community (FAIR4Fusion[2] project), using the IMAS[3] standard promoted by ITER Organization for its data, and contributes to international fusion databases. CEA-IRFM has also set up a center for remote participation in fusion experiments, in the context of the Broader Approach and future operation of the JT-60SA tokamak by European laboratories. The first tests of this center under real conditions were carried out on the WEST tokamak.

Broadcast of the inauguration of JT-60SA, live from the remote experimentation room "La Bergerie"

Références
[1] https://fr.wikipedia.org/wiki/Fair_data
[2] https://www.fair4fusion.eu/
[3] Integrated Modelling and Analysis Suite

2. Plasma Physics and Modeling

2.1. Demonstration of a link between magnetic chaos and current dynamics during disruptions

CEA-IRFM, in close collaboration with ITER Organization and several international laboratories, made a new breakthrough in the modeling and understanding of disruptions in tokamak plasmas in May 2023. For the first time, simulations reproduced the plasma current peak—a well-known characteristic feature of disruptions—and revealed the link between this peak and the magnetic chaos inside the plasma during the disruption.

In a tokamak, a disruption is a sudden termination of the plasma pulse caused by Magneto-HydroDynamic instability (MHD). Disruptions are a major topic of research because they can cause severe damages in large tokamaks. An international task force was created in 2018 to study this problem and prepare a disruption mitigation system for ITER (***https://www.ITER.org/newsline/-/3183)***. This involves efforts in terms of simulation, notably using nonlinear 3D MHD codes, like JOREK, historically developed at CEA-IRFM and now used in many other laboratories. The work with JOREK particularly concerns disruptions triggered by massive gaz injection.

Alongside studies for ITER, work is also being undertaken to validate simulations on current tokamaks. JOREK has thus been used to simulate a disruption triggered by a massive injection of argon atoms into the plasma in JET [1]. The simulations have been compared with experimental measurements, such as magnetic field or radiation field (bolometry), using synthetic diagnostics (measurements reconstituted by simulation). The simulations provide a good reproduction of the overall dynamics of the disruption and the destabilization of so-called "tearing" MHD modes, which significantly alters the magnetic topology. The magnetic surfaces, initially fitted one inside another like Russian dolls, are gradually destroyed by the appearance of structures called «magnetic islands» that grow and finally overlap, leading to magnetic chaos, which causes a violent change in the magnetic topology. This is illustrated in figure 1.

The first consequence of this overall magnetic chaos is that the heat contained within the plasma core is transported towards the edge in less than 1 ms, as shown on the electron temperature cross sections, Te, at the bottom of figure 1. However, this magnetic chaos also affects the distribution of electric current density, j_φ, whose cross sections are shown at the top of figure 1. These effects are mainly linked to the propagation of Alfvén waves along the field lines. These waves tend to spread the current density homogeneously throughout the chaos zone. Using an MHD theorem on the conservation of magnetic helicity (volume integral of A.B, where A is the potential vector and B the magnetic field), we show that this spreading leads to an increase in the total current carried by the plasma. This is actually what happens during the simulation and this increase corresponds to that observed in experiments, as shown in figure 2.

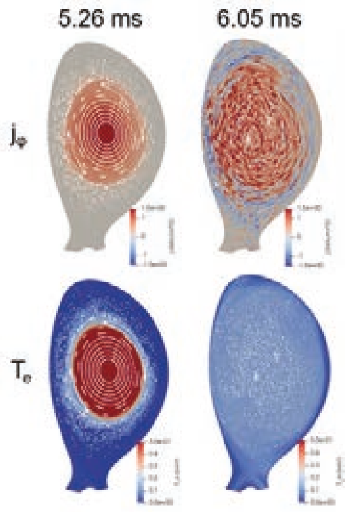


Figure. 1: Cross-sections of current density (top) and electron temperature (bottom) and magnetic topology (white dots representing Poincaré sections) at two successive moments (left and right) in a JOREK simulation of a disruption triggered by a massive injection of argon in JET. At t=5.26 ms, a chaotic zone exists at the edge of the plasma, characterized by a "random" distribution of white dots, while further inside, there are two magnetic islands, then a series of intact magnetic surfaces. At t=6.05 ms, 800 µs later, the chaotic zone has spread throughout the plasma.

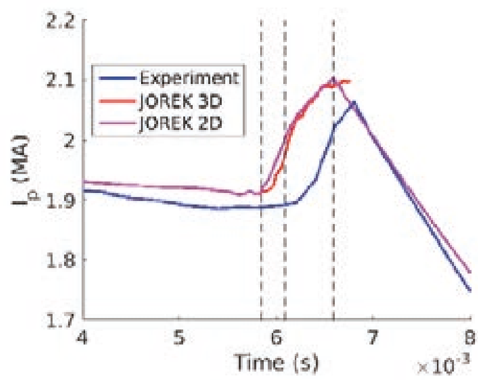


Figure 2 : Evolution of the plasma current during the experiment (blue) and during the JOREK simulation (red and cyan)

MHD simulations were previously unable to reproduce this characteristic trait of disruptions because of resolution and numerical stability limitations. The JOREK simulation presented in [1] is therefore a first, marking a major step forward in the validation of simulations and our understanding of the physics of disruptions.

References
[1] E. Nardon, K. Särkimäki, F.J. Artola, S. Sadouni, the JOREK team and JET Contributors, On the origin of the plasma current spike during a tokamak disruption and its relation with magnetic stochasticity, Nuclear Fusion 63 (2023) 056011

2.2. Plasma rotation prediction in ITER

Rotation control is a key challenge in obtaining stable, well-contained plasma in a future tokamak fusion power plant. Although energy neutral particle injectors enable partial control of the plasma in medium-sized tokamaks, the large plasma volume in ITER will limit the control capacity. However, rotation plays a fundamental role in plasma stability and quality of confinement. Inadequate control of this rotation can, in some cases, cause a loss of plasma confinement and damage to the machine. It is therefore important to determine whether phenomena inherent to the plasma enable the generation of plasma rotation. A recent publication in Physical Review Letters [1] shows that "first principle" simulations help understand how this rotation is generated.

One fascinating observation is that a tokamak plasma rotates, even when there is no external rotation source. This phenomenon is called "intrinsic rotation." There are two mechanisms that cause intrinsic rotation: the forces exerted by turbulence and the effects of "magnetic braking". In tokamaks, the latter is due to the fact that there are a finite number of coils generating the magnetic field and that this is therefore modulated in the toroidal direction, as shown in the figure below. This modulation is called "magnetic ripple."

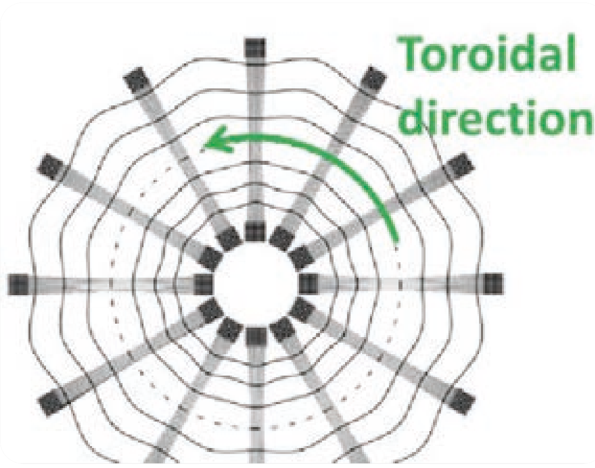


Figure 1 - Diagram of the top of a tokamak. The thick lines indicate the coils and the wavy lines show the shape of the field lines.

The ripple effect is not the same throughout the plasma and depends on the number and proximity of the coils, in particular.

The competition between these two mechanisms that cause plasma rotation has been investigated. The idea was to determine whether the ripple effect can dominate the turbulence effect and, if so, where. Based on previous work, a theoretical model taking into account these two effects was developed to determine the critical ripple amplitude above which the turbulence becomes sub-dominant in the control of intrinsic rotation. Its validity was corroborated in 2022 with the GYSELA code, using "gyrokinetic" simulations that take into account the two mechanisms self-consistently. The figure 2 shows the plasma velocity in the toroidal direction at each position with deactivation of the magnetic ripple in the simulation (left) and in the presence of the ripple (right). Without ripple, rotation is controlled by turbulence only and its direction gradually changes, becoming closer to the edge. However, rotation with magnetic ripple is oriented in a preferred direction.

Using the critical ripple expression, the first estimates for ITER seem to indicate that the ripple effect is not negligible close to the edge and should be taken into account in future simulations.

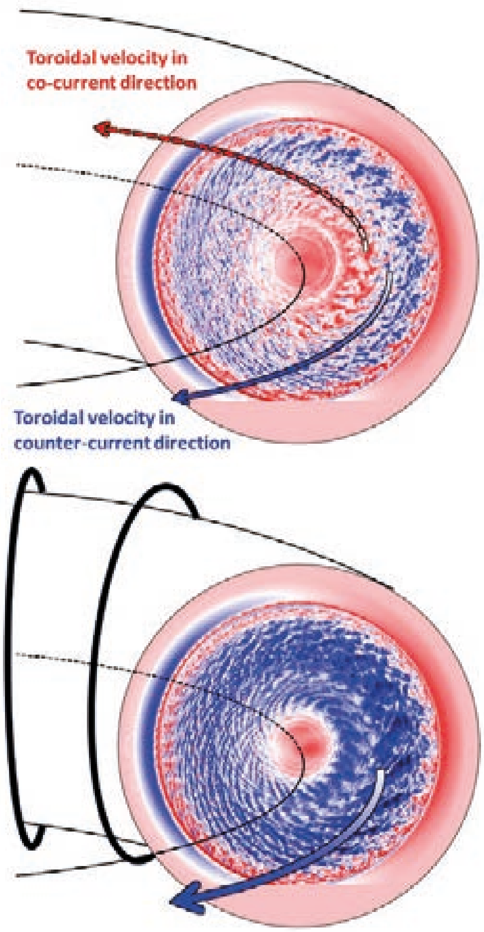


Figure 2: Rotation toroidale avec (bas) et sans (haut) ripple magnétique due au nombre fini de bobines générant le champ magnétique. Ces résultats de simulations obtenus avec GYSELA montrent que le ripple change fortement la rotation du plasma.

References :
[1] - R. Varennes et al, Phys. Rev. Lett. 128, 255002

2.3. Progress in predicting plasma-wall interaction in ITER

Control of plasma-wall interaction in a tokamak demands good prediction of the plasma's properties. This requires simulation of all the phenomena involved, particularly turbulence, which determines the transport of heat and particles. Thanks to an interdisciplinary collaboration project between CEA-IRFM and the M2P2 research laboratory (Aix-Marseille Université/CNRS/ Ecole Centrale de Marseille), an original self-consistent model of turbulent flows was introduced into the transport simulation codes in February 2023, offering new predictive capacities with an almost unchanged calculation time.

The scientific operation of ITER must be based on modeling tools that best describe the physical phenomena involved, with reasonable calculation times. Predicting the properties of the plasma in the boundary layer in interaction with the wall or edge plasma is critical for the development of a fusion plant. These properties are used to deduct the heat and particle flows that affect the wall elements, thus optimizing their design and experimental implementation, and their influence on the core plasma and plasma performance.

The phenomena at play in the boundary layer are currently modeled using a set of numerical tools that range from a short execution time with a plasma description based on reduced models, to tools that take into account physical phenomena characterized by much shorter space and time scales, such as turbulence, but whose execution times are much longer. In this context, and inspired by achievements in fluid mechanics, an approach has been developed to improve the prediction capacities of so-called transport codes for the edge plasma without reducing their effectiveness in terms of calculation time.

These transport codes are often used to interpret experiments in magnetic fusion machines. They enable the processing of plasma-wall interaction phenomena and plasma dynamics in the direction perpendicular to the magnetic field. However, to ensure a reasonable execution time with respect to the so-called turbulence codes (a few days as opposed to a few months), plasma dynamics perpendicular to the magnetic field, such as scattering, are processed with ad hoc coefficients selected to reproduce experimental profiles; it is impossible to predict the value of these coefficients for future experiments. In this mainly interpretive approach, the value of the transport coefficient at each point in space is a free parameter.

To reduce the number of free parameters, a self-consistent determination of transport coefficients in the direction perpendicular to the magnetic field was elaborated using simplified equations with respect to a detailed resolution of the turbulence. The idea was to determine the turbulent energy and its characteristic dissipation time using a two-field model with two transport equations, one for turbulent kinetic energy "k" and, according to the approach widely used in fluid mechanics, another for "epsilon," which plays the role of the turbulent energy dissipation. The local dynamics of these two fields recall prey-predator models and amplitude equations. These fields enable evaluation of a diffusion coefficient proportional to "k*k/epsilon" at any point in space. By considerably reducing the number of free parameters in the model, it offers new prediction capacities for transport codes for an almost unchanged calculation time.

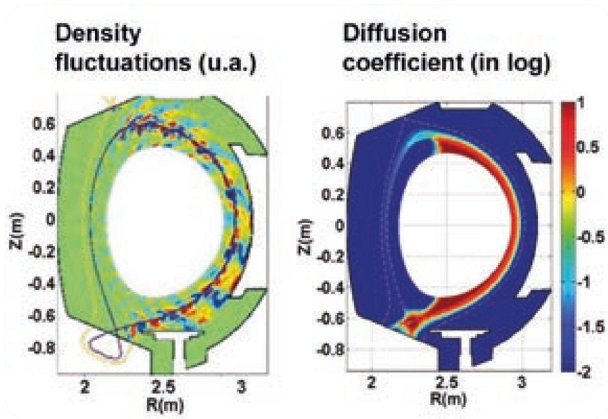


Figure 1: Comparison between a "high-fidelity" 3D WEST simulation with the SOLEDGE code in its turbulent 3D version (left panel), and a 2D simulation using SOLEDGE in its "transport" version coupled with a k-epsilon model. Note the capacity of the "reduced" k-epsilon model (right panel) to reproduce the spatial distribution of material transport, which is higher when the turbulence code predicts larger density fluctuations (left panel).

The k-epsilon model was implemented in the SOLEDGE code, which simulates the edge plasma of the tokamak. The goal was to compare the results obtained using this code with those obtained from turbulent simulations and existing experimental data, notably in the TCV (Tokamak à Configuration Variable - Variable Configuration Tokamak) and WEST tokamaks. These initial simulations reproduced certain overall characteristics of turbulent transport, such as spatial distribution (see Figure), as well as local characteristics, such as the width of the boundary layer, a key parameter in power extraction. It was predicted with a deviation of just 20%, i.e., much more accurately than by using a scaling law.

These results give credit to the k-epsilon approach, opening perspectives for the elaboration of new divertor configurations and a better understanding of the essential phenomena at play at the plasma edge.

Reference :

S. Baschetti, H. Bufferand, G. Ciraolo, Ph. Ghendrih, E. Serre, P. Tamain and the WEST team «Self-consistent cross-field transport model for core and edge plasma transport», Nucl. Fusion, 61, 106020 (2021)

2.4. Stabilization of tokamak plasma turbulence using alpha particles

D-T fusion reactions produce high-energy ions (alpha particles). These are considered to be potentially noxious in the confined plasmas of tokamaks because they can interact with the Alfvén waves and cause strong perturbations. A recent study by CEA-IRFM and published in Nature Physics [1] shows that MeV-range ions can effectively stabilize the turbulence and thus reduce the transport of heat. These results suggest that plasmas with a high proportion of fusion-born alpha particles with an energy of 3.5 MeV, as expected in ITER, could bring performance enhancements superior to expectations.

The use of very high-energy ions to heat the plasma by Coulomb collision with the thermal ions is a well-established concept in tokamak confined plasmas. The high-energy ions are introduced into the plasmas by accelerators (Neutral Beam Injection, NBI), or produced by resonance with specific waves injected from outside (Ion Cyclotron Resonance Heating, ICRH). The study of these ions is a fundamental topic for tokamaks because in future fusion plants, the Deuterium-Tritium reactions will produce 3.5 MeV alpha particles, which will be the main plasma heating mechanism.

Experiments have shown that the presence of high-energy ions (from 100 keV to several MeV) increases the temperature of the plasma (typically by about 10 keV, i.e., 100 million degrees), which is necessary to attain the temperatures required to generate sufficient fusion power. However, at the same time, the high-energy ions can resonate with specific waves, called Alfvén waves, which can capture a large amount of their energy and thus damage the plasma confinement.

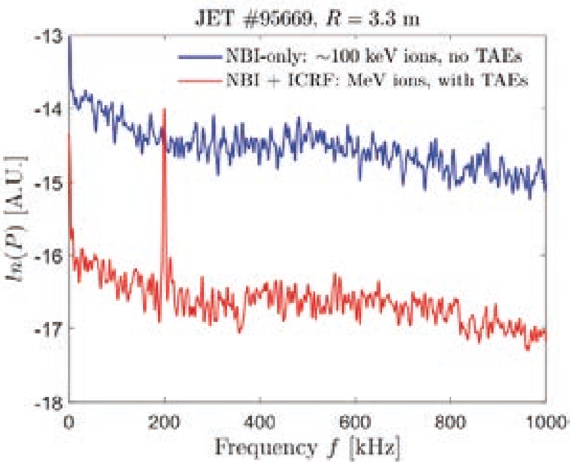


Figure 1: Spectrum of plasma density fluctuations with ions of ~100 keV injected by NBI (blue), and with ~MeV energy ions generated with ICRH (red). The intensity of the fluctuations is strongly reduced by a factor of about 10 at all frequencies, except for the Alfvén wave frequency (peak around 200 kHz). This means that the energy losses in the plasma, due to density fluctuations, are reduced in the presence of ~MeV energy ions.

In 2022, CEA-IRFM researchers analyzed the plasmas produced in the European JET tokamak with a large population of MeV-range ions, accelerated by ICRH heating waves and in the presence of strong Alfvén perturbations. The energy containment was compared with that of equivalent plasmas heated only by neutral beam injection (NBI), which generates lower energy ions (around 100 keV). Surprisingly, energy confinement is 40% higher in the presence of MeV-range ions. A reflectometry diagnostic confirmed the improved confinement in the presence of these MeV ions, clearly related to turbulence. It revealed a strong reduction in plasma density fluctuations, which are the main cause of energy losses in a tokamak (Figure 1).

Multi-scale analyses of turbulence and transport were carried out with multi-scale simulations, which are required to take into account both the small scales of turbulence and the large-scale fluctuations of Alfvén wave perturbations. This experimental behavior was well reproduced by simulations, and the physical mechanism causing the improved containment was identified. The zonal flows, similar to the famous rings of Jupiter (figure 2), play a key role. There is a clear interaction between the Alfvén scales and the zonal flows, which almost totally eliminates the transport of energy by the ions if the pressure of the high-energy ions is high enough to destabilize the Alfvén waves.

These promising results suggest that previously unexpected favorable conditions could be obtained in ITER and future tokamaks in the presence of very high-energy alpha particles.

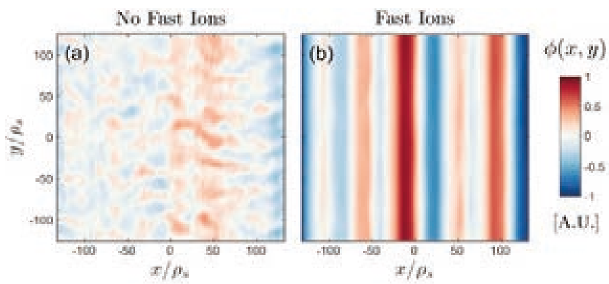


Figure 2: Simulations of electrostatic potential fluctuations without fast ions (a) and with fast ions (b). x is the radial direction, y the poloidal direction, and ρs the radius of the mm-range cyclotronic movement of the ions around the field lines (Larmor radius). In case (b), the radial propagation of fluctuations has almost disappeared and the fluctuations therefore no longer generate energy transport in the radial direction. The pattern observed evokes the zonal flows of Jupiter's atmosphere (Figure 2).

References

[1] - S. Mazzi et. al. « Enhanced performance in fusion plasmas through turbulence suppression by megaelectronvolt ions », Nature Physics (2022) ; doi.org/10.1038/s41567-022-01626-8

2.5. Tungsten and fusion: cohabitation without contamination

In 2020, physicists from CEA-IRFM looked into the issue of transporting atoms of tungsten, which is the material that covers the walls of the tokamak vacuum vessel; it must not contaminate the plasma. Its build-up in the core of the plasma causes the high radiative losses often observed on the European JET tokamak and on Asdex-Upgrade in Germany. On the WEST tokamak, there is an explanation for the rarity of this type of event.

Operation of a magnetic confinement fusion reactor implies strong constraints for the materials selected to cope with plasma heated to several million degrees. They must withstand high heat flows without absorbing the radioactive tritium that is part of the reactive mixture. Tungsten, a metal that melts at very high temperature, is suitable for this environment, but it can tend to build up in the center of the plasma. There, it can cause high radiative losses that are not compatible with the temperature required to maintain the nuclear reactions, and can even extinguish the plasma completely if they exceed the heating power. On the WEST tokamak, unlike many other machines, this build-up phenomenon is extremely rare.

Tungsten, like other particles in the plasma, is transported via two channels, one collisional channel and one turbulent channel. The latter is generally largely dominant, but for heavy ions (such as tungsten), which attain high ionization degrees at typical fusion plasma temperatures (~10 keV), two mechanisms can make the collisional transport much higher: rotation of the plasma, which exerts a centrifugal force proportional to its mass, and asymmetries in the electrostatic potential, which exert a force proportional to the charge of the ions. In many tokamaks, the plasma is heated by neutral beam injection. This heating system generates strong rotation, favoring the build-up of impurities on the external side due to the centrifugal force. However, in usual conditions, the flux of impurities enters from the external side and exits from the internal side of the plasma. Having more impurities on the external side therefore causes an overall incoming flux, and a build-up of impurities in the center of the plasma. The problem of this build-up is significant and has been observed on the JET and Asdex Upgrade tokamaks, for example. However, in WEST, the plasma is heated by radio-frequency waves and its rotation is weak. In ITER, the situation will be similar to that of WEST because the pulse created by the high-energy neutral beams (1 MeV) will be limited. In these conditions, the turbulent transport can continue to dominate and the tungsten will not build up. This has been confirmed by CEA-IRFM researchers with calculations using turbulence and collisional transport codes applied to the experimental conditions of WEST [1], which constitutes an important result for ITER operation and the design of future fusion reactors.

In the context of this work, an original approach to evaluate collisional transport coefficients was developed and enables six orders of magnitude to be saved on the calculation time thanks to a few simplifications validated by a more comprehensive code [2]. This means it can be implemented in a plasma evolution simulator created in 2021 for a European project. It also facilitates investigation into the few cases of build-up that remain to be explained and for which electrostatic potential is one of the options being considered.

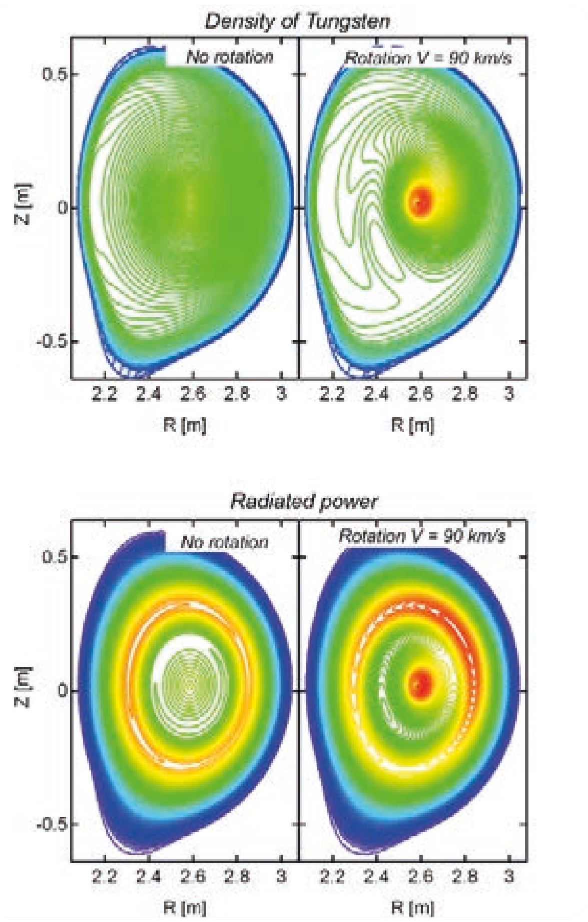


Figure 1: Simulation for a WEST case
Cross-section of the density of tungsten (top) and radiated power (bottom)
- for a static plasma (left, similar to the experiment situation)
- and in the hypothetical situation of a strong rotation in the direction perpendicular to the cross section (right).
The red color indicates a greater density and radiation in a rotating plasma (FACIT code)

Références :

[1] *X. Yang et al, Core tungsten transport in WEST long pulse L-mode plasmas, Nuclear Fusion 60 086012 (2020)*

[2] *P. Maget et al, An analytic model for the collisional transport and poloidal asymmetry distribution of impurities in tokamak plasmas. Plasma Physics and Controlled Fusion, 62 105001 (2020)*

2.6. FAIR4Fusion project: WEST pioneering open data

The capacity to provide a large scientific community with easy, efficient access to data is essential for the development of scientific research in many areas. The FAIR[1] approach aims to generate data that are "findable, accessible, interoperable and reusable." At the end of 2022, magnetic confinement fusion took the first step towards "fair" data with the European FAIR4Fusion project[2].

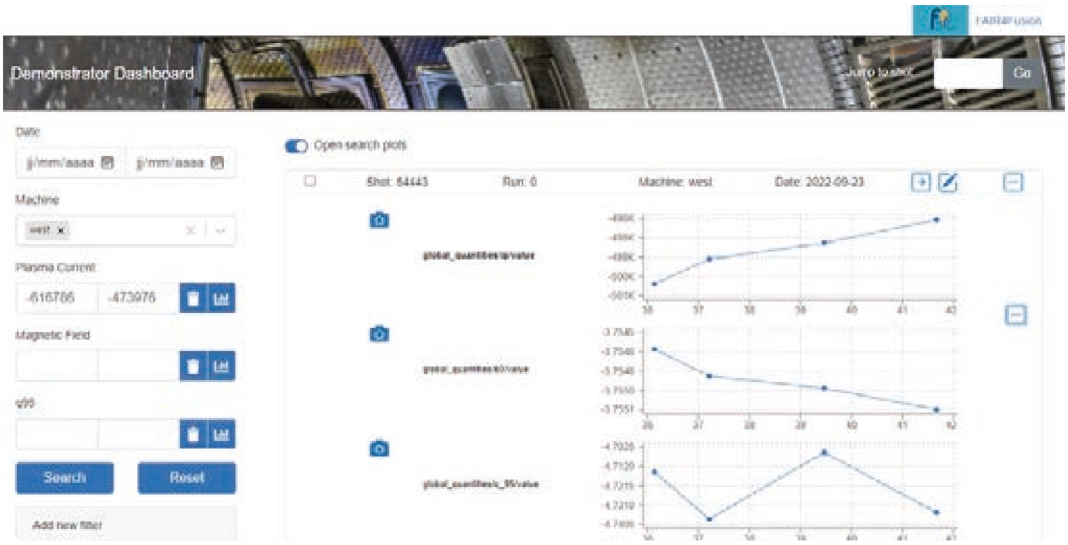
The European scientific community in the field of magnetic confinement fusion has set about making its data "FAIR" with the FAIR4Fusion project. This project, in which CEA-IRFM participated, developed recommendations and software prototypes to upgrade management of the data of the European fusion community, according to "FAIR" principles. Historically, each fusion laboratory in Europe managed the data from its experimental devices independently and without guaranteeing interoperability. There was no centralized data catalog and each laboratory had its own terminology for physical data and its access methods. It was therefore very difficult to carry out research concerning several tokamaks at the same time, which was an obstacle to exploiting the full potential of European fusion experiments.

The FAIR4Fusion project produced recommendations in four areas:

- Deploying a centralized catalog containing the metadata for all European fusion experiments. These data were made interoperable by applying the "IMAS"[3] fusion data standard developed by ITER Organization;
- Enabling access to full experiment data via the selections made on this catalog;
- Making the data reusable by improving the documentation of their origin and life cycle, by allocating a unique and permanent identifier, and by enabling their annotation and the creation of links with the publications that used them;
- Ultimately making the data accessible to all, in the spirit of the Open Science movement.

The data catalog developed by the FAIR4Fusion project has been deployed for the WEST tokamak and now enables multi-criteria searches on the physical metadata characterizing the experiments. Catalog deployment was relatively simple because WEST already used the IMAS standard for its processed data. The catalog has been filled with data from previous experiment campaigns. It is now updated systematically after each new WEST plasma discharge. The data are processed automatically in the same way for all discharges and a particular effort is made to document their origin (raw data and processing method used).

WEST was a pioneer in the use of this catalog, which was deployed by the EUROfusion consortium in 2023 as a centralized service containing the metadata from all European fusion experiments. This initiative enables much more detailed and systematic exploitation of all the results of European experiments and, in particular, makes it easier to compare European simulation models and tools with experiments.



View of the FAIR4Fusion catalog deployed on WEST for a search on discharges whose plasma current is within a given interval. The left section enables the user to enter search criteria. The right section shows the time evolution of certain physical measurements summarizing the discharge process for each plasma discharge that meets the selected criteria.

References :
[1] FAIR is an acronym for Findable Accessible Interoperable Reusable, for example, see https://fr.wikipedia.org/wiki/Fair_data and https://www.go-fair.org/fair-principles/
[2] https://www.fair4fusion.eu/
[3] F. Imbeaux, S.D. Pinches, J.B. Lister et al, Design and first applications of the ITER integrated modelling & analysis suite, Nucl. Fusion 55 (2015) 123006, doi:10.1088/0029-5515/55/12/123006, https://hal.archives-ouvertes.fr/DSM-IRFM/cea-01576460

2.7. IMAS upgraded with the integration of a WEST surface model

The 3D surface model of the WEST wall became available in March 2023 in IMAS (ITER Integrated Modelling and Analysis Suite), the first essential step to integrating synthetic imaging diagnostics into this numerical platform.

The 3D surface model of the WEST machine became available in the IMAS software suite (ITER Integrated Modelling and Analysis Suite) in March 2023. IMAS is a software infrastructure proposing a standard framework for data exchange and code interfacing for the scientific operation of ITER. WEST was the first machine to use IMAS natively to access and process data. A further step was taken with the integration of a surface model of WEST in IMAS.

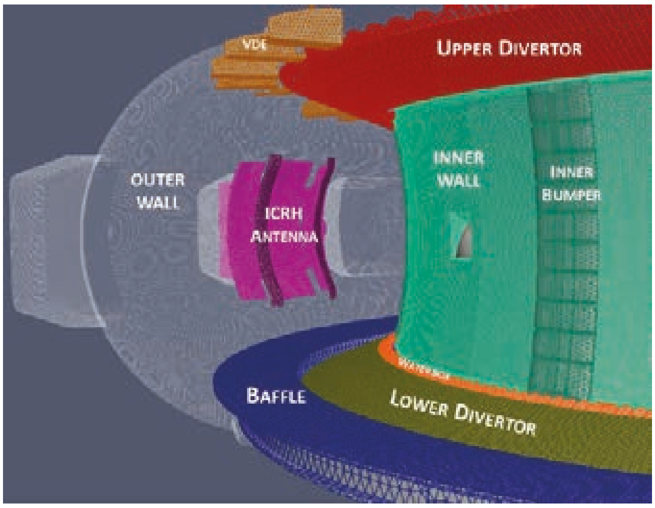
This achievement required a major preparation phase to transform the CAD (computer aided design) models from the CEA-IRFM design office into a model that could be used by the calculation codes. The transfer of CAD data to calculation software is a recurrent problem with no simple solution. From one calculation to another (neutronics, thermal, radiative), from one application to another (design, materials, virtual reality, synthetic diagnostic), different calculation models are required: solid geometry or mesh geometry, 3D volume or 3D surface, simple or detailed components, faceted, exact, configured, etc.

The model available in IMAS is a 3D surface model of all of the internal components of WEST. This 3D surface model is described by a set of surfaces that can be used to define surface properties, like the texture often used to create visual effects for movies or video games. In the case of synthetic diagnostics, these are the exact thermo-radiative properties (emission and reflection model) of the materials.

This model is generated by a two-step process: a first step to prepare the topology of the calculation model (defeaturing) and a second meshing step. The first step consists in simplifying the CAD model to make it lighter and to optimize it for meshing and calculations. This “defeaturing” step actually deletes certain characteristics or details that are unnecessary for the analysis and have no impact on the results (e.g., elimination of small diameter holes) and adapts the geometry to improve the quality and feasibility of meshing (e.g., to simplify a curve, repair a surface). The meshing of the resulting geometry is also a critical stage that can use several meshing techniques (Delaunay, Octree, etc.) to resolve the highly specific shapes of the various internal components of the machine.

After these steps, the final action is the conversion of the meshed 3D surface model of WEST to the standard IMAS format. This format is intended to be generic and follows a well-defined structure so that all users can reconstitute the geometry of WEST from the description of nodes and facets (triangles). It also includes additional information, such as the wall materials, temperatures and other geometric parameters.

This work has enabled demonstration of the feasibility of integrating the detailed geometry of a complete tokamak wall (over 40 components, 1.5 million nodes, 2.8 million triangles) in IMAS (Figure 1): a first essential step for integrating synthetic imaging diagnostics in IMAS for ITER.



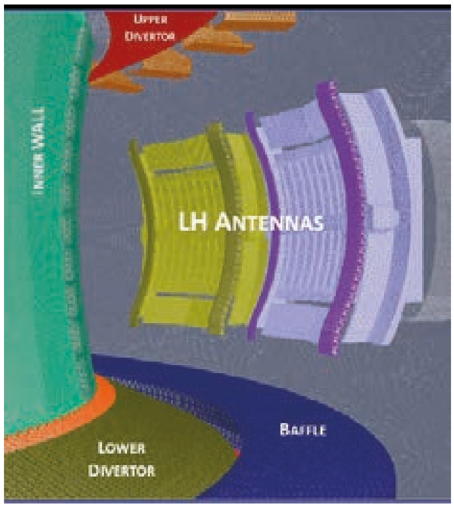


Figure 1: 3D surface representations of the 3D wall of WEST (40 components, 1.5 million nodes, 2.8 million triangles)

2.8. Testing participation in WEST experiments from La Bergerie

The REC project (Remote Experiment Center) "La Bergerie" is part of the Broader Approach project (https://www.ba-fusion.org/ba/); it aims to set up a remote participation room for fusion experiments. This room located outside the CEA center in Cadarache, in a place called "La Bergerie," was successfully tested to follow experiments on WEST during the C7 campaign in early 2023.

After three years of preparation, the main room is ready. It was tested on March 29 and April 11, 2023, with the launch and monitoring of experiments on WEST during the "High Fluence" phase of the C7 campaign (Figure 1). The participation day on April 11 was attended by several representatives from Aix-Marseille Université and ITER Organization.

During the two tests, the scientific team was divided into two:

- The Session Leader (SL), Scientific Coordinator (SC), Diagnostic Coordinator (DC), heating manager and a few diagnostics technicians were in La Bergerie
- The Engineer in Charge (EiC), Data Acquisition Processing (DAP) operator, Plasma Protection Operator (PPO) and operators were in the IRFM Command-Control room.

Communication between La Bergerie and the Command-Control room was via a Skype channel. Displays of key operating elements, such as the countdown, HD video, pressure graph, etc., on the screen walls in La Bergerie enabled the scientific team to view each impact in real time.

Figure 1: Monitoring experiments on WEST from the remote experiment room in "La Bergerie"

2.9. A new multi-machine database for studying long-duration plasma pulses

One of the challenges in the development of fusion energy is to achieve high fusion power while controlling it over long periods of time. This is one of the missions of ITER addressed by WEST (France) and a key step in the development of a fusion plant that can supply stable electrical power to the grid. Significant recent achievements in terms of duration and performance were analyzed in October 2023 using a multi-machine (tokamaks and stellarators) database set up by an international group of experts, coordinated by CEA under the auspices of the International Energy Agency (IEA) and the International Atomic Energy Agency (IAEA).

Producing long-duration plasma pulses in a magnetic fusion device requires control of the plasma for a period that largely exceeds the confinement time of the energy and particles, which is around 100 milliseconds in the WEST tokamak and 2 seconds in ITER. The aim is to approach the equilibrium timescales of plasma-wall interactions, where the physical processes, such as hydrogen saturation of the walls or surface erosion, evolve over long periods. The simultaneous improvement of fusion duration and performance requires an integrated vision of the physical and technological aspects. An international project is being implemented comprising:

- an experiment and simulation program on tokamaks and stellarators;
- a technological R&D program on specific facilities (such as actively cooled components, superconductive magnets);
- control methods to keep a self-maintained fusion plasma within a safe, stable operating domain that can be "transferred" to ITER and beyond.

With this in mind, in 2020, the International Energy Agency (IEA) and the International Atomic Energy Agency (IAEA) created a network of experts, called CICLOP, for "Coordination on International Challenges on Long-duration OPeration." The reference to Greek mythology is no coincidence, because Cyclopes were reputed to be talented craftsmen who constantly developed their expertise! [See ITER news ***https://www.ITER.org/newsline/-/3823]***. The group is headed by an expert from CEA (France) with two vice-presidents from the Max-Planck-Institut für Plasmaphysik (Germany) and the National Institute for Fusion Science (Japan).

The CICLOP group aims to promote and coordinate experiments, and to collect and share best practices for the implementation of long-duration plasma pulses.

In 2022, this activity was presented at the IAEA conference on long-pulse operation of fusion devices [November 14-16, 2022, IAEA head office, Vienna, Austria, ***https://conferences.iaea.org/event/258/***].

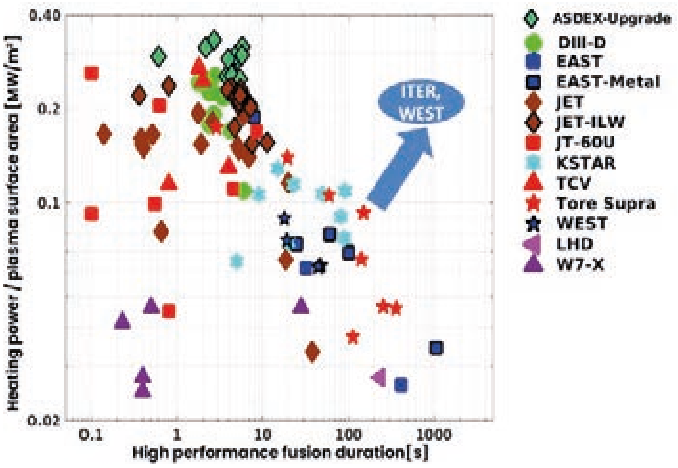


Figure 1: Heating power normalized to plasma surface area (P/S in MW/m²) as a function of high performance fusion duration. The experiments carried out with a metal wall (ASDEX-Upgrade, EAST, JET ITER Like-Wall and WEST) are shown by the symbols outlined in black. The ITER operating zone is indicated, corresponding to the operating objective of WEST at full power (phase 2).

Significant efforts have been deployed to compile and analyze a multi-machine database with data from experiments on ten tokamaks[1] and two stellarators[2]. The database is accessible via an open page of the IAEA website.

In practice, continuous high performance operation is difficult because it requires injection of high power to heat the plasma, then continuous extraction without exceeding the technological limits of the plasma-facing components. To characterize and compare the heat exhaust capacity of the various facilities worldwide, a simple indicator has been defined as the ratio of plasma heating power, P, normalized to plasma surface area, S, P/S. This ratio is shown according to the duration of the high performance fusion phase (Figure 1). The analysis highlights the challenge in terms of heat extraction for long-pulse operation, which is being addressed in France on the WEST tokamak in preparation for ITER operation.

References :
[1] ASDEX Upgrade (Germany), DIII-D (USA), EAST (China), JET (UK), JT-60U (Japan), KSTAR (Korea), TCV (Switzerland), TFTR (USA), Tore Supra and WEST (France).
[2] LHD (Japan), W7-X (Germany)

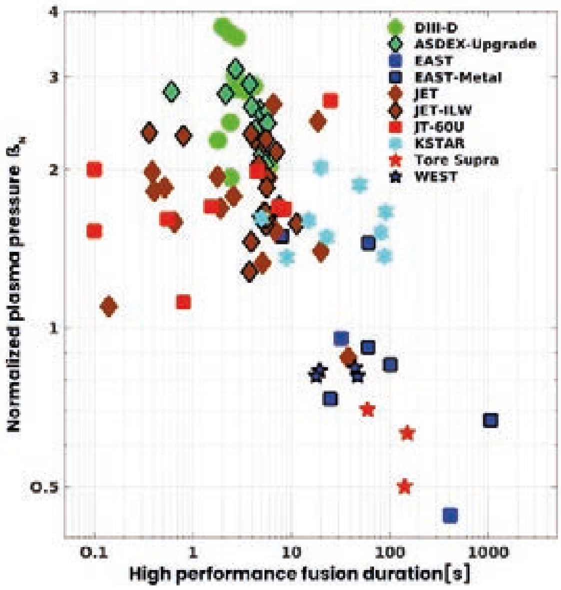


Figure 2: Normalized plasma pressure β_N, as a function of the duration of the high performance phase [s] for the CICLOP database.

Furthermore, the fusion power increase shown in figure 2 by the normalized plasma pressure, noted β_N (thermal pressure of the plasma divided by the pressure of the magnetic field multiplied by a normalization factor), has been calculated and plotted according to the duration of the high performance fusion phase. For continuous operation and a significant increase in fusion power, stable operation at high β_N values is necessary. The figure shows a significant reduction in fusion performance as duration increases. The reference scenario of ITER targets a β_N of 1.8, although fusion plant projects often aim for β_N>2.5. Note that JT-60SA, commissioned on December 1, 2023, has been tasked with exploring high β_N scenarios over long durations with the goal of optimizing the efficiency of fusion plants.

Analyses show that the ITER objectives in terms of performance duration have been met and even surpassed independently, but highlight the obstacles that must be overcome in order to achieve them together. This is the challenge facing ITER.

The CICLOP group also identified gaps in physics and engineering between current results and the operation objective of ITER and future plants, such as the need to implement highly radiative plasmas and to study how the aging of plasma-facing materials affects performance. These subjects will be a focal point of exciting and interesting discussions at IAEA's next conference on long-pulse operation in magnetic confinement fusion devices in 2024.

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Reference :
X. Litaudon, H.-S. Bosch, T. Morisaki, M. Barbarino, A. Bock, E. Belonohy, S. Brezinsek, J. Bucalossi, S. Coda, R. Daniel, A. Ekedahl, K. Hanada, C. Holcomb, J. Huang, S. Ide, M. Jakubowski, B. V. Kuteev, E. Lerche,T. Luce, P. Maget, Y. Song, J. Stober, D. Van Houtte, Y. Xi, L. Xue, S. Yoon, B. Zhang and JET contributors «Long plasma duration operation analyses with an international multi-machine (tokamaks and stellarators) database» Nucl. Fusion, 64, 01500 (2024)

3. Plasma scenario
3.1. A deuterium “airbag” to counter the harmful effects of plasma instabilities

In April 2021, a collaboration project between CEA-IRFM, the European EUROfusion consortium, ITER Organization and American partners, showed that injecting deuterium in the form of frozen shards could provide effective protection for the internal structures of a fusion reactor. These structures can suffer damage caused by the very high-energy electrons that escape from the plasma after a major instability.

To obtain nuclear fusion reactions, hydrogen isotopes (deuterium and tritium) must be heated to around 150 million degrees and the resulting plasma must be kept away from the reactor walls, i.e., “confined” very rigorously by magnetic fields in a tokamak.

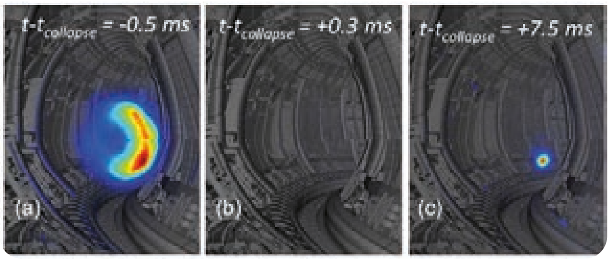


Figure 1: Infrared images showing the synchrotron radiation due to decoupled electrons. (a) Before the instability causing their dissipation. (b) 0.3 ms after the instability dissipating the decoupled electrons. (c). Re-acceleration of a small beam of decoupled electrons because of insufficient deuterium. (1)

However, instabilities occur in the plasma which is composed of very high temperature ions and electrons; these instabilities can result in energy being deposited on the walls. The plasma then disappears almost immediately during a phenomenon called disruption. In some cases, the electrons that escape from the plasma (known as “decoupled”) are accelerated at a speed close to the speed of light in the moments following the disruption, dragging other electrons along in their wake, in an avalanche-like process (Figure 1).

However, the speed at which this avalanche of decoupled electrons develops increases exponentially with the size of the tokamak. In the future international fusion experiment ITER, these electrons might deposit almost as much energy as that contained in the fusion plasma itself (around a hundred megajoules) onto an extremely small surface!

To protect against potential damage from disruptions, the preferred method consists in injecting heavy atoms (argon or neon), but this favors the appearance of high-energy electrons. The solutions implemented until now could therefore be counter-productive in larger tokamaks, such as ITER.

During a unique experiment carried out on the DIII-D tokamak operated by General Atomics in the USA, physicists observed that a large injection of deuterium could dissipate the energy of decoupled electrons very quickly. This idea, first published in 2018, was taken up, expanded upon, and confirmed by several experiments on the JET tokamak in the UK in 2019 and 2020. Scientists found that “clean” dissipation of the energy of decoupled electrons was possible, with no measurable heat load on the internal components of the reactor, if a massive injection of frozen shards of deuterium was made immediately after the disruption.

Using numerical simulation, researchers described the contributions of two physical processes to this protective effect (Figure 2). The deuterium atoms increase the instability of the beam of decoupled electrons, thus favoring the spreading of the energy deposit and driving the impurities that participate in the re-acceleration of the electrons out of the plasma.

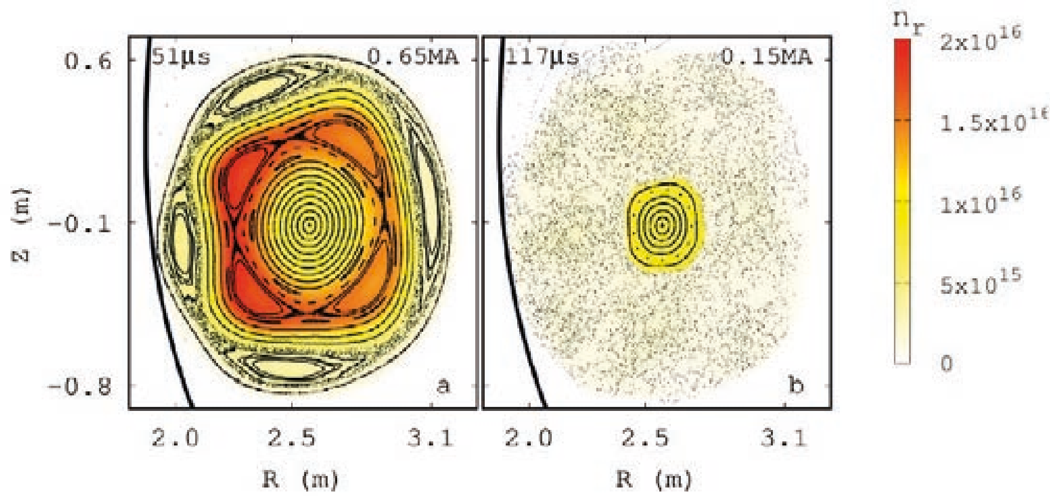


Figure 2: Magnetohydrodynamic simulation of the instability dissipating the decoupled electrons. The magnetic surfaces on which the electrons move are destroyed within a few tens of microseconds, resulting in a spreading of the electron deposit zone along the walls.(1)

Référence :
(1) Cédric Reux, Carlos Paz-Soldan, Pavel Aleynikov, Vinodh Bandaru, Ondrej Ficker, Scott Silburn, Matthias Hoelzl, Stefan Jachmich, Nicholas Eidietis, Michael Lehnen, Sundaresan Sridhar, and JET contributors, «Demonstration of safe termination of mega-ampere relativistic electron beams in tokamaks», Physical Review Letters, 126, 175001 (2021)

3.2. Record fusion energy achieved in a historic JET experiment

Scientists from the EUROfusion consortium, in which CEA-IRFM participates, recorded the production of 59 megajoules of fusion energy for several seconds in February 2022 in the Joint European Torus (JET), the world's only operational tokamak using deuterium and tritium, located in the UK. They successfully controlled the fusion plasma in an environment similar to that of ITER. This demonstration supports both the ITER project and the potential of fusion energy.

In 2021, the JET tokamak (Figure 1) produced 59 megajoules of fusion energy for 5 seconds, crushing its previous record set in 1997 (21.7 megajoules). This experiment campaign was carried out by EUROfusion scientists, including around thirty researchers, engineers and technicians from CEA.

The campaign aimed to characterize the behavior of the fusion plasma, with the new JET wall that is similar to that of ITER and using new diagnostics, in deuterium-tritium experiments and with the conditions expected in ITER to prepare the control of this future fusion demonstrator. The mission revealed a good match between the fusion power predictions and the observations for the maximum durations possible with JET.

To achieve this new record, the European tokamak underwent a major overhaul, in which CEA-IRFM was an active participant (Figure 2). In 2011, the carbon of the inner tokamak walls was replaced with beryllium and tungsten, which absorb much less tritium than carbon. The configuration of JET is thus much more similar to that of ITER, enabling important results to be obtained in preparation for future ITER experiments.

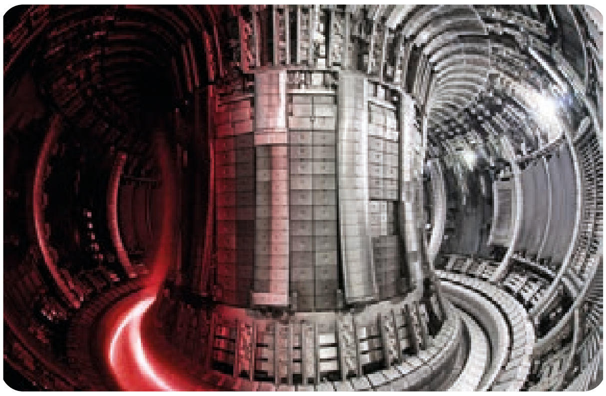


Figure 1: JET tokamak

Figure 2: During the experimental campaign

3.3. Highly radiative plasma scenario by controlled injection of impurities in WEST: approaching the operational conditions envisaged for fusion plants

Highly radiative plasma pulses, representative of ITER operating conditions and future fusion plants, were produced and maintained for several seconds during the WEST experiment campaign of the first semester 2023.

Magnetic confinement fusion plants face a critical challenge: extracting the extreme heat flux deposited locally by the plasma via the inner wall. The WEST project aims to test the plasma facing materials (tungsten) in actual operating conditions. The experiment program, previously developed on the tokamak, aimed to demonstrate plasma control over long periods, while maximizing the heat flux deposited on the materials of the divertor, the component designed to extract heat. One example scenario is shown in the figures 1a & b (1a top image, 1b: blue time plots): heat flows of several megawatts per square meter are deposited on the divertor for several tens of seconds. However, such scenarios are not representative of the operating conditions of future plants or even of ITER: the heat flux envisaged will be greater by one or two orders of magnitude, thus exceeding the extraction capacities of the components. The scenarios envisaged for future plants are based on the injection of radiating gases to dissipate the energy of the peripheral plasma electrons by photon emission. The photons distribute the heat flux over the entire wall of the tokamak, rather than just a small area of the divertor surface

The development of this type of scenario in WEST was launched during the winter 2023 campaign. Starting from a reference scenario, a controlled nitrogen injection enabled a stable dissipation scenario to be achieved for almost 15 seconds for the first time ever! Heat flux on the divertor was reduced by one order of magnitude thanks to the appearance of a highly localized radiation pattern around the magnetic X-point of the plasma configuration. This pattern, already achieved on other tokamaks over shorter times, is called the "X-point radiator."

Obtaining these conditions in WEST requires bifurcation of the plasma conditions around the X-point: the plasma condenses in a few hundred microseconds from a hot (~20 eV) relatively low density state to a very dense, cold state (~ 3 eV). Once this bifurcation has been established, the dissipation conditions can either be enhanced by increasing the amount of nitrogen, possibly leading to overall destabilization of the confinement (disruption), or weakened by reducing the amount of nitrogen. In this case, bifurcation takes place in a hot, low density state, with no X-point radiator. The more power injected into the plasma, the more room for maneuver: the system is more resilient to dissipation when the power to be dissipated is high. Based on a real time interferometry diagnostic, giving access to the electron density around the X point, it was possible to construct a control algorithm to enter these dissipation conditions and then stabilize them over a longer time. In this scenario, nitrogen injection and divertor cooling provide a significant improvement in confinement performance at the same time. The nitrogen ions diluted in the confined plasma stabilize part of the turbulence responsible for heat losses (ionic modes), leading to an increase in the temperature of the electrons and the deuterium ions of the plasma. At the same time, divertor cooling brings a significant reduction in the tungsten sources, reducing tungsten ion contamination of the central plasma by half. Radiation losses at the plasma center are reduced, thus ensuring more efficient particle heating. The rate of neutrons produced by the D-D fusion reactions increases by a factor of 4 in these scenarios.

The improvement and extension of these dissipative scenarios, through detailed understanding of the physics, are now specific objectives of the WEST operation program.

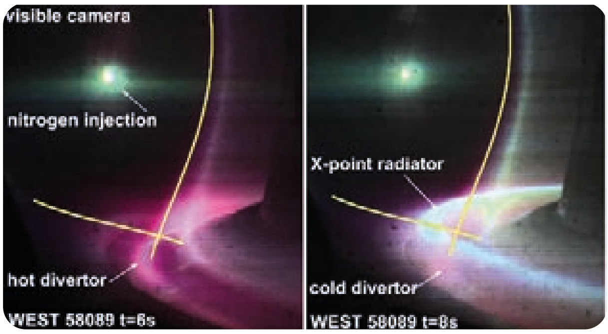


Figure 1a: Camera views

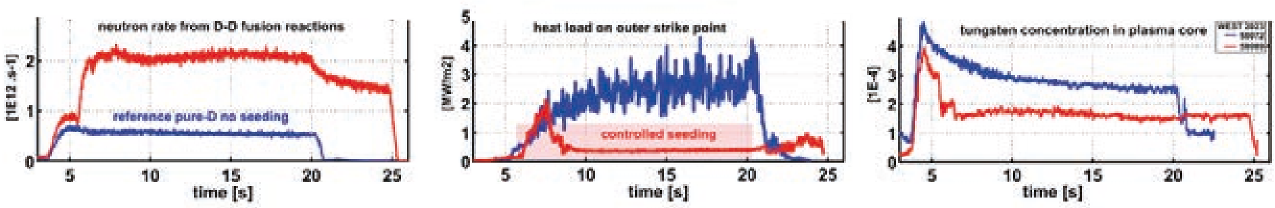


Figure 1b: Comparison obtained on WEST between the radiating scenario (cold divertor, right camera view and red lines) and the traditional scenario (hot divertor, left camera view and blue lines).

4. Plasma Wall Interactions

4.1. Testing the ITER grade divertor under tokamak conditions: the first high fluence campaign of WEST

In next step fusion devices, the plasma facing components (PFC) will be submitted to unprecedented heat and particle loads. This is in particular true for the divertor, which is the most heavily loaded PFC in a tokamak. WEST, newly equipped with an ITER grade divertor, has successfully performed a first dedicated campaign to assess the behaviour of this key component for ITER under high particle loading.

The first divertor of ITER is expected to operate until the end of the first DT campaigns, which corresponds to more than 2000 hours of plasma over more than a decade of operation. WEST was recently equipped for its second phase of operation (WEST phase 2) with a full ITER grade tungsten (W) actively cooled divertor (Figure 1). This divertor features the same technology as foreseen in ITER: divertor Plasma Facing Units (PFU) made of tungsten monoblocks (MB) assembled on a copper heat sink, actively cooled in the same conditions as foreseen for ITER.

In order to explore the impact of high particle fluence (particle flux integrated over plasma duration) on this key component ahead of ITER operation, a dedicated campaign was performed in WEST, taking advantage of its long pulse capability. The objective was twofold, investigating both the impact of:

- high fluence plasma loading on the divertor performance, and in particular its heat exhaust capability
- the evolution of the divertor under high fluence loading on plasma performance.

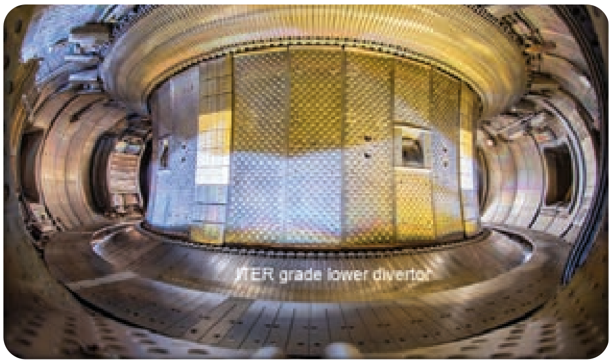


Figure 1: The WEST tokamak in the phase 2 configuration, equipped with a full tungsten actively cooled ITER grade lower divertor.

The "High Fluence" campaign was performed in spring 2023. The objective was to cumulate an ITER relevant plasma fluence on the divertor by running repetitive ~1 minute long pulses (L mode plasma regime, plasma current I_p = 400 kA, additional power from the Lower Hybrid system P_{LH} = 3.8 MW, plasma density $<n_e>$= 3.2 1019 m^{-3}, pulse length ~ 60-70 s, Figure 2).

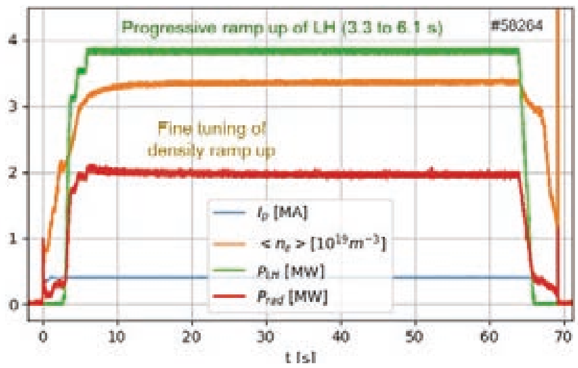


Figure 2: Main plasma parameters selected for the pulse run during the High Fluence campaign (plasma current I_p, plasma density $<n_e>$, Lower Hybrid power P_{LH}, radiated power P_{rad})

Close to 450 repetitive discharges were performed over ~1 month campaign, cumulating ~3 hours of plasma operation and 30 GJ of energy coupled to the plasma. For comparison, an equivalent plasma cumulated time would take up to year(s) of operation in other short pulse tokamaks. A total fluence of $5x10^{26}$ D/m^2 was reached at the outer strike point, which corresponds to the range foreseen for ~2 shots in the Pre-Fusion Plasma Operation (PFPO) phase of ITER [1]. The divertor fluence cumulated during the High Fluence campaign is shown in Figure 3.

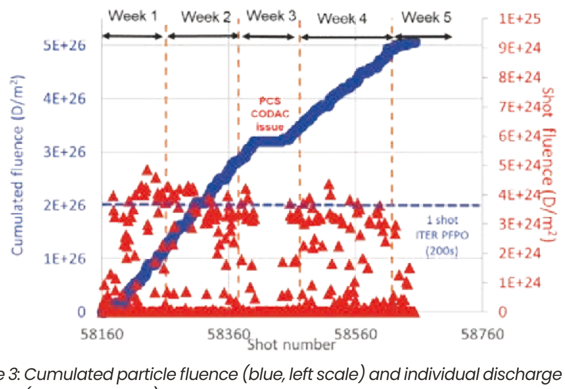


Figure 3: Cumulated particle fluence (blue, left scale) and individual discharge fluence (red, right scale) measured at the WEST divertor outer strike point during the High Fluence campaign. A technical issue with the Plasma Control System (PCS) interrupted week 3, while the slower progress in cumulated fluence by the end of the campaign is due to UFO hampering plasma operation. The level of divertor fluence of one typical 200s discharge foreseen for the ITER Pre-Fusion Operation phase (PFPO) is also indicated for reference (dashed blue line).

Références
[1] R. Pitts et al., "Physics basis for the first ITER tungsten divertor", Nuclear Materials and Energy 20 100696 (2019)

The main limitation of the campaign was due to intermittent impurity ingress due to material flakes penetrating the plasma, so called "UFO", significantly hampering operation after ~2 weeks of repetitive pulses. They were evidenced by intermittent peaks on the radiated power (see for instance time traces of discharge #58360 in after ~2 weeks of High Fluence campaign in Figure 4, to be compared with a clean discharge #58264 in Figure 2). A significant fraction of the observed UFO seems to originate from the High Field Side (HFS) of the divertor, where thick-deposited layers were seen to continuously grow.

First visual inspection of the divertor after the high fluence campaign confirmed the formation of thick deposits on the HFS (several 10's of microns), as illustrated in Figure 5. In addition, deposits were also observed in the areas shadowed by the toroidal bevel of the WEST ITER grade divertor, designed to protect leading edges from plasma overloading. This was the first time this type of deposits were observed on ITER like bevelled PFU after exposure in a tokamak environment. In contrast with the adherent HFS thick deposits, these deposits have a different thin foil like structure, and tend to delaminate easily. Although the ITER grade divertor did not show any sign of degradation in terms of heat exhaust capability during the High Fluence campaign (as expected under those plasma conditions), thin cracks were observed on the monoblock top surface in the outer strike point area. This is in contrast with results from phase 1, where cracks appeared mainly on the exposed monoblocks leading edges. Dust collected after the high fluence campaign also exhibits distinct features compared to previous campaigns, with a higher number of large flakes (~100 microns length).

This first High Fluence campaign has brought feedback on operating repetitive pulses, showing the importance of minimizing erosion / deposition on the divertor. This has set the basis for future investigation under different plasma regimes, in particular semi-detached plasmas with impurity seeding as foreseen for ITER.

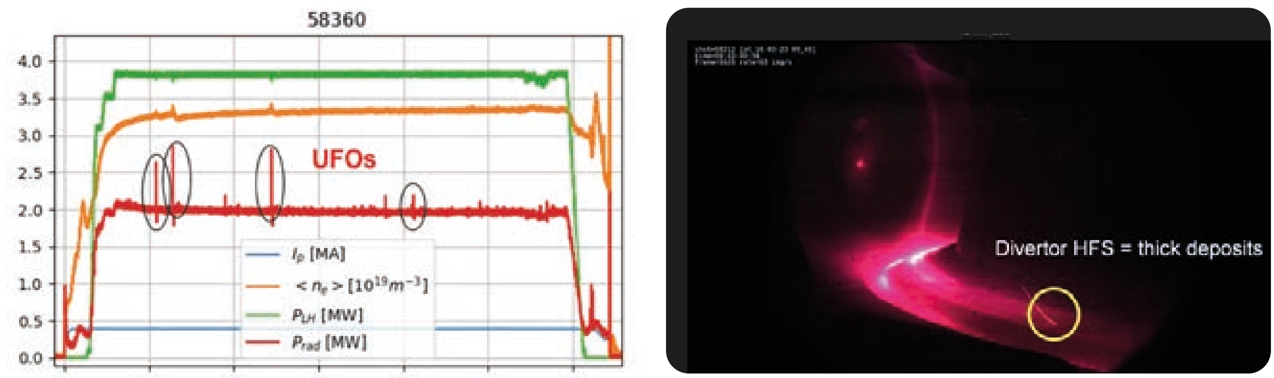


Figure 4: illustration of UFO detected during the High Fluence campaign, evidenced as peaks on the radiated power (left) or observed in the visible camera field of view, starting from the divertor High Field Side (HFS), which is an area with thick-deposited layers (right).

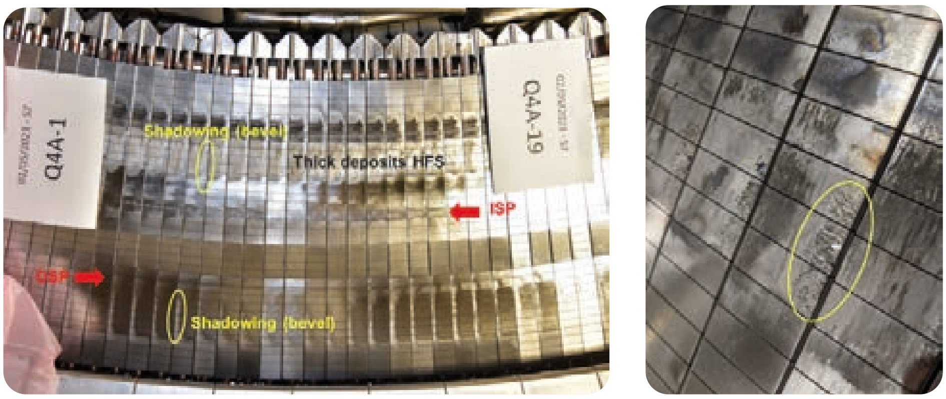


Figure 5: visual inspection of the WEST ITER grade divertor after the High Fluence campaign. Left: picture of divertor sector Q4A (PFU 1 to 19), showing the main features observed after the High Fluence Campaign. The position of the outer/inner strike point during the High Fluence campaign is indicated (OSP/ISP respectively), as well as the position of shadowed areas by bevels. Thick deposits on the High Field Side (HFS) are also indicated. Right: zoom on thin foil like deposits found in the areas shadowed by the toroidal bevel of the ITER grade PFU.

4.2. First W-melting experiments on actively cooled ITER-grade Plasma Facing Unit in WEST

In ITER or next step fusion devices, melting of the metallic wall could happen in case of excessive heat load (exceeding the technological limits) or failure in the component affecting the heat removal capability. The consequence of tungsten (W) melting in the divertor on plasma operation is therefore a high priority issue for ITER. Sustained and controlled W-melting experiment was achieved for the first time in WEST in December 2020. A leading edge was deliberately exposed to a parallel heat flux of about 100 MW.m^{-2} for up to 5 s providing a melt phase of about 2 s without noticeable impact of melting on plasma operation. The surface temperature of the melted MB was monitored by a high spatial resolution (0.1 mm/pixel) infrared (IR) camera viewing the melt zone from the top of the machine. The melting discharge was repeated three times resulting in about 6 s accumulated melting duration leading to material displacement from three similar pools. Cumulated on the overall sustained melting periods, this leads to excavation depth of about 230 µm followed by a re-solidified tungsten bump of 200 µm in the JxB direction.

Several experiments have been successfully carried out in different tokamaks (TEXTOR, AUG, JET) to investigate sustained or transient repetitive melting of tungsten by ELM on special geometries, overexposed leading edge (LE) or sloped surfaces. These experiments have been performed with bulk tungsten and no active cooling such as ITER-grade Plasma Facing Unit (PFU), therefore with a strong dependence on surface temperature and pulse history. No impact on the main plasma was observed in JET despite a strong increase of the local W source consistent with evaporation, while more detrimental effects were observed in AUG (W ejection followed by disruption). The goals of the experiment in WEST were to generate a new and unexplored melting regime with the ITER-grade PFU and achieve high surface temperature measurement using the very high spatial resolution IR camera [1] to follow the melt build-up during the experiment, a unique feature in the present-day tokamaks.

In order to perform the melting of the ITER-grade PFU, a dedicated groove (Figure 1) was machined on a monoblock (PFU7-MB28) located on the low field side of the divertor, in an area far away from the usual outer strike point position. A specific magnetic configuration was developed to locate the outer strike point in the groove of MB28. Fine-tuning of the strike point position was ensured through improved magnetic controllers, and monitored using the Very High Resolution InfraRed (VHR IR) camera of WEST.

A new challenge to overcome was to detect the start of the melting during the experiment with the IR thermography system because of the surface emissivity (ε) of the tungsten, which is low and varying with temperature. Thermal modelling was performed prior to the experiment based on experimental heat fluxes measured in WEST to pave the way up to the melting point. A series of discharges relying on a robust LH based scenario was performed, with a scan in LH power (Figure 2, left). Once getting close to the expected melting threshold (above 4 MW of LH power), the LH power was increased progressively step by step. The qualitative observation of the material displacement with the VHR IR camera, rounding the initially straight leading edge, was the first evidence of melting. Melting was observed from the VHR IR data for shots performed at 5.5 MW (Figure 2, right). In order not to modify the W melt motion in successive discharges and facilitate the subsequent modelling effort, the LH power was decreased as soon as melting was confirmed. After the session, in vessel inspection using the Articulated Inspection Arm (AIA) of WEST was performed, confirming melting of MB28. In-situ visual inspection performed afterward with the AIA confirms the melting and its extension over two millimeters in the poloidal direction.

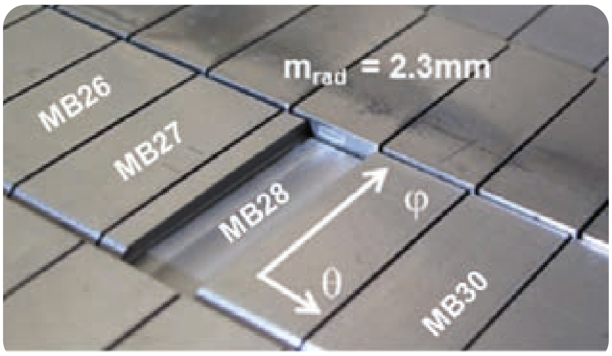


Figure 1 : Picture of the dedicated groove machined on an ITER like PFU for the melting experiment (PFU7, MB28)

During melting, radiated power and tungsten impurity content remained stable at constant input power, without noticeable impact of melting phases and no melt ejection was observed [2]. The qualitative observation of the material displacement with the VHR IR camera, rounding the initially straight leading edge, was the first evidence of melting. The heat load derived from the VHR IR data analysis has then been used as input to MEMENTO (MEtallic Melt Evolution in Next-step TOkamaks) code to model the melt dynamic in collaboration with KTH Royal Institute of Technology, Space and Plasma physics laboratory (Sweden). MEMENTO simulations show that the surface temperature crosses the W melting point shortly after 1 s and that melt displacement throughout the remaining exposure time leads to a final deformation in the 200-300 µm range which is consistent with post-mortem measurement (confocal microscopy). The observed bump of resolidified melt located next to the excavated melt crater is also consistent

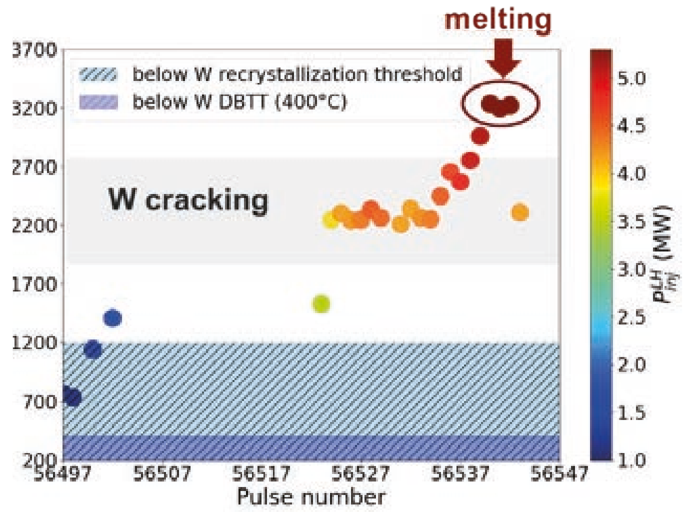
Références

1] M.Houry et al., «The very high spatial resolution infrared thermography on ITER-like tungsten monobloks in WEST tokamak,» Fusion Engineering and Design, vol. 146, pp. 1104-1107, 2019.

[2] Y.Corre et al., «Sustained W-melting experiments on actively cooled ITER-like plasma facing unit in WEST,» Physica Scripta, vol. 96, p. 124057, 2021.

[3] S.Ratynskaia et al., «Experiments and modelling on ASDEX Upgrade and WEST in support of tool development for tokamak reactor armour melting assessments,» Nuclear Materials and Energy, vol. 33, p. 101303, 2022.

with a flow in the JxB direction (inboard here) as predicted by MEMENTO, where J is the replacement current generated by the thermionic emission. The measurements collected during this experiment serve to further validate the underlying physics models of the MEMENTO code in order to improve its predictive capability for the assessment of potential surface damage by accidental melt events in ITER in case of excessive heat loads



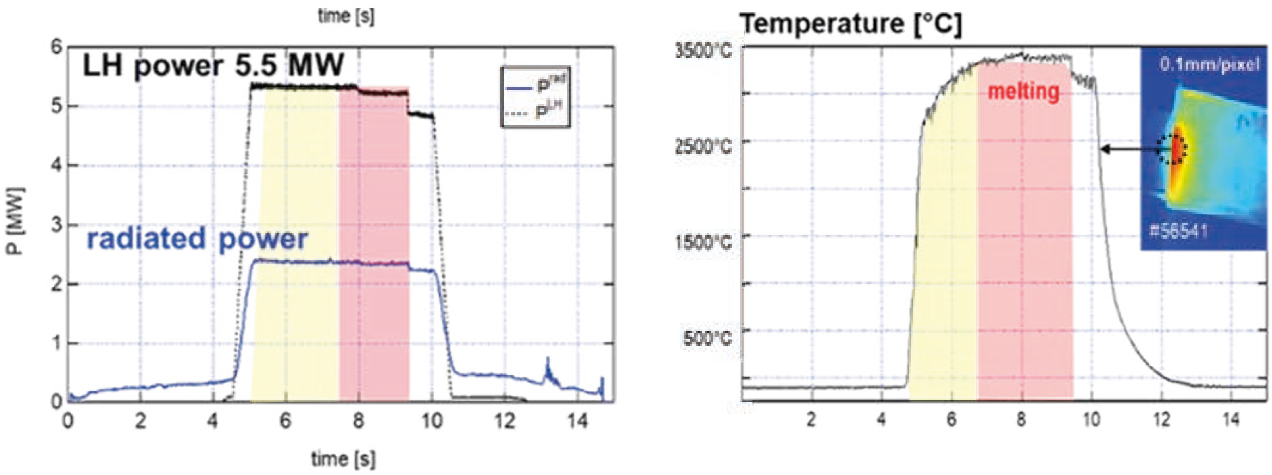


Figure 2: (Top) Summary of the experimental session to reach W-melting in controlled and safe way. (Bottom) Power and tungsten temperature time traces during melting experiment (2.5 s melting duration with steady-state plasma condition).

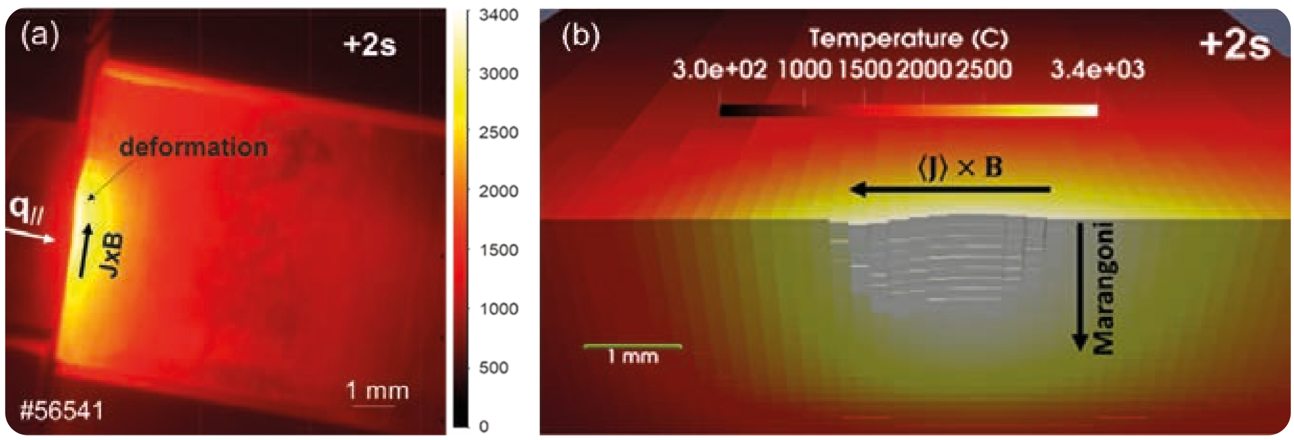


Figure 3: (a) IR images during W-melting experiment at 2s after RF heating is applied in the machine. Rounding of the poloidal LE is used to determine the time when the melting is reached. (b) MEMENTO simulation output at the same timing showing the melt displacement, crater on the strike line and hill few mm away in the JxB direction.

4.3. Hot spot predictions confirmed by experiments in the WEST tokamak

In August 2021, for the first time, WEST demonstrated the presence of hot spots located on the leading edges of the tungsten monoblock components of its divertor precisely where the numerical predictions positioned them. These observations represent an important result for ITER which will be using the same type of component for its divertor.

In tokamaks, heat is mainly transported along the magnetic field lines. The heat flux deposited on the walls intercepting the field lines can be extremely high, particularly if the wall element is perpendicular to the field lines. Plasma-facing components should be designed with surfaces as closely aligned with the field lines as possible. However, producing such elements to be cooled with pressurized water requires the use of "monoblocks," i.e., perforated cubes. The internal hole enables insertion and welding of a cooling tube. This is the technical solution retained for the experimental ITER reactor. Each component (or PFU for Plasma-Facing Unit) thus comprises a series of these monoblocks, with a space (called a "toroidal gap") between them that is almost perfectly aligned with the magnetic field lines (Figure 1). It is therefore difficult to prevent a very thin plasma beam from impacting the surface of the next monoblock. It is important to include this in the modeling, and then to validate it by experiments. The numerical studies carried out at CEA predicted risks of local tungsten melting in the ITER's PFUs[1].

To test the tungsten components of the ITER divertor in a tokamak environment, similar components (PFU) were tested in the WEST tokamak. These PFUs, made of tungsten monoblocks, were gradually installed on one of the bottom divertor sectors during the first operating phase of WEST. Efforts were made to respect the vertical alignment specifications and to align the spaces between each monoblock as accurately as possible.

For the first time in an experiment, the infrared camera images obtained during disruptions during the operation showed the presence of hot spots on the monoblock surfaces, thus confirming the predictions of numerical studies (Figure 2). These results were published in the journal Nuclear Fusion[2] and presented at the IAEA conference in May 2021.

The effects on the components were evaluated at the end of the last two experiment campaigns (early 2019 and early 2020) during maintenance operations involving the removal of certain components and thus enabling microscopic observation of the surface of the monoblocks.

These observations revealed the penetration of loaded particles in the spaces (toroidal gaps) between the monoblocks of the PFUs and their impact on the leading edges of the monoblocks, thus causing the formation of hot spots in the deposit locations of the heat flux. These hot spots, also called "optical hot spots," were observed on the components with chamfered and non-chamfered edges, by magnetic projection of the "toroidal gaps" of the monoblocks located upstream of the flux (Figure 3). These local plasma-surface interactions are indicated by a surface modification, with local melting of the tungsten and the formation of cracks on the monoblocks in the area where thermal loading is highest (strike points).

This damage (volume of melted material, extent of cracks, etc.) will be examined in detail during the post-mortem analyses, in collaboration with our international partners, to enable the experiment results to be compared with results from the models.

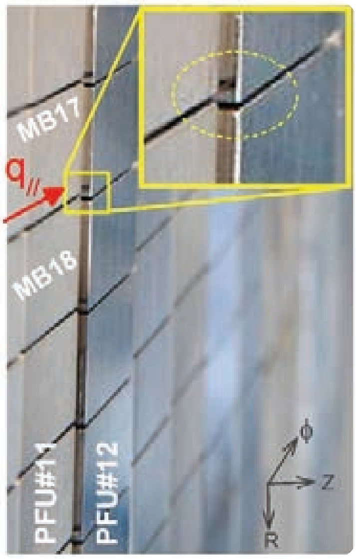


Figure 1: Tangential view of the WEST divertor showing traces of plasma impact on the PFU due to misalignment of the spaces between the monoblocks

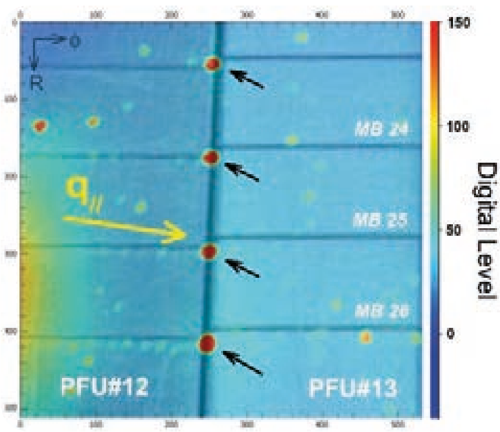


Figure 2: Image taken with an infrared camera during a disruption, clearly showing the presence of hot spots on the PFUs of the WEST divertor

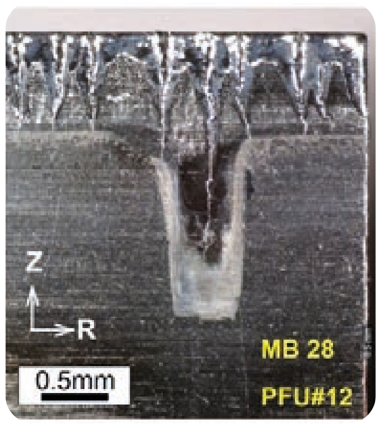


Figure 3: Local melting and cracking of the tungsten in the hot spot areas

Références
[1] J.P. Gunn et al. Nucl. Fusion 57, 046025, 2017
[2] M. Diez et al. Nucl. Fusion 60, 054001, 2020

4.4. AI to protect the walls of magnetic fusion facilities against plasma

The plasma-facing walls of magnetic confinement fusion facilities are subjected to intense heat flux that can easily exceed their extraction capacities, causing severe damage. Real-time protection is therefore a crucial challenge for the operation of such facilities. CEA-IRFM is developing an automatic hot spot detection and identification system on its WEST tokamak using Artificial Intelligence (AI) to protect the walls against plasma.

In magnetic fusion facilities able to operate with long-duration plasma pulses, the walls are covered with centimeter-thick components, made up of an assembly of refractory material (generally tungsten) on a structure material with high thermal conductivity (usually a copper alloy) cooled by pressurized water. To maximize the energy production of fusion plasmas, these components are used near their technological limits. The protection systems are therefore designed to prevent these components from being permanently damaged, while enabling exploration of the broadest possible operational domain of the facility.

CEA-IRFM has set up a system on its WEST tokamak combining infrared thermal instrumentation, modeling of heat transfer and photon emission, signal processing and understanding of the physics of the plasma-wall interaction to ensure optimized and controlled protection of the components in this metal environment.

Wall protection system (Figure 1) requires a rapid response while managing huge amounts of data. Action must be taken in real time, within a few tens of milliseconds, based on the analysis of the infrared cameras monitoring the wall (12 cameras are currently installed in WEST). Depending on this analysis, a mitigation strategy might be initiated, such as the reduction or interruption of the plasma heating power, distancing of the plasma from the wall or the injection of impurities into the plasma to cool it down.

CEA-IRFM is developing and installing advanced methods combining the expertise and knowledge acquired with artificial intelligence techniques, i.e., the Faster R-CNN model, based on convolutional neural networks [1]. Based on annotation databases created by experts in wall protection and automatic learning techniques, these methods detect and identify events in the thermal scenes of infrared films, enabling classification according to risk level (Figure 2). At the same time, multi-modal approaches are being studied to enable exploitation of all the data produced (infrared films, time data from other sensors); there are also methods injecting a priori knowledge physics into models to improve the characterization of certain types of heat deposits, for example, using a Max-Tree type decomposition to classify anomalies [2].

2023 saw the first recorded demonstration of the capacities of these techniques, from the last WEST campaign, displaying excellent detection and recognition of events, that are made available to experts just a few minutes after each experiment. This technique was tested in real time during the experiment campaign at the end of 2023 and opens new and promising perspectives for the future operation of ITER.

Figure 1: View of the plasma-facing components in WEST

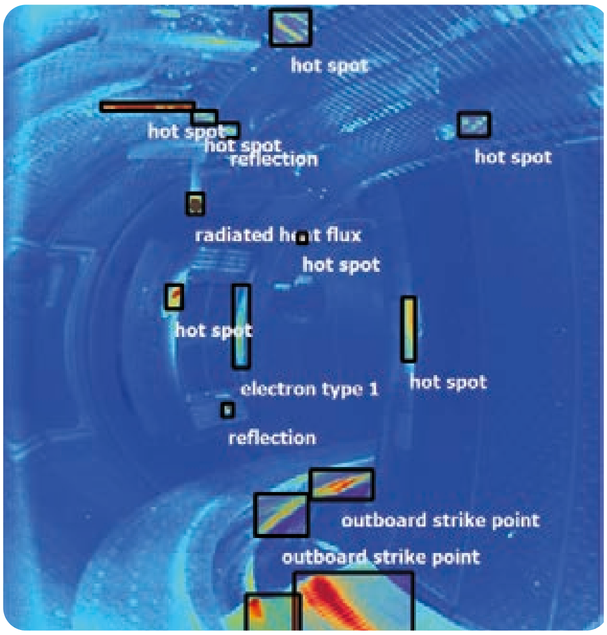


Figure 2: Image from the infrared surveillance video in the WEST tokamak enclosure. The events standing out from the background noise are detected (square outline) and identified (label) with Faster R-CNN, and monitored over time using the SORT algorithm (Simple Online and Real-time Tracking).

References :
[1] E. GRELIER et al., Fusion Eng. Des. 192 (2023) 113636
[2] V. GORSE et al., Rev. Of Scientific Instruments, Vol. 94, Issue 8 (2023)

4.5. Tungsten nanoparticles produced by tokamak plasmas

The interactions between the plasma and the walls of fusion machines can generate dust from the materials exposed to the plasma. These dust particles can affect tokamak operation if they enter the plasma, contaminating it to the point of extinction. Furthermore, in next-generation machines, like ITER, this dust is tritiated and activated, constituting a radiological inventory that must be monitored. Studies conducted using today's machines are required to understand how the dust forms, the morphology of the particles, and their characterization to learn lessons for future fusion facilities. This is a current project for the WEST tokamak, as part of a collaboration between CEA, CNRS (French National Centre for Scientific Research) and IRSN (Institute for Radiation Protection and Nuclear Safety).

In fusion devices, plasma interactions with the walls and maintenance phases can produce micro-debris or dust in the vacuum vessel (dust is defined as particle matter with a size of up to ~1 mm).

In ITER, the dust produced is mostly tungsten, a heavy element used for the plasma-facing components. During experiments, the dust particles can be mobilized and transported to the plasma core, becoming a source of pollution that deteriorates plasma performance and potentially causes its extinction. These radiotoxic particles also represent a safety risk in the event of accident discharge into the environment. To control these risks, the dust inventory is strictly limited in ITER. This objective demands better understanding of the dust creation phenomena in conditions as similar as possible to those of ITER.

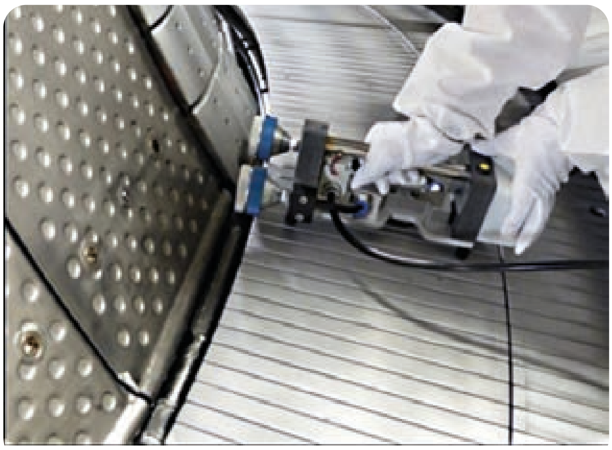


Figure 1: The "Duster box," developed by IRSN to collect dust in the tokamak environment, in action on the first tungsten-coated components of the WEST divertor

CEA-IRFM, CNRS (PIIM laboratory for Physics of Ion and Molecular Interactions, and the CP2M multidisciplinary center of electronic microscopy and microanalysis of Aix-Marseille Université), and IRSN have been collaborating for several years to study the physical and chemical behavior of the dust produced in tokamaks, and develop dust measurement and collection systems for the vacuum vessel.

Although much of this research is done in the laboratory using numerical codes, the WEST tokamak, operated by CEA-IRFM at the Cadarache center, is an essential tool for the study of this dust in ITER-like conditions. The plasma-facing components of WEST are also made of tungsten, either solid or with a coating of a few microns. This is the case of the divertor, the component with the highest thermal load in the machine, which also includes components using the same tungsten monoblock technology as ITER.

Since 2017, samples have been taken from the divertor in the vacuum vessel after each experiment campaign (and before any human intervention) to help improve our understanding of the dust formation processes. Two dust collection techniques are used: the "duster box," a device designed and supplied as part of the collaboration with IRSN, which enables the suspension and collection of the dust particles present[1] (Figure 1), and a suction technique, through a collaboration with PIIM. The dust is then characterized by electron microscopy at CP2M to identify the morphology of the particles and their chemical composition, and to determine their possible origins. An analysis is also carried out to estimate the amount of dust created.

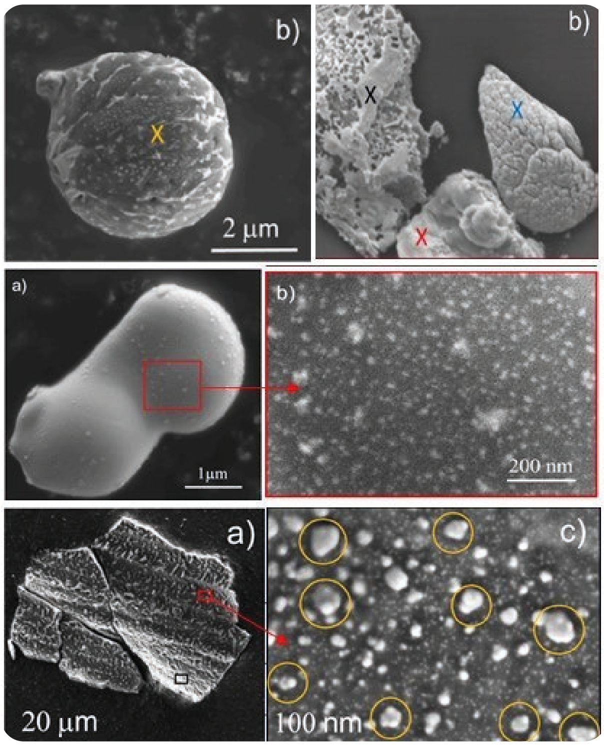


Figure 2: Examples of dust particles collected from the WEST divertor. Separate and agglomerated nanoparticles are visible on the surface of this dust.

The recently published [2] results of these studies indicate the presence of two distinct populations of dust after the 2020 experiment campaign.

The first is dominated by particles whose size varies from a few microns to a few tens of microns, containing mostly tungsten, but also boron, carbon and oxygen. This dust comes from the delamination of tungsten coatings, the emission of droplets of melted materials after intense thermal loading, or the formation of dust particles due to material erosion by the plasma. Nanocavities were also found on the surface of tungsten dust particles collected after a WEST operating phase with helium plasmas. This can be attributed to the tungsten capturing nano-bubbles of helium, a phenomenon identified in the laboratory and observed in tokamak conditions.

The second population of dust particles identified by this study is more surprising: it comprises tungsten nanoparticles (Figure 2). These were mostly found on the surface of particles measuring a few µm and may be due either to the condensation of a supersaturated vapor above melted tungsten, or to the ion-neutral deposits that develop in low temperature plasma regions until solid particles appear.

This characterization work was carried out in 2023 on the dust collected during the first experiment campaigns of WEST (phase I), which was equipped with a divertor with tungsten coatings, and is being continued with WEST, now equipped with a divertor with full tungsten ITER-technology components (Figure 3).

The impact of these changes will be investigated in terms of dust production and behavior.

Figure 3: New ITER-technology solid tungsten components from the WEST divertor

Références :
[1] S. Peillon et al., Nuclear Materials and Energy 24 (2020) 100781
[2] C. Arnas et al., Nuclear Materials and Energy 36 (2023) 101471

4.6. World first: real-time irradiation and characterization of the size and shape of helium nano-bubbles in tungsten

In magnetic fusion devices, the plasma-facing components are subjected to high-energy particle flux made up of ions and atoms of hydrogen (fuel) and helium (product of the fusion reactions). The helium implants and aggregates in the form of nano-bubbles that then alter the properties of the material in question, such as tungsten. To improve our understanding of how these helium bubbles form in the tungsten, a real-time implantation and characterization experiment by grazing incidence small angle X-ray scattering was carried out at the end of 2023 at the ESRF (European Synchrotron Radiation Facility) synchrotron in Grenoble, France. A world first!

In magnetic fusion devices like WEST and ITER, the plasma-facing components must withstand continuous and transient thermal loads. The divertor, the most exposed component, must withstand thermal loads of 10 MW/m² (continuous) and 20 MW/m² (transient). These components are also subjected to particle flux comprising ions and atoms of hydrogen (fusion fuels deuterium-D or tritium-T), helium and neutrons (produced by the D-T fusion), and impurities.

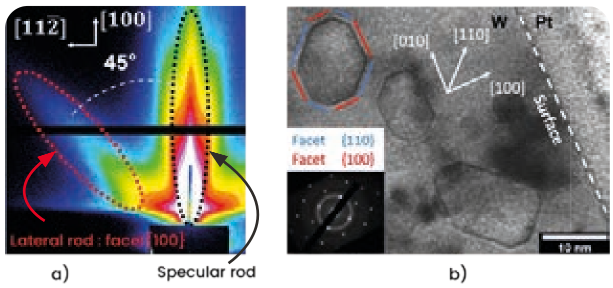
Although low in energy (<100 eV), the helium around the divertor implants, aggregates and forms nano-bubbles of helium under the surface of the component. The bubbles modify the microstructure, both in the volume and on the surface of the component, altering the properties of the plasma-facing material (physical properties and hydrogen-retention properties).

In a fusion device, the thermal loading cycles, the associated thermal gradients, the surface chemistry of the components (oxygen, boron, hydrogen, etc.) and their microstructure (polycrystalline, pre-existing crystalline defects) make it difficult to understand the growth and evolution mechanisms of the bubbles. A laboratory approach is therefore required to control and evaluate all these parameters individually. A collaboration between CEA-IRFM, CINaM (interdisciplinary Nanoscience Center of Marseille) and PIIM (Physics of Ion and Molecular Interactions laboratory) was initiated for this purpose in 2021.

It aims to highlight the formation and evolution mechanisms of the helium bubbles to improve the prediction of their presence and effects on the macroscopic properties. Tungsten samples were implanted in helium and characterized in real time by grazing incidence small angle X-ray scattering[1] (GISAXS) at the ESRF (European Synchrotron Radiation Facility) in Grenoble. By measuring the scattering of X-rays and applying appropriate data processing, this technique enables definition of the shape and size of bubbles over several square millimeters of the sample, unlike Transmission Electron Microscopy (TEM) which only provides highly localized information (~0.1 µm²).

Monocrystalline tungsten is studied to dissociate the potential effects of the microstructure on the mechanisms. The samples are prepared at CINaM to verify the surface state (minimization of roughness and surface contamination).

The helium is implanted at ESRF on the BM32 line using an ion gun. Ion implantation energy is thus controlled and set at 400 eV or 2 keV, i.e., below and above the threshold for the hydrogen to create gaps in the tungsten (~1,000 eV). Finally, sample temperature is kept constant at different values between ambient temperature and 1,000 °C (surface temperature of the ITER divertor in normal conditions). This experiment is the first of its kind in the world. After implantation, the samples are annealed under ultra-high vacuum up to 1550 °C to reproduce the heating undergone during plasma transients. The effects of thermal annealing are also monitored by GISAXS in-situ.



Characterization of a monocrystalline tungsten sample, implanted and annealed. By GISAXS (a), the lateral rod (red dots) is caused by the scattering of X-rays on the surface facet {110} of the bubble. By MET (b), helium bubbles are visible on the near surface. The bubbles are faceted by facets {110} (blue) and {100} (red).

Figure (a) is an example of a GISAXS photo obtained after annealing. The vertical rod, called the "specular rod," is related to the roughness of the sample. The presence of a lateral rod indicates that the bubble shape is not spherical but faceted. The crystallographic orientation of the facet, and therefore the 3D shape of the bubble, can be deducted from the angle between the rod and the vertical [2]. This shows that the shape of faceted bubbles is made up of facets {100} and {110}. Post-mortem transmission electron microscopy analyses match the results obtained by GISAXS (Figure b). They also show that all the bubbles have the same facets, indicating that they are close to thermodynamic equilibrium. Finally, the width of the lateral rods on the GISAXS photos is related to the average size of the facets that generated them. By analyzing the data during implantation, the kinetics and growth mechanisms of the helium bubbles in the tungsten can be deduced by GISAXS.

These results are valuable for defining the impact of the shape and growth kinetics of bubbles on the loss of tungsten properties.

References :
[1] G. Renaud, R. Lazzari, and F. Leroy, Surf. Sci. Rep. 64, 255 (2009).
[2] L. Corso et al., Nucl. Mater. Energy 37, 101533 (2023).

IV FUSION TECHNOLOGY

1. Introduction

The scope of CEA-IRFM's technology activities is vast. They are implemented on WEST, ITER and the fusion plant project being developed in Europe (DEMO).

The first part will illustrate the R&D work on superconducting magnets, mostly to support implementation of the various components of ITER's magnets.

Development activities on RF heating systems will be described by a progress review on the installation of the new Electron Cyclotron Resonance Heating (ECRH) system on WEST, an example of R&D efforts for ITER's Ion Cyclotron Resonance Heating (ICRH) system, and the tests carried out on an innovative antenna concept that may be used on the future fusion plant.

IRFM is expert in the design, manufacturing and qualification of actively cooled plasma-facing components, among other things. Notable achievements during the period of reference include the implementation of the HADES high-heat flux test station, and the production and installation of the WEST divertor equipped with its ITER-grade components. IRFM also applies its expertise to the development of actively cooled plasma-facing components for the JT-60SA machine.

The area of diagnostics design and development is illustrated by the implementation of three new instrumentations on WEST: LIBS tests in the WEST vessel using the poly-articulated robot AIA, the integration of three new X-ray measurement diagnostics developed in collaboration with the Princeton Plasma Physics Laboratory, and the deployment of temperature measurements by Fiber Bragg Grating in WEST divertor components. Finally, this overview of diagnostics activities concludes with a review of one of IRFM's key projects, the development of the ITER equatorial Vis/IR diagnostic, carried out by a consortium headed by CEA-IRFM, in association with one French industrial group and two Spanish laboratories.

Controlling the fuel cycle in a future fusion plant is a major challenge. IRFM's activities concern modeling of the retention/diffusion of hydrogen in relevant materials associated with tritium experiments carried out in collaboration with CEA's Institut Joliot, the development of measurement techniques using LIBS and support for ITER's TBM (Test Blanket Module) team, integration studies for these systems in ITER's ports and the specific areas where CEA's nuclear expertise is valuable.

Finally, the last two key events illustrate the engineering activities of IRFM.

Hall Tore WEST Tokamak

2. Magnets
2.1. MIFI: multi-expertise support for ITER magnets

With the MIFI (Magnet Infrastructure Facilities for ITER) project, CEA-IRFM, thanks to its proximity to ITER, its operational capabilities and its pool of expertise in superconducting magnets, is supporting ITER teams in the acceptance, qualification and testing of ITER magnet components.

In 2014, the ITER project initiated a large number of in-kind supply contracts with Domestic Agencies (DA) representing the countries involved in the ITER agreement. The superconducting magnets at the heart of the machine and all their related subsystems (instrumentation, joints, insulators, etc.) were among the first to be defined, qualified and manufactured. These magnets use technologies considered strategic at the start of the project and their manufacture was deliberately shared among all the project partners.

At the time, ITER Organization was starting to receive the first deliveries of magnet equipment at Cadarache, either to be stored/archived, or to be qualified and have its properties validated, or to begin the huge work of integration and assembly preparation. The ITER site was not able to host the laboratories and expertise required for these activities. ITER therefore asked CEA-IRFM, located nearby, for help due to its operational capacities and expertise in superconducting magnets.

CEA-IRFM has know-how in the four areas identified as critical: Instrumentation and low voltage / High voltage and insulation / Mechanics and assembly / Superconducting materials and cryogenics. Laboratories, storage space and, most importantly, experts were called upon to interface directly with the managers of ITER about the project's requirements to ensure maximal reactivity: the MIFI (Magnet Infrastructure Facilities for ITER) project was born.

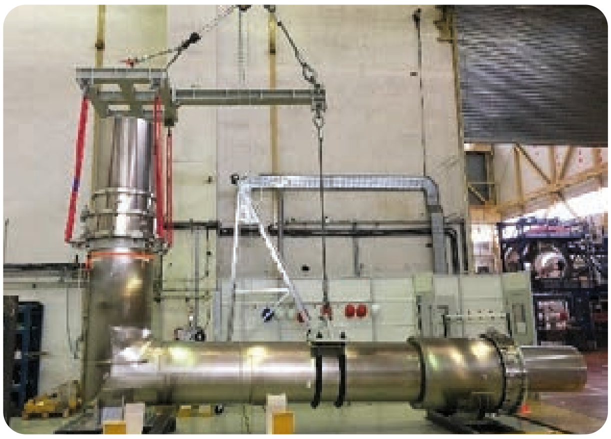


Figure 1: PF4-CFT d'ITER

CEA-IRFM provided substantial support for the acceptance testing, qualification and testing of ITER magnet components, in order to prepare the operating instructions for the assembly phase. There were a number of key events during this period, attesting to the excellent collaboration between CEA-IRFM and ITER teams, and only a few emblematic examples are presented here:

- Instrumentation/low-voltage laboratory:
  Qualification of the vacuum brazing procedure for the thermal blocks of the magnet temperature sensors.
- High-voltage/insulation laboratory:
  Implementation of Paschen tests on the magnet instrumentation wires.
- Mechanics/assembly laboratory:
  Development of the welding procedure for the superconducting joints.
- Superconductiing materials and cryogenics laboratory:
  Liquid nitrogen thermal cycling (77K) of instrumentation chains.

Furthermore, MIFI was a key player in ITER's acceptance of the PF4-CFT (subcomponent of a current carrier for coil PF4), the first major magnet component, manufactured by the Chinese partner and accepted by ITER in Cadarache (8x4x2m and ~10t) in autumn 2019.

CEA-IRFM and ITER teams worked very well together, as did various external participants (ITER subcontractors, domestic agencies) on this exceptional technical platform.

2.2. Superconducting joints for ITER: from concept to on-site production

CEA-IRFM has always maintained strong expertise in the design, development and qualification of superconducting junctions, which are used to transmit huge currents (up to 70kA) from one superconducting cable to another with very low resistance (of the order of 1nΩ). This know-how is used to develop and test superconducting joints for ITER, and to train future operators who will assemble these connections on ITER.

CEA-IRFM expertise in the design, development and qualification of superconducting joints was at the origin of the most common design in modern fusion machines (EAST/JT60-SA/ITER): Twinbox connection (Figure 1).

In 2019, ITER came up against a problem with a joint developed by an industrial company for the central solenoid (CS). The initial concept involved a highly complex manufacturing procedure and the resistance was found to be unacceptable by the electrical tests, making it potentially impossible to operate the magnet. In collaboration with ITER, CEA-IRFM revised the joint design entirely, developing a much simpler process using an interface of indium wires and Rutherford cable quadrants. This concept eliminated the risks related to the brazing operation while also facilitating the skills transfer to the future industrial partners responsible for ITER assembly.

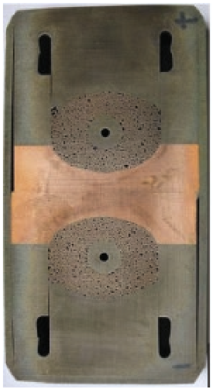


Figure 1: Twinbox Connexion

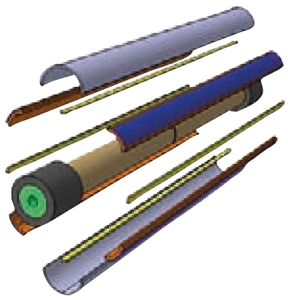


Figure 2: New joint design for ITER central solenoid.

Following an ITER request, CEA-IRFM and its experts develop a test station to conduct resistance measurements at nominal current (70kA) and cryogenic temperature (4.2K). This station would then be used to train and qualify the personnel responsible for assembling the real joints of the magnet at the ITER site. The station, called SELFIE (for SELf FIEld joint test facility, Figure 2), was developed from 2020 to 2022. The first tests were carried out in January 2022, demonstrating its capacity for the rapid, reliable testing of superconducting joints. The station uses a superconducting transformer, a high technology component that can create a high current (70kA) from a low-current power supply (250A).

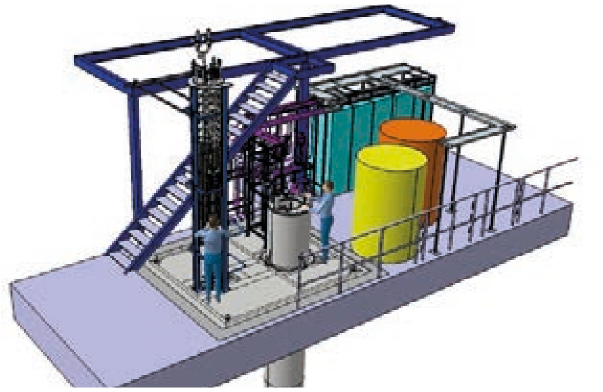


Figure 3: CEA view of the SELFIE station

At the same time, CEA-IRFM was designing and producing an environment mockup (CJEM for CS Joints Environment Mockup) to simulate the restrictive environment in which the joints were to be produced. The mockup also enabled testing and qualification of the positioning and handling tools to be used for the fragile superconducting terminals of the cables. The prototype samples tested in SELFIE were therefore produced in conditions very similar to the real conditions of the ITER site.

During 2023, the team in charge of the SELFIE operation qualified three pairs of assembly technicians using the conclusive test of joint prototype samples to ensure the best possible training and qualification, the only way to guarantee good performance of the real joints. This activity came to an extremely successful conclusion with the participation of CEA-IRFM experts in the production of the first superconducting joint of the central solenoid at the ITER site (Figure 3).

Figure 4: Production of the first superconducting joint of the central solenoid at the ITER site

2.3. The challenge of DEMO magnets: multi-physics dimensioning and prototype conductor

CEA-IRFM is involved in the design of the superconducting magnets for the DEMO project within the EUROfusion consortium of laboratories, in particular in multi-physics analyses to validate the design of the proposed magnet architectures. The latest design iteration has resulted in an CEA-IRFM proposal for the DEMO magnets, based on the knowledge acquired during the ITER development and qualification phases, and the launch of prototype production for testing at the SULTAN test station (Switzerland).

Within the EUROfusion framework, the DEMO design activities consist in proposing a detailed design for each of the main magnet systems, starting from a preliminary machine version provided by the EUROfusion central team: TF (toroidal magnet), CS (central solenoid) and PF (poloidal magnets). Multi-physics analyses must also be provided to enable validation of the design of the proposed magnet architectures.

Working directly with all the European laboratories involved in this working group (ENEA, KIT, SPC, INFLPR, etc.), CEA-IRFM defined a dimensioning sequence covering all the important aspects of superconducting magnets: design and superconducting performance, electromagnetism, mechanics, thermohydraulics and protection. These activities also include the prototyping and testing of superconducting cables.

The final design iteration enabled a CEA proposal to be highlighted for the DEMO magnets, based on the knowledge acquired during the development and qualification phases of ITER. The proposals was based on conductors made of niobium-tin (Nb_3Sn) for the toroidal magnet system and the central solenoid, and poloidal magnets made from niobium-titanium (NbTi). The coils were analyzed thermally and mechanically (Figure 1) to anticipate operating conditions and sufficient margins.

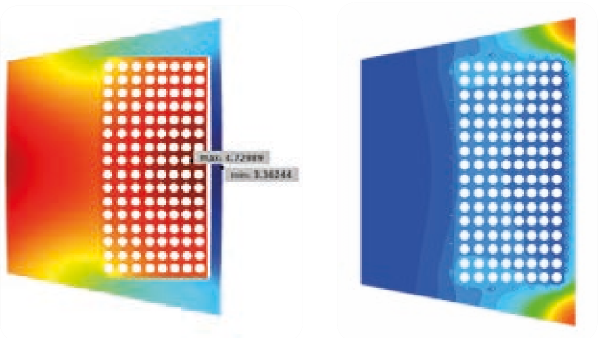


Figure 1: Thermomechanical analysis of a DEMO the toroidal magnet system

For this project, the European laboratories were asked to develop superconducting cable prototypes for the cables proposed in their magnet designs. These cables were then tested in the SULTAN test station (Switzerland) to measure and check their performance.

CEA-IRFM thus launched the manufacture of a sample conductor representing its design, i.e., a superconducting cable with 1750 strands in Nb_3Sn, able to transport up to 90kA under a target magnetic field of 12T. After completing the design, cabling tests were carried out, resulting in confirmation of the geometric parameters required (compaction rate, internal and external dimensions). Two 4m prototype lengths were produced to assemble a qualification sample for testing in the SULTAN test station in 2024. It has been more than 20 years since CEA-IRFM last proposed a fusion superconducting conductor. This new conductor is expected to enable validation of the technological choices of recent years. Analysis of its cryogenic performance will also be a valuable source of scientific and technical data for the French and European community.

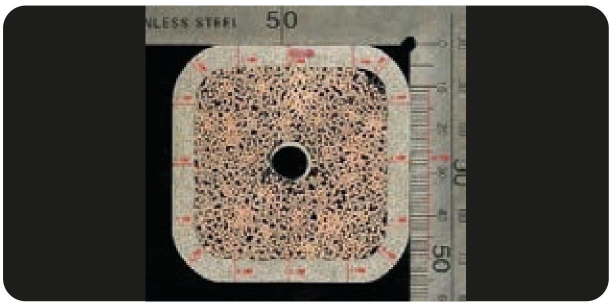


Figure 2: TF conductor prototype for DEMO

More recently, CEA-IRFM also proposed to work on a conductor concept for the high-field layers of the CS, using innovative HTS-type materials (High Temperature Superconductors). The proposed design is based on Rebco ribbons, able of transporting high currents under a 20T magnetic field. CEA-IRFM aims to develop a prototype for this type of conductor in coming years.

3. RF Heating

3.1. An innovative radiofrequency antenna concept validated at CEA-IRFM for fusion plasma heating

A TWA (Traveling Wave Array) antenna mockup was tested at high power using the TITAN test bench at CEA-IRFM in May 2021. This mockup represents the first step towards producing a TWA antenna for the ions-heating in a tokamak. The mockup was tested up to 2MW for 3s (maximum generator power) and for longer periods (500kW for 60s). The measurements obtained are conform to modelling results: the broadband antenna (10 MHz) requires few adjustments and the voltage inside is lower than that of conventional antennas for similar power levels, representing an essential advantage for the ions-heating antennas of future fusion machines.

WEST and most tokamaks use antennas in the frequency range between 30 and 60 MHz to heat the plasma ions: these are known as ICRH (Ion Cyclotron Resonance Heating antennas). Before 2021, all the antennas used were described as "resonant," which meant they were small enough to be inserted into the machines via a port.

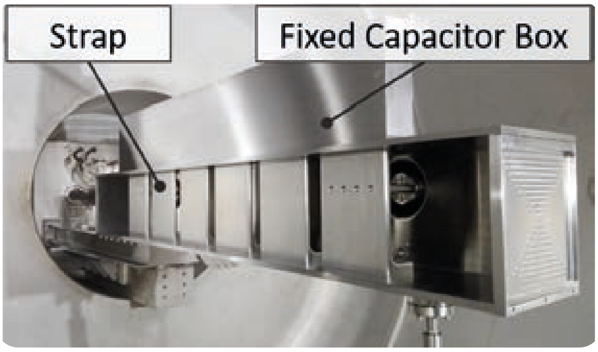


Figure 1: TWA antenna mockup (Travelling Wave Array)

Like a guitar or violin, these antennas had to be tuned, i.e., set to operate at a specific frequency. Future fusion devices, like ITER or DEMO, also aim to use this type of antenna. However, because the space available is limited, they require even more power for an available area that is barely larger. The power density required is therefore higher, reaching the limits of current know-how...

To resolve this problem, several different antenna concepts were proposed.

Traveling wave antennas can reduce power density at the cost of a larger plasma-facing area (not necessarily requiring the use of a port). Having more radiating elements than a conventional antenna, they are also expected to provide better coupling between the radiofrequency power and the plasma. Finally, these antennas are broadband, i.e., able to operate over a wide range of frequencies without adjustment and without mobile elements inside the machine (an important requirement in terms of reliability). In spite of these advantages, and because they require more space, this type of antenna has never been used for ICRH heating.

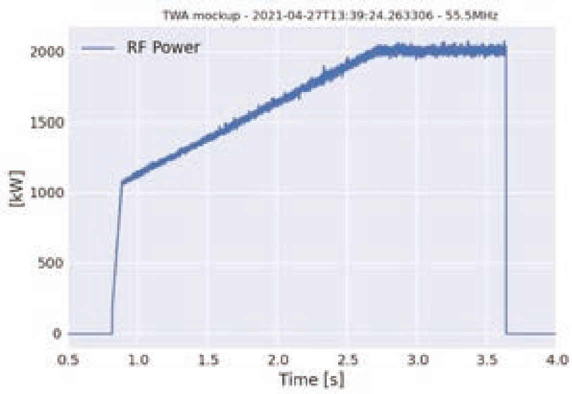


Figure 2: TWA antenna tests up to 2 MW

A simplified mockup (flat and not cooled) of a TWA (Traveling Wave Array) antenna was tested at high power on the TITAN test bench in March 2021 (Figure 1).

This mockup was designed within an EUROfusion project coordinated by the LPP laboratory of the Royal Military Academy (Belgium) and manufactured at ASIPP in China. The mockup, assembled at CEA-IRFM and placed in the TITAN vacuum vessel (Figure 2), was tested up to 2MW for 3s (maximum generator power, Figure 3) and for longer periods (500kW for 60s). The measurements (RF, infrared, etc.) were conform to model results: the broadband antenna (10 MHz) requires no adjustments during use and the voltage inside is lower than that of conventional antennas for similar levels of power. These advantages are essential for the ion heating antennas of future fusion machines. These positive results, conform to simulations, enable consideration of the next step: why not have one or more TWA antennas in WEST?

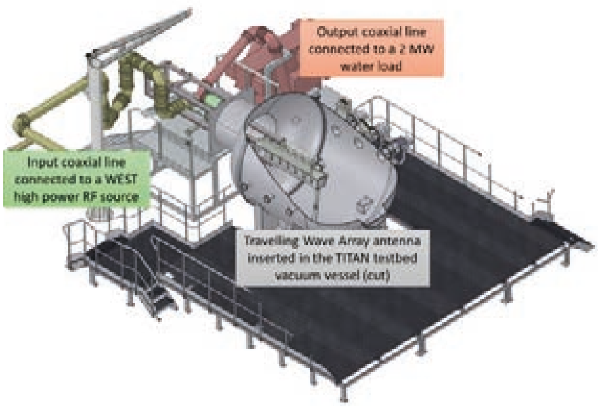


Figure 3: Diagram of the TWA antenna in the TITAN facility

3.2. Towards the future continuous ECRH system for WEST

The ECRH (Electron Cyclotron Resonant Heating) project was initiated in 2021 with the goal of equipping the WEST tokamak with 3 MW of heating power at the electronic cyclotronic frequency of 105 GHz. This new heating system will provide the plasma with more radiofrequency (RF) power to enable transition to H mode, and will also help to reduce the concentration of heavy impurities (W tungsten) and radiative losses at the center of the discharge. The ECRH system will thus expand the WEST's operational domain.

Until 2011, CEA-IRFM had an ECRH system at a frequency of 118 GHz that was not suitable for WEST with an emitter comprising two gyrotrons and a cooled antenna equipped with three mobile mirrors. To enable a new ECRH system to be launched as quickly as possible and to optimize costs, the scope of ECRH project consists in:

- Repairing the existing ECRH antenna with its operating limits (1 MW/10 s per channel);
- Replacing the RF sources (by three new 105 Ghz gyrotrons) and their auxiliary systems;
- Reusing as many transmission line components as possible;
- Restoring the ECRH generator to enable the conditioning of the loaded gyrotrons and their use on plasma.

Since the project began, the four major supply contracts (gyrotrons, superconducting magnets, water-cooled HF load, line components) have been or are currently being finalized. The first of the series of three gyrotrons manufactured by THALES was installed in November 2023 on its superconducting magnet at KIT (Karlsruhe Institute of Technology, partner of WEST). This laboratory is equipped with a test bench (Figure 1).

Figure 1: ECRH components being tested at KIT

In December 2023, the first TH1511 gyrotron was tested up to 1 MW for 3 ms, demonstrating that RF performance is attained during short pulses. Long-pulse tests are being continued throughout 2024.

The generator is currently being transformed with the installation of the components. Major modifications concerning the present platform are required to ensure personnel safety (renovation of the floor, repair of platforms and gateways). The cooling circuits and command-control systems are also being modified to control the three new gyrotrons. Some components, like the tanks and supports to hold the gyrotrons, have been in place on the generator since 2023 (Figure 2).

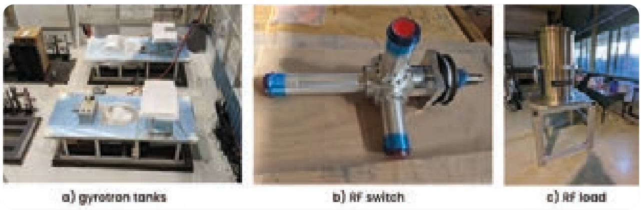


Figure 2: Components already installed (a) or to be installed on the ECRH platform (b) (c)

As soon as the generator is in its final configuration, it will be connected to the antenna (installed in the tokamak) via corrugated circular waveguide-type transmission lines. The ECRH antenna has been modified and its reliability improved mainly to ensure movement of the plasma-facing mirrors that enable the RF beam to be oriented in different positions. A third channel with position control has also been set up. After implementation of these modifications, the antenna was tested in a vacuum vessel at 70 °C. Following the validation of antenna operation for WEST operating conditions, it was installed in October 2023 in the tokamak (Figure 3).

The first plasma pulses with the ECRH system are scheduled for December 2024 with a first gyrotron. After installation and testing of the other two gyrotrons, the ECRH power available for experiments on WEST will reach 3 MW in 2025.

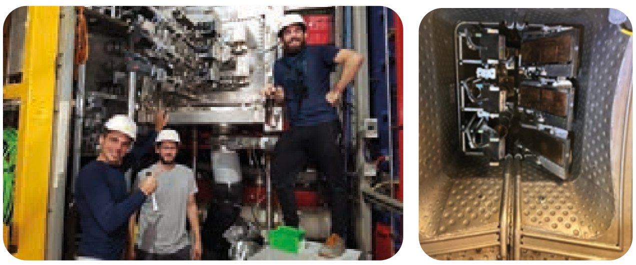


Figure 3: ECRH antenna assembled in the tokamak with its three channelss

3.3. R&D support to develop the ICRH system for ITER

CEA-IRFM is involved in R&D and engineering activities to support the ITER team responsible for developing ITER's ICRH system. Notable results were achieved in 2022 and 2023 with finalization of the Faraday screen design, sliding contact testing, experimental qualification of an innovative manufacturing method for a key element of the antenna, and the design of a specific test-bench for qualifying the antenna's HF components.

The Faraday screen design project was completed. ITER's ICRH antenna is composed of four identical RF modules, built into a complete port plug. Each module has its own Faraday screen, providing protection against plasma radiation (Figure 1). The screen is an actively cooled component close to the plasma. It is subjected to severe constraints of multiple origins (thermal load from the plasma, high frequency, electromagnetism, thermal expansion, etc.). The Faraday screen design is complex and required several iterations with calculation phases and manufacturing processes to take into account the multi-material assembly constraints. Hydraulic, thermal, electromechanical and structural studies were thus conducted throughout the project, resulting in the design validated by ITER Organization in 2023.

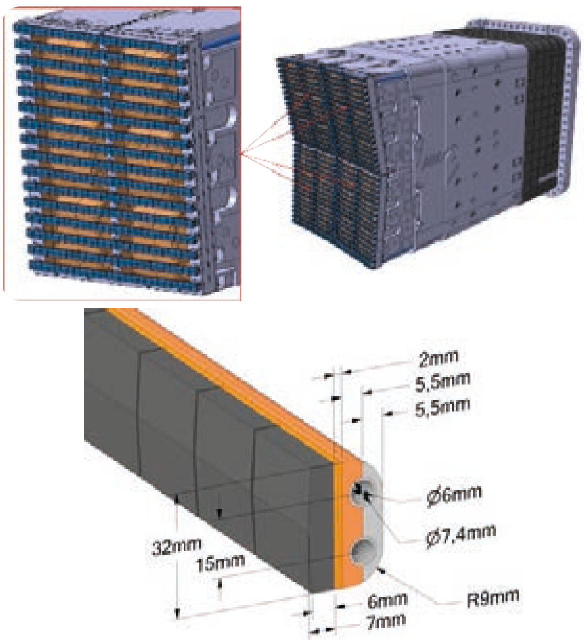


Figure 1: (up) Complete antenna and front screen, (down) section of a bar from the screen, composed of a Be/CuCrZr/stainless steel assembly.

ITER's ICRH antenna has "sliding" radiofrequency contacts. These enable decoupling of certain parts of the antenna to facilitate the assembly phase while also enabling thermal expansion of the transmission line during the conditioning phases at 250 °C or during the operating phases. CEA-IRFM performed HF tests on two types of radiofrequency contacts provided by ITER Organization. These tests were performed at 63 MHz, in a vacuum at 90 °C, with the goal of achieving a 2.5 kA current for periods of 3,600 seconds. The tests were carried out on a specific RF resonator set up at CEA-IRFM (Figure 2).

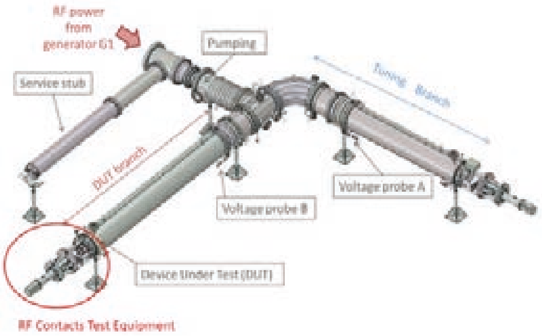


Figure 2: Resonator for RF contact testing.

The results enabled discarding the concepts that failed to achieve the required performance levels in favor of a concept developed by ITER Organization which was successfully tested up to 2kA for 10 s and 1.5kA for 3600 s (Figure 3).

Figure 3: RF contacts after testing at 1.5 KA for 3,600 s.

ITER Organization entrusted CEA-IRFM with the design and production of a resonator to test the sealed windows of the antenna. The difficulty lies in reproducing the severe environmental conditions (vacuum on one side, pressure on the other, with operating temperatures of 90 °C and baking temperatures of 240 °C). Furthermore, there are two types of windows (front and rear) with different dimensions and structures (Figure 4). The windows must be tested for maximal voltage or maximal current conditions, which means the resonator must be configurable for four different tests (Figure 5).

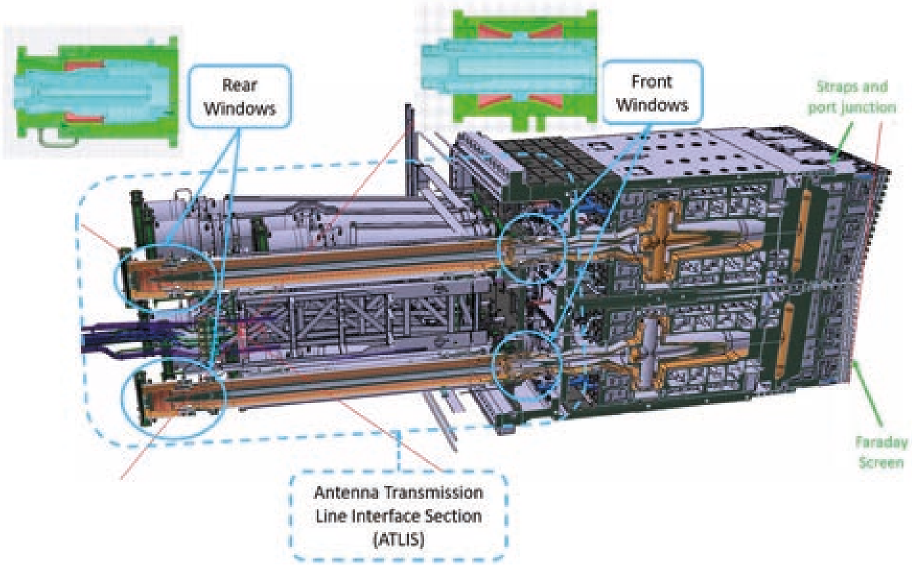


Figure 4: ICRH antenna with its rear and front windows

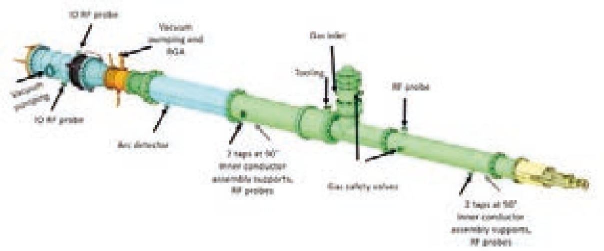


Figure 5: Resonator in front window configuration and maximal voltage test

Resonator design was finalized at the end of 2023 and its manufacture began immediately. The resonator is due to be delivered at the end of 2024 or early 2025.

A final step was to test a strap prototype produced by 3D printing. Three-dimensional printing is justified by the complexity of the component, and particularly its cooling system. In particular, a network of capillary-type cooling channels passes through the strap, and this network cannot be produced by traditional manufacturing methods (Figure 6).

Figure 6: Set of cooling channels in the strap.

The first test involved leak testing under He with thermal cycles to check the integrity of the component and its capacity to withstand the water pressure.

The second test aimed to validate the prototype's mechanical resistance in nominal thermal load conditions during operation (1,000 cycles at 0.35 MW/m²). On the ITER's antennas, the straps are located behind the Faraday screen of actively cooled horizontal bars. The CEA-IRFM's high-heat flux testing platform (HADES) based on an electron gun-type source was used.

To simulate the presence of the Faraday screen bars, the HADES electron gun was configured to produce these bands of power flow (Figure 7).

The tests revealed that the strap conserves its power extraction capacity during thermal cycling in normal operating conditions.

The rest of the activities will concern the tests of the ICRH antenna windows with a bench designed for this purpose, before potential tests on a prototype of a quarter antenna in a vacuum case at CEA-IRFM.

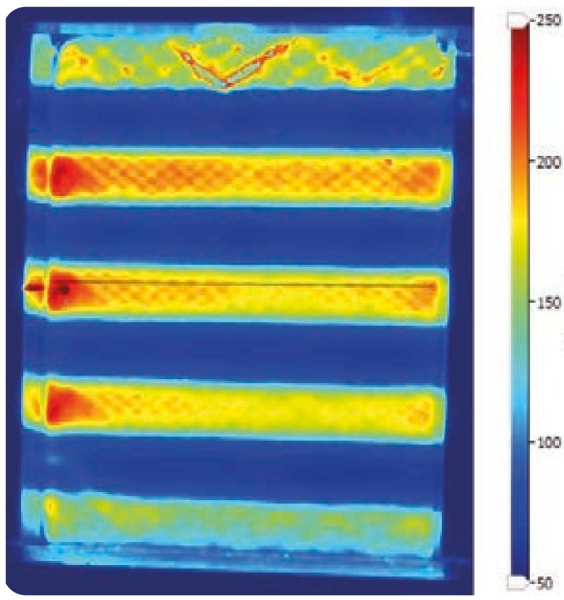


Figure 7: Example of temperature distribution on the front face of the strap during the HADES test

4. Plasma Facing Components
4.1. First beams on HADES

On October 13th, 2020, CEA-IRFM commissioned a high-flux testing platform HADES (High heAt LoaD tESt facility), based on an electron gun-type source. This facility, the only one of its kind in France, is designed for the development and characterization of materials, components and instrumentation in extreme environments.

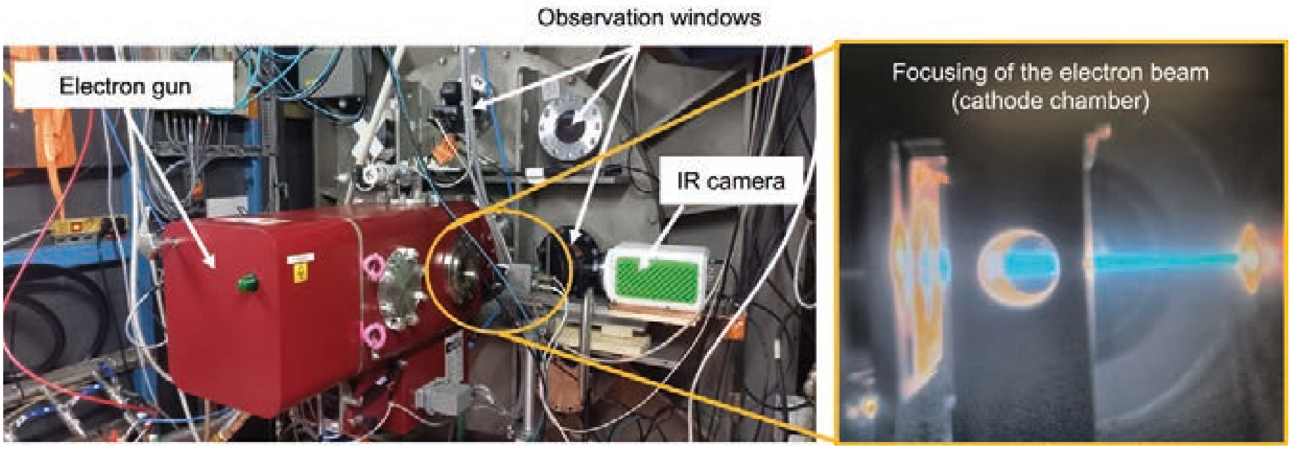


Figure 1: HADES high-power electron gun (commissioned Oct. 2020)

The HADES project was launched mid-2017 with the transfer of the main subsystems from the FE200 high-heat flux testing platform in Creusot to Cadarache, followed by reassembly and commissioning at CEA-IRFM. This project came after AREVA's decision at the end of 2016 to terminate the operation of this facility jointly funded by CEA and FRAMATOME in the 1990s.

After this first phase, the subsystems, including a vacuum vessel > 8m^3, a pressurized hydraulic circuit, dedicated diagnostics and a command-control system, had to be completed with the supply of a high-power electron gun (Figure 1) (150 kW max.), to be interfaced with the existing subsystems and the facilities available at CEA-IRFM.

The final phase involved ensuring the conformity of the different subsystems in the gun's environment (HV power supply cabinets, gun cooling, gun vacuum pump system, cabling, command-control cabinet for the gun, etc.), connection of the actual gun (Figure 1), connection to the main power supplies, and connection to the decarbonated water circuit. It will end with specific tests on an actively cooled plasma-facing component.

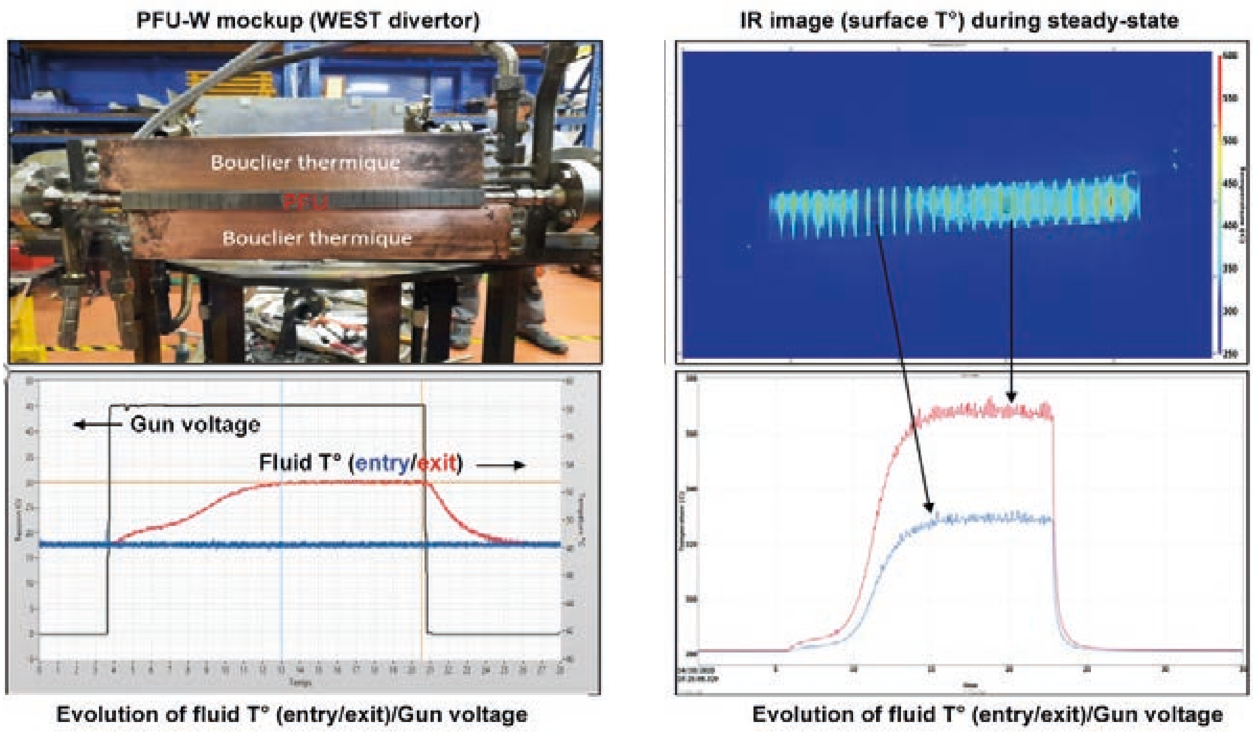
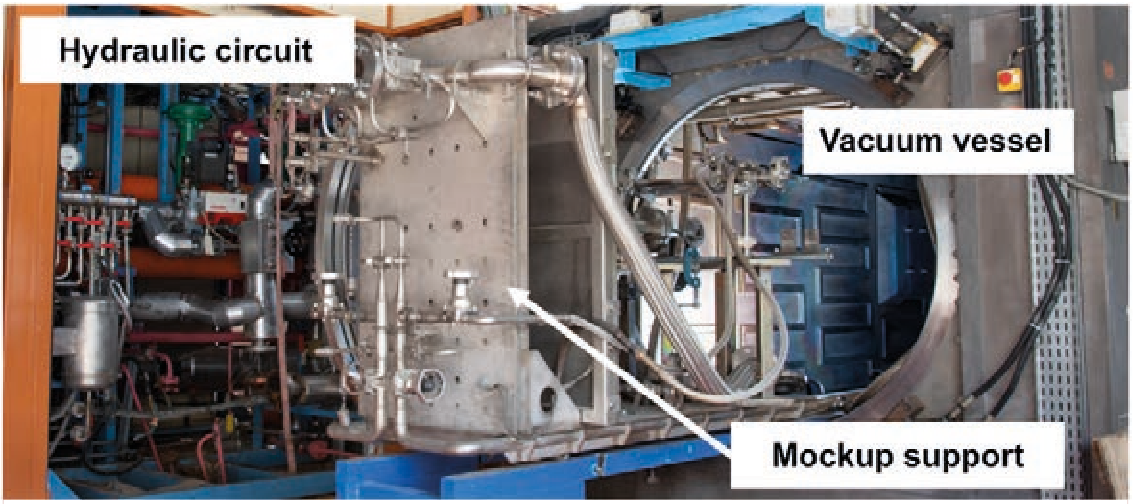


Figure 2: Functional tests of the HADES electron gun on a PFU-W component (WEST divertor phase 2)

The functional acceptance tests were carried out on a PFU-W component (PFU – Plasma Facing Unit) representative of WEST's new tungsten divertor (Figure 2), with a controlled progressive power ramp-up to max. power (150 kW incident). The gun was formally accepted on October 13, 2020, and the HADES platform was commissioned on October 15, 2020, under the supervision of APAVE and CEA's radiation protection department.

This permitted the launch of the first acceptance tests of the ITER-like PFUs of the WEST divertor (phase 2) and qualification of the prototype mockups of the DEMO divertor (EUROfusion).

With HADES and its technical properties (Figure 4), the CEA-IRFM has a facility that is unique in France for the development and qualification of materials, components and instrumentation in extreme environments (vacuum, high temperature, high heat flux).



Electron gun :

- Power : 150kW (45kV)
- Heat flux : 1-50 MW/m^2 (fatigue tests) to 1GJ/m^2 (disruption simulation)
- Advanced sweeping system : 10 kHz +/- 25°
- 0,1s to 6,000s

Vessel : 8m^2

Cooling circuit (water) :

- Pressure up to 35 bar
- 50-230°C
- Output : 6kg/s

Diagnostics :

- Infrared camera 3-5µm
- 2 pyrometers (300-3,300K), 1 pyroreflectometer, 1 two-color pyrometer
- Real-time acquisition & command-control system
- Calorimetry; flowmeters; pressure gages; thermocouples etc.

Figure 3: Hydraulic circuit & vacuum vessel with support for the mockups being tested. Main specifications of the HADES facility

4.2. WEST adopts the monoblock

The actively cooled tungsten divertor (W), the main plasma-facing component of the WEST tokamak based on technology similar to that of the ITER divertor, was installed in September 2021 in the WEST's vacuum vessel. This major milestone for WEST, achieved through partnerships and teamwork, thus paves the way to long-pulse operation (up to 1,000 s), with high levels of heat flux (> 10 MW/m^2), notably enabling the testing of these ITER-grade plasma-facing components in a tokamak environment.

WEST now has an actively cooled full tungsten divertor (Figure 1). The supply (manufacture and associated inspection testing) of this complex (multi-material) component is very similar to the ITER divertor.

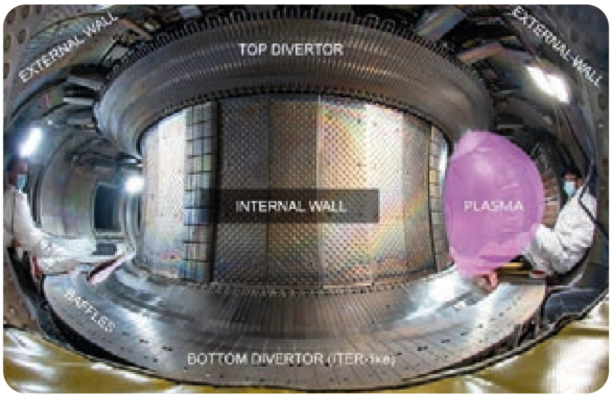


Figure 1: Inside view of the vacuum vessel of the WEST tokamak with its various plasma-facing components.

In particular, it is based on the monoblock concept, with specific assembly techniques developed in Europe over several years. It thus uses W as the plasma-facing material and a reinforced copper alloy (CuCrZr) for the cooling structure. Assembly requires an inter-layer material (soft copper) able to accommodate the differential expansions (Figure 2). The full WEST divertor W, described as an "ITER-grade divertor," has 456 actively cooled components (called Plasma Facing Units – PFUs), each equipped with 35 tungsten monoblocks assembled over 30 ° sectors (38 PFUs per sector – Figures 3a, 3b).

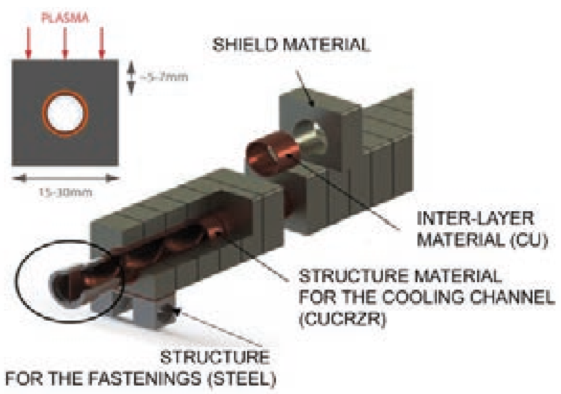


Figure 2: Actively cooled monoblock plasma-facing component

Most of this industrial production (i.e., 456 PFUs, representing a total of almost 16,000 blocks of tungsten), was manufactured by AT&M (China), with support from the Chinese laboratory ASIPP, in the context of the CEA-China collaboration (SIFFER, SIno French Fusion Energy centeR). A few components, supplied by the European domestic agency Fusion for Energy (F4E) were also provided by ITER Organization, in the context of the collaboration agreement between the international organization and CEA. This huge production project, upstream of ITER, has already produced a lot of relevant information on the mass production of such components (optimization, standardization, tolerances, statistics, etc.). This information will be key to optimizing the manufacture of the ITER divertor, while minimizing the inherent risks, and operating the ITER equipped with a cooled, "full tungsten"* divertor.

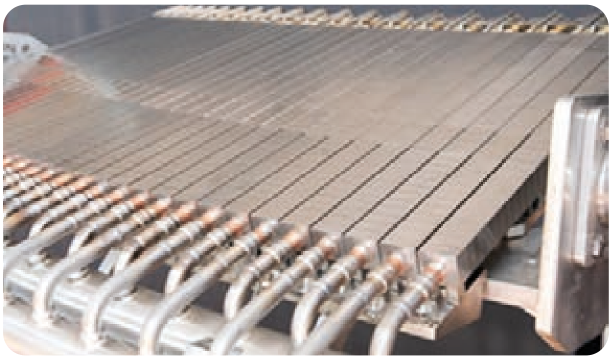


Figure 3a: Plasma Facing Units (PFUs) assembled on a 30° sector

The WEST tokamak, equipped with an ITER-grade actively cooled tungsten divertor and instrumentation (Langmuir probes, Fiber Bragg Grating optical fiber temperature sensors and thermocouples), can operate with long-duration plasma pulses (up to 1,000 s) in thermal load conditions similar to those of ITER. It can provide a lot of information on the integration of this type of component in a tokamak, the resistance to intense heat flux (10-20 MW/m^2) in steady-state combined with transient flux, the compatibility of the tungsten with plasma performance, and the synergy effects of the various stresses typically found in a tokamak environment.

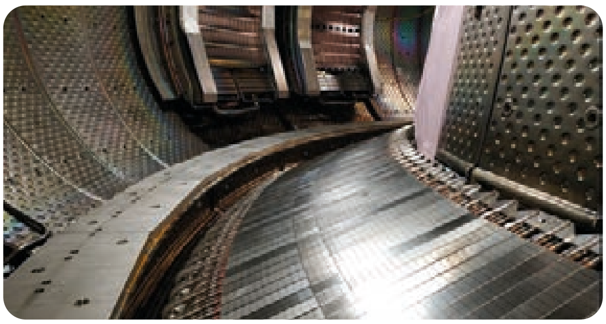


Figure 3b: The sector after installation in WEST

**Note that the complete ITER divertor will have over 2,000 actively cooled components, i.e., almost 300,000 blocks of tungsten assembled on CuCrZr tubes. The industrial production of the ITER divertor is thus a challenge in itself because of the specification requirements (performance, tolerances, etc.).*

4.3. CEA-IRFM, coordinator of R&D activities for the future divertor of JT-60SA

The JT-60SA tokamak was designed, funded and built by Europe and Japan at the Naka site (Japan), with a substantial contribution from France, as part of the Broader Approach signed at the same time as the agreements for the ITER site in Cadarache. In 2021, CEA-IRFM took over responsibility for the R&D activities concerning the first wall components under the most stress: the elements of the divertor for this superconducting tokamak, the largest ever built before ITER.

CEA was a major player in the construction of the JT-60SA tokamak in Japan, providing half of the toroidal superconducting magnet system, the cryogenic unit and electrical power supplies. JT-60SA is designed to support ITER operation and to help research into the design of a future fusion plant. The plasma-facing components of the first wall are among the key elements of tokamaks. JT-60SA will start its experiments with a divertor made of a graphite assembly on a molybdenum alloy (TZM) before moving on to a tungsten technology, in a configuration similar to that of ITER. The European agency F4E (Fusion For Energy) is responsible for the design, manufacture and assembly of the elements of this divertor (Figure 1).

In 2021, CEA-IRFM took over responsibility for the R&D activities for the elements of the JT-60SA divertor for the EUROFusion consortium. In 2022 and 2023, the qualification tests (tests at high heat flux on CEA-IRFM's HADES platform) on representative mockups were unsuccessful in spite of the successive modifications made to the design and different assembly methods (TZM/steel 316L, TZM/graphite). F4E is continuing its design efforts to find a technical solution that is compatible with the expected thermal loads.

CEA-IRFM is also managing the technical coordination of the project to develop (design, qualification) the concept of the targets of the future tungsten divertor for JT-60SA. In response to this need, innovative technologies (additive manufacturing, high performance plasma-facing material, etc.) are being evaluated in terms of performance improvement (fatigue resistance under heat flux above 15 MW/m², resistance during energy events in the order of the GW/m² for a few ms, etc.). In 2022/2023, a first concept based on the hypervapotron geometry was modeled and optimized on CuCrZr (Figure 2). Mockups based on this concept will be produced in 2024 by additive manufacturing methods (laser powder bed fusion), then tested and qualified under thermal load (HADES).

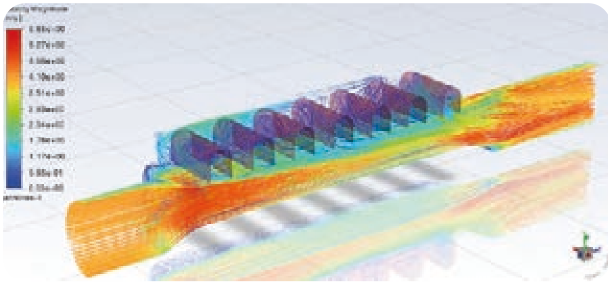


Figure 2: Current lines and fluid velocity on hypervapotron-type geometry – CFD modeling

Figure 1: (Left) Carbon divertor component (F4E sources); (Right) JT-60SA divertor sector

5. Diagnostics

5.1. Fibered LIBS tests in the WEST tokamak: a world premiere for fusion

The fibered LIBS (Laser-Induced Breakdown Spectroscopy) technique was tested for the first time in the vacuum vessel of a fusion device in January 2022 by teams from CEA (IRFM and DPC - Physical Chemistry department of the Energy Division) and CORIA[1], with support from CEA's transversal skills program for instrumentation and detection (EXLIBRIS projects). Installed on an inspection robot equipped with articulated arms (AIA - Articulated Inspection Arm), the LIBS tool is equipped with a fiber transporting the incident laser light and the light emitted during the beam's interaction with the material studied. It will enable characterization of all the in-vesselsurfaces of WEST and monitoring of their evolution during WEST operation.

During operation of a fusion machine, it is important to know the composition of the surfaces that interact with the plasma. The surface of these plasma-facing components (PFCs) evolves (erosion, oxidation, etc.), modifying the conditions of the interaction. For ITER, we also want to determine:

- The tritium concentration of the PFCs to monitor the tritium inventory, which is limited, inside the ITER vaccum vessel;
- The presence of helium bubbles in the PFCs resulting from the interaction of fusion neutrons with the materials or the disintegration of the tritium trapped in the PFCs. The helium bubbles deteriorate the thermomechanical properties of the PFCs designed to evacuate the high thermal flux (~10 MW/m^2).

The LIBS (Laser-Induced Breakdown Spectroscopy) technique is particularly suitable for these measurements. LIBS consists in focusing a pulsed laser beam for nano, pico or even femtosecond durations on the material to be analyzed to obtain a high laser intensity on its surface (Figure 1). The interaction between the laser and the material results in ablation of the material and creation of a plasma plume. Spectral analysis of the plasma emission provides information on the elemental composition of the ablated material and the composition of the material studied. This technique is widely used in industrial applications requiring regular inspection of material composition. It has also been used for remote characterization of the composition of rocks on the planet Mars [see the CEA/CNES ChemCam project for the Curiosity mission and the SuperCam system for the Perseverance mission].

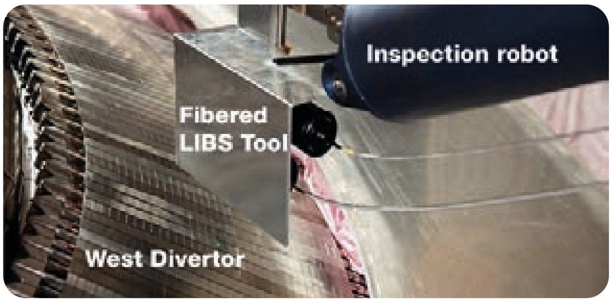


Figure 1: LIBS installed on the Articulated Inspection Arm

CEA-IRFM has an inspection robot with articulated arms, called AIA (Articulated Inspection Arm), which enables inspection of the condition of internal components during operation between plasma pulses thanks to an integrated camera.

A fibered LIBS tool was installed on this robot to enable characterization of the PFCs during operation (Figure 2). It was designed and tested with support from the CEA transversal skills program for instrumentation and detection (EXLIBRIS project) in close collaboration with CEA-IRFM, CEA-DPC (physical chemistry department of the Energy Division), and CORIA1.

The first in-situ tests took place in December 2021 and were implemented at atmospheric pressure. The nanosecond laser source and the spectrometer enabling optical analysis of the plasma created during laser shots were positioned outside the WEST vacuum vessel.

The AIA, equipped with the fibered LIBS tools, was deployed in the tokamak enclosure to take several measurements on one of the divertor's solid tungsten PFCs.

The figure 3 presents the evolution of the intensity of the copper, nickel and boron lines according to thickness, each dot corresponding to a laser pulse. These elements are impurities found in a layer deposited on the surface of the tungsten PFC being analyzed. The deposit layer results from the erosion of other PFCs due to plasma-material interaction, and transport of eroded material by the plasma.

From this figure, we can deduce that the layer deposited during plasma pulses is approximately 600 nm thick.

After this first fibered LIBS experiment in a tokamak, CEA-IRFM and CORIA's goals now include finalization of the developments to enable regular deployment of LIBS measurements on WEST, and consideration of the use of picosecond laser to enable precise measurement of hydrogen concentrations (H, D, T).

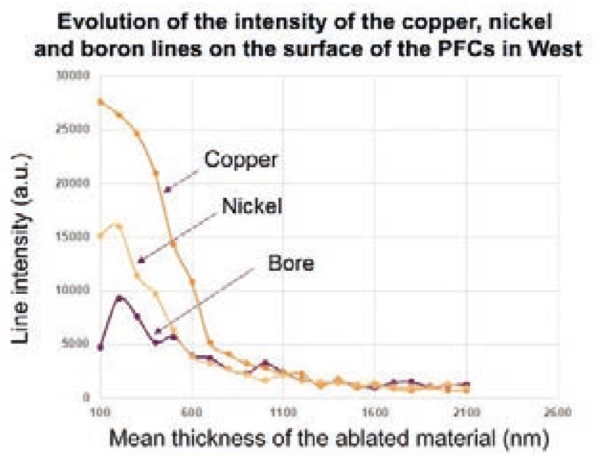


Figure 3: Evolution of the intensity of the copper, nickel and boron lines according to thickness, each dot corresponding to a laser pulse

5.2. PPPL diagnostics to examine every aspect of X-rays in WEST

In April 2022, in collaboration with Princeton Plasma Physics Laboratory (PPPL), CEA-IRFM equipped the WEST tokamak with three new diagnostics: multi-energy hard X-ray and soft X-ray cameras and a compact imaging system.

The WEST tokamak at CEA-IRFM specializes in producing steady-state discharges lasting up to several minutes. One of the key elements is the Lower Hybrid radiofrequency heating system (Lower Hybrid, LH, at 3.7GHz) installed on the tokamak. The 7MW LH power available in WEST enables generation of most of the current circulating in the plasma in a totally steady-state via the excitation of suprathermal electrons, i.e., electrons whose energy is much higher than average. Unlike the first configuration of the Tore Supra tokamak, a pioneering machine in terms of steady-state discharges, WEST has an internal wall made of tungsten. If even a tiny amount of this element reaches the plasma core, it can radiate a non-negligible fraction of the injected power, thus reducing fusion performance. Producing discharges in a WEST-type tokamak therefore requires detailed characterization of the population of suprathermal electrons to maintain spatial distribution of the current with the characteristics required for steady-state conditions, and characterization of the impurities in the discharge to prevent them from building up in the center of the plasma.

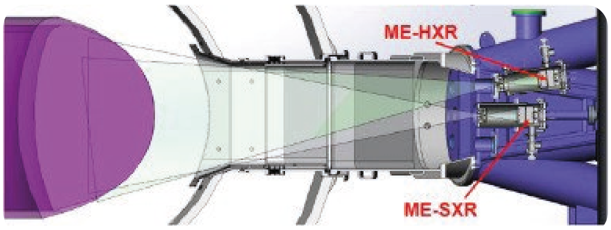


Figure 1: Integration of two hard X-ray (ME-HXR) and soft X-ray (ME-SXR) cameras from PPPL in the WEST tokamak environment. The plasma is shown in purple (left), and the two cameras measure the X-rays from outside the vacuum vessel.

The systems provided by PPPL enable these analyses by measuring radiation in the X range. The ME-HXR camera observes the plasma emission in the "hard X-ray" range. This type of radiation in the range of 10-100keV (wavelength 0.01-0.1nm), directly from the suprathermal electrons, provides information on the electron population generated by the LH wave, and therefore on how the current is distributed within the plasma. This diagnostic is based on a "PILATUS3"-type X-ray detector made up of cadmium telluride (CdTe) sensors spread over a matrix of approximately 100 thousand pixels. The ME-SXR camera measures "soft X-rays" corresponding to the energy range 5-10 keV (wavelength 0.1-0.2nm). The electron temperature of the plasma can be obtained from this measurement, along with information on the transport of impurities in the plasma, including tungsten. This camera is also based on a PILATUS3-type detector, made up of silicon (Si) sensors on a matrix of the same geometry as for the ME-HXR (100,000 pixels).

Figure 2: Arrival of the ME-HXR and ME-SXR cameras dispatched from Princeton at CEA-IRFM.

The two cameras were developed at PPPL, thanks to continuous collaboration with the CEA-IRFM teams to ensure the best possible conditions for their integration in WEST. Two post-doctoral students from PPPL based at CEA-IRFM prepared the integration (Figure 1). At the same time, they were developing synthetic diagnostics corresponding to each system, i.e. modeling to anticipate the physical measurements expected and thus enable optimization of their future use for the development of scenarios on WEST. In spite of the complications of the Covid pandemic, the cameras were sent from Princeton to IRFM, arriving safely on March 18, 2021 (Figure 2). They were then installed on the machine (Figure 3). A third "compact" X-ray imaging system (cXICS) will produce an X-ray range image of the plasma to provide a direct picture of the impurity content during a discharge.

This fruitful collaboration between the two laboratories provides WEST with innovative measurement systems to optimize the scenarios implemented for higher-performance steady-state discharges in the future.

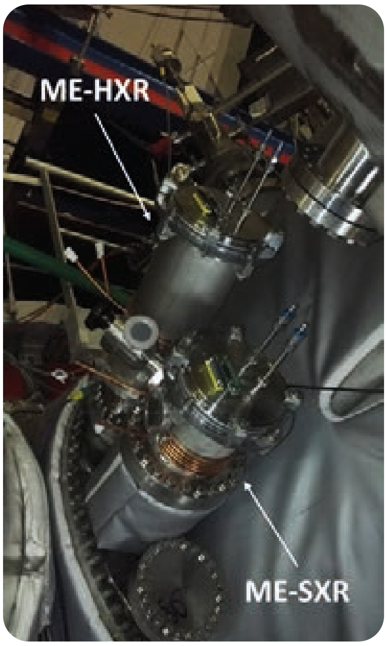


Figure 3: The two ME-HXR (top) and ME-SXR (bottom) cameras installed on WEST. Having passed the tightness tests, they are ready for the first measurements during the next experiment campaign on WEST.

5. Diagnostics

5.3. Integration of Fiber Bragg Gratings in a tokamak divertor

FBG sensors may be extremely useful in magnetic fusion facilities, being insensitive to electromagnetic interference and enabling temperature measurements to be taken at multiple points along a single fiber. These fibers, developed by CEA's LIST laboratory (integration of systems and technologies), were integrated for the first time in the tungsten plasma-facing components of the WEST tokamak divertor. They provide temperature measurements that are used to evaluate and monitor the performance of heat extraction from these components that prefigure those of the future ITER divertor.

An FBG (Fiber Bragg Grating) is an optical fiber in which a spatial periodic modulation of the refractive index is created in one segment. The measurement principle is based on the determination of the offset of the Bragg peak wavelength induced by the network period changes due to the fiber's expansion or retraction according to thermo-optic and elasto-optic coefficients. The modification of this network is measured by an interrogation device using a wavelength-tunable laser.

A compact, innovative system for multiplexed temperature measurement based on femtosecond Fiber Bragg Gratings (fs-FBG) was integrated in an actively cooled tungsten plasma-facing component of the WEST tokamak. This measurement system was developed as part of the FIBRA-W project of CEA's transversal skills in instrumentation program, involving CEA-IRFM, CEA-LIST and CEA-IRESNE.

CEA-LIST designed and produced it, along with the acquisition system. The manufacturing process involved femtosecond laser engraving methods [1] to produce fs-FBGs that are more stable at high temperature than conventional FBGs. The period and length of the networks are designed to provide fourteen networks per fiber (see the WEST fs-FBG spectrum in the first figure), regularly distributed along a 17cm length (providing a spot measurement for each tungsten monoblock, each of which is 12 mm wide). Measurements can be taken up to 1,200 °C with temperature gradients up to 200 °C/mm perpendicular to the fiber[2].

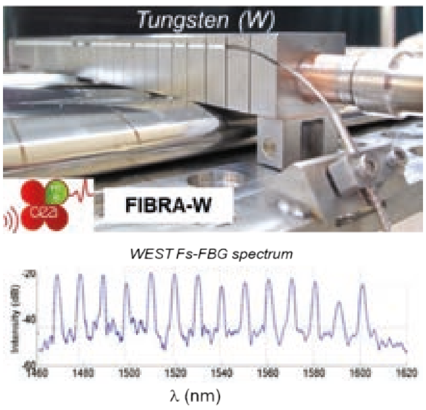


Figure 1: Integration of a Fiber Bragg Grating on an actively cooled tungsten plasma-facing component in W of WEST.

Fiber integration was performed by CEA-IRFM. The fiber is integrated in a 2.5mm groove engraved 5mm from the top of the tungsten components (Figure 1). This measurement system is installed in the tokamak area that receives the most intense heat flux. Five fs-FBGs were deployed, delivering a total of 70 (14x5) spot temperature measurements.

These measurements were taken during the experiment campaign from December 2022 to April 2023 on WEST equipped with its ITER-like divertor. The measurements are used in collaboration with Aix-Marseille Université (AMU), to evaluate the thermal flux reaching the component surface, in particular [3]. An example is provided in figure 2, which shows the temperature distribution according to the position of the tungsten elements. In this example, the maximal thermal flux and decreasing length of the thermal flux are calculated at 1 $MW.m^{-2}$ and 30mm respectively on the internal elements, and 4 $MW.m^{-2}$ and 10mm respectively on the external elements.

The fs-FBG system deployed in WEST is unique in the fusion community. The first experiment results show exceptional performance for measuring temperature and thermal loads on plasma-facing components. The tests carried out in a tokamak environment also illustrate how promising FBG technology is for all power-plant type environments, even at high temperatures, to guarantee good operating conditions for the components.

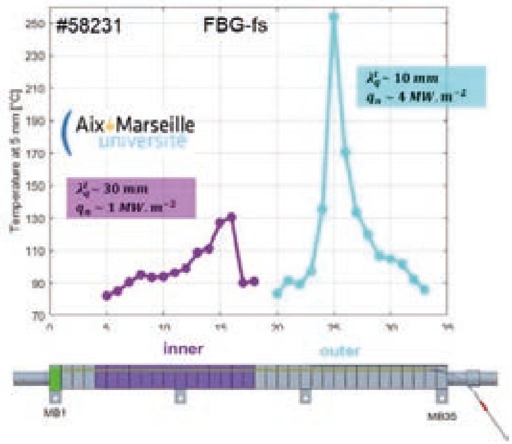


Figure 2: Temperature distribution according to the position of the tungsten elements

Références :

[1] A. Lerner, R. Cotillard, T. Blanchet, N. Roussel, G. Bouwmans, and G. Laffont, "An intrinsic sensitivity calibration scheme for high temperature measurements using femtosecond point-by-point written fiber Bragg gratings". Optics & Laser Technology, 2023. DOI: 10.1016/j.optlastec.2023.110278.

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5.4. CEA-IRFM continues its participation in the project to design a visible and infrared diagnostic for ITER

After successfully completing the final design review for the wide angle visible/infrared monitoring diagnostic for the ITER plasma vessel in equatorial port 12, CEA-IRFM is pursuing its activities with the European domestic agency Fusion for Energy (F4E) to complete its design for the other three equatorial ports (3, 9 and 17).

The first wall of ITER will mostly be protected by a wide angle monitoring system for the machine's internal components: the Wide Angle Viewing System (WAVS). This set of diagnostics, installed in four of ITER's equatorial ports, operates at visible and infrared wavelengths, making it an important instrument for tokamak protection. The first part of this diagnostic (three lines of sight from port 12 – Figure 1) must be operational for ITER's first plasma pulse.

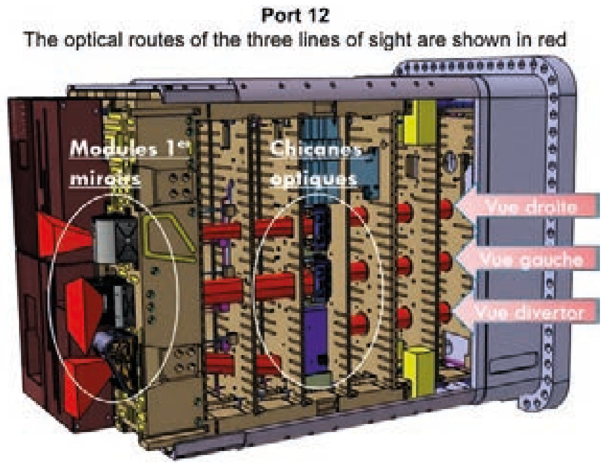


Figure 1: Line of sight of the WAVS and view of the optical routes inside the port-plug 12

Following the signature of the procurement arrangement in September 2019 by F4E and ITER Organization (IO), the team from CEA-IRFM presented the preliminary design review in December 2019 and the detailed review in July 2022 before a panel of international experts. These reviews concerned the components of the three lines of sight of port 12 located in the vacuum vessel of the tokamak; these components are being designed by CEA-IRFM. After each review, the panel congratulated the CEA-IRFM team for the robustness of its design.

The studies produced for the reviews showed that the complex opto-mechanical elements (cooled by pressurized water for those closest to the plasma), are compatible with the interfaces and conform to the requirements of ITER. These components must withstand severe constraints, including neutron and thermal flux, electromechanical and seismic forces. They enable the transfer of photons from the vacuum vessel to the inside of the tokamak building towards the sensors, and include advanced functions for calibration of the optic line or in-situ cleaning of the plasma-facing mirrors.

The European consortium that performed the studies on port 12 has been renewed on the basis of these good results to finalize the WAVS design for the other three ports. The consortium, coordinated by CEA-IRFM, includes the fusion laboratory of CIEMAT (Spain), the national INTA laboratory (Spain) and Bertin Technologies (France) via a contract with the European domestic agency for ITER, Fusion for Energy (F4E). The design level required for the other three ports will ultimately be the same as for port 12.

The WAVS design team also relies on the experience acquired during the production of several prototypes developed to validate the critical elements of port 12 (Figure 2). In particular, a shutter prototype attained more than 120,000 actuation cycles at 100 °C in ultra vacuum conditions, resulting in validation of its design.

All these positive results put CEA-IRFM in a good position to participate in the production and installation of these components in the different ports of ITER. Through its involvement in the design of this diagnostic, CEA is pursuing its long-term goal to participate in the scientific operation of the port on ITER.

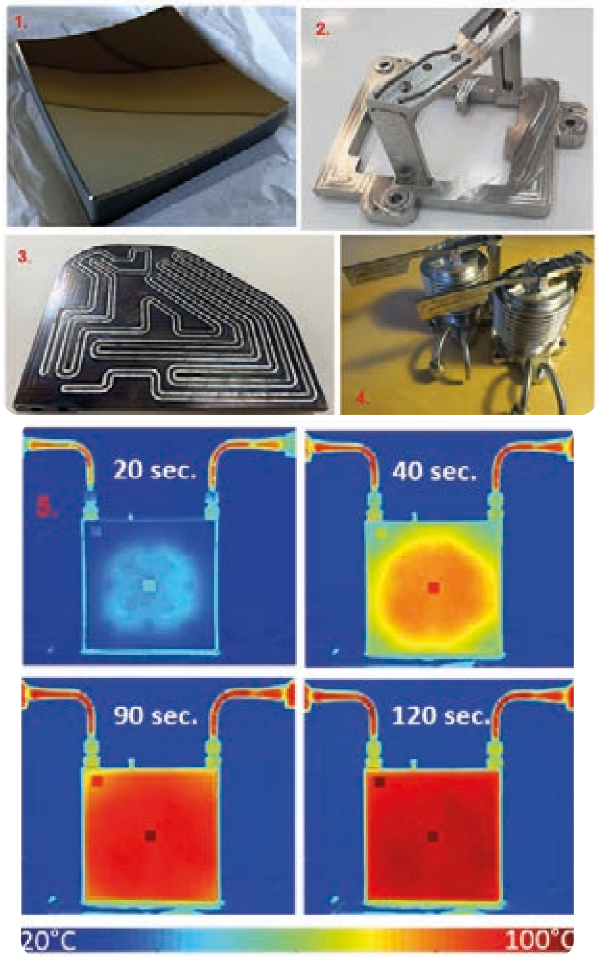


Figure 2: prototypes for port-plug components
1/ Mirror with rhodium surface coating
2/ Support flange for two front mirrors
3/ Lateral plate of the first mirror block with integrated cooling channels
4/ Shutter prototypes used for vacuum and atmospheric pressure tests
5/ Mirror cooling test on the SATIR facility

6. Tritium
6.1. Tritium transport modeling in the walls of the ITER divertor

In November 2020, Nature Scientific Reports published a study by CEA-IRFM researchers presenting a method to estimate the total amount of tritium trapped in the ITER divertor. This tool makes it possible to ensure that the amount of tritium remains below the limits set by the nuclear safety authority for different plasma scenarios.

Deuterium and tritium, two hydrogen isotopes, are the fuels for the fusion reaction of future tokamaks, such as ITER. In these tokamaks, the divertor is equipped with plasma-facing components (PFCs, Figure 1) and receives particles that escape the plasma. These components are subjected to high thermal flux (10 MW/m² continuous) and the flux of particles from the plasma, containing deuterium, tritium, helium and impurities. The hydrogen isotopes, including tritium, enter into the component materials of the divertor (tungsten, copper alloy). They can remain trapped or be diffused to the cooling water circuits.

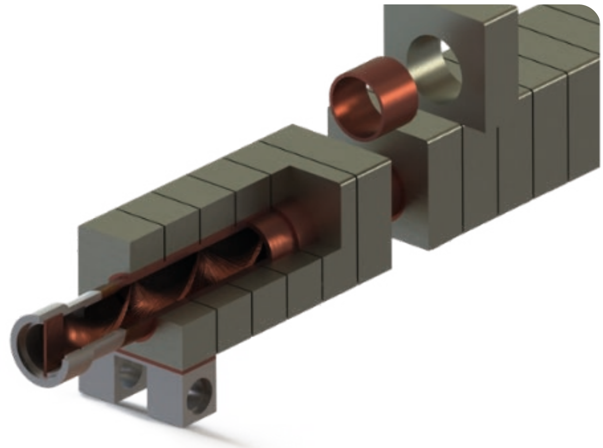


Figure 1: Example of an ITER-like actively cooled plasma-facing component.

For nuclear safety reasons, it is therefore crucial to be able to predict the amount of tritium trapped in the PFCs, as well as the proportion of tritium that will be diffused to the cooling circuits.

A study published in 2020 in Nature Scientific Reports [1] presents a method to estimate the total amount of tritium trapped in the ITER divertor. The behavior of the hydrogen in ITER's "monoblock" PFCs (Figure 2) was simulated using the FESTIM calculation code (Finite Element Simulation of Tritium In Materials) developed as part of a collaboration project involving CEA-IRFM and LSPM (Université Sorbonne Paris Nord).

By varying certain simulation parameters, such as monoblock surface temperature or incident particle flux, the divertor can be modeled entirely using optimization algorithms.

This modeling enables prediction of the tritium distribution in the monoblocks, the evolution of the amount retained in the divertor as a whole and the permeation flux to the cooling circuits (Figure 2). For various plasma scenarios, this tool makes it possible to ensure that the amount of tritium in the PFCs remains below the limits defined (750g for ITER).

The originality of this tool lies in its multi-physics coupling and 2D resolution. It is a key asset for the operational monitoring of reactors like ITER and its successor DEMO.

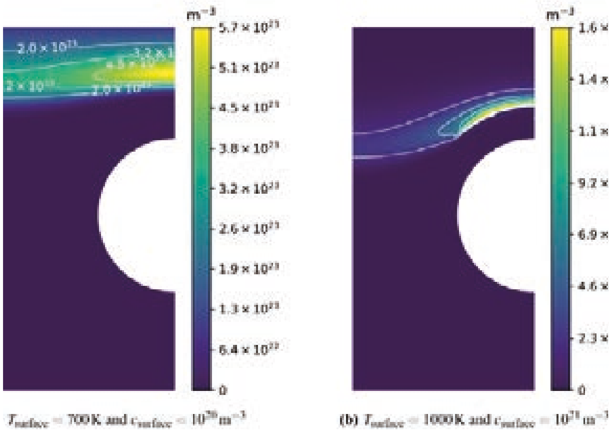


Figure 2: Results of the hydrogen concentration simulation in an ITER monoblock for a surface temperature of 700K (left) and 1,000 K (right).

References :
[1] Parametric study of hydrogenic inventory in the ITER divertor based on machine learning, Rémi Delaporte Mathurin[1,2], Etienne Hodille[1], Jonathan Mougenot[2], Gregory De Temmerman[3], Yann Charles[2] & Christian Grisolia[1], Nature Scientific Reports (2020) https://doi.org/10.1038/s41598-020-74844-w,

6.2. First tritium measurement in a material by laser ablation

The capacity of LIBS (Laser-Induced Breakdown Spectroscopy) measurement to discriminate between the presence of hydrogen and its deuterium and tritium isotopes in a sample was demonstrated by a laser ablation experiment on a titriated sample in 2022 as part of a collaborative project involving teams from CEA (IRFM, Joliot, DPC) and the CORIA laboratory (CNRS/Université de Rouen).

The European "TRANSversal Actions for Tritium" project (TRANSAT), involving 16 research institutes, was completed in late 2022. The purpose of TRANSAT was to improve our knowledge of tritium in fission and fusion facilities, and particularly the management of tritiated waste, tritium dosimetry, radiotoxicity and radiobiology (see ***https://transat-h2020.eu/***). One of the R&D actions was to study tritium content measurements in materials, in relation to the challenges raised by this content for nuclear safety and operations. LIBS (Laser-Induced Breakdown Spectroscopy) showed promise as a technique for monitoring the tritium inventory. It involves collecting and analyzing the light emitted by the plasma created during laser ablation of the material studied. Furthermore, this technique can be deployed inside the vacuum vessel of a fusion facility to take measurements on components interacting with the plasma.

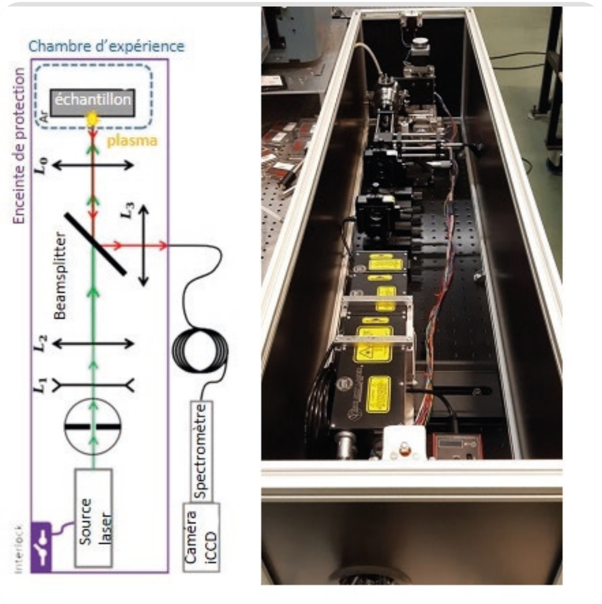


Figure 1: Principle diagram of the LIBS3H platform (left) and the experiment set-up (right). The beam from the laser source (in green) is extended by lenses L1 and L2, passes through the beamsplitter then lens L0, which focuses it on the tritiated sample in the experiment vessel (sealed). The laser-induced plasma (in yellow) emits light: the part (in red) reflected on the beamsplitter is focused by lens L3 and collected by the optical fiber leading to the spectrometer for analysis.

In March 2023, specific samples previously loaded with tritium were prepared and placed in a specially designed sealed enclosure to enable the laser-material interaction without dispersion of the radioactive dust created during ablation in the experiment vessel (Figure 1), which is kept at atmospheric pressure with argon. The detailed spectroscopy was then carried out around the tritium emission line by reconstructing the collected light spectrum. This study provides information on the presence of tritium in the ablated layer of material (Figure 2). The results confirm the potential of the LIBS technique for measuring the concentration profile of the tritium trapped deep in the materials (by progressive ablation). The method therefore enables better monitoring of the tritium inventory in the components of the vacuum vessel of a tokamak, where this information could affect operation.

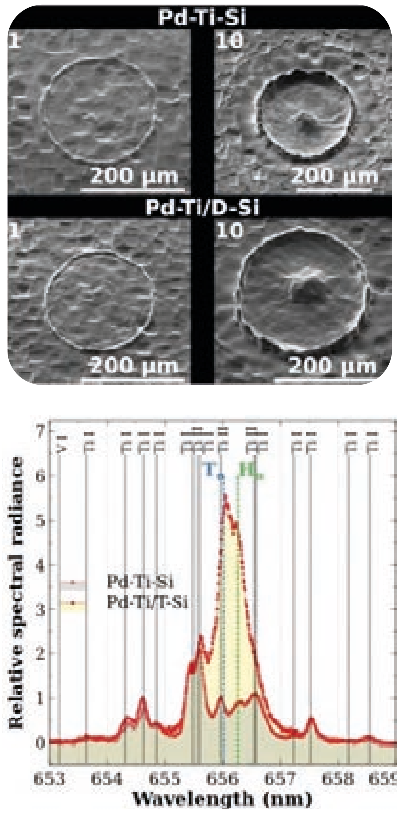


Figure 2: (A) Evolution of the laser ablation crater after 1 (Top) and 10 shots on two targets (bottom), one being neutral and the other implanted with tritium. (B) Normalized spectra obtained from a single laser shot, clearly distinguishing the contribution of the two hydrogen isotopes H and T (in yellow) compared with a non-loaded target (in gray).

The various results of TRANSAT opened new perspectives that led the European Commission to launch the follow-up project in September 2022: TITANS (Tritium Impact and Transfer in Advanced Nuclear reactorS), a three-year project coordinated by CEA-IRFM.

References

- *Double pulse laser-induced plasmas on W and Al by ps-LIBS: Focus on the plasma-second pulse interaction, Fusion engineering and Design 168, 2021, 112364, A. Favre et al.*
- *Towards ps-LIBS tritium measurements in W/Al materials, Fusion engineering and Design 146-B, 2019, 1971-1974) A. Bultel et al.*
- *Publications by TRANSAT institutes represent a broader outlet: https://transat-h2020.eu/resources/#1510849951728-ddfb0a1d-2024*

6.3. Virtual reality for ITER breeding blanket modules

Future fusion plants will have to produce their own tritium to feed the reaction by bombarding lithium with fusion neutrons. One of the goals of ITER is to test different technological solutions with breeding blankets containing lithium. Several TBS (Test Blanket System) mockups will be tested throughout ITER's service life to collect experimental data on their operation in a nuclear fusion environment. CEA is contributing to their design by proposing solutions for the integration and maintenance constraints with an approach that minimizes the doses to which workers are exposed.

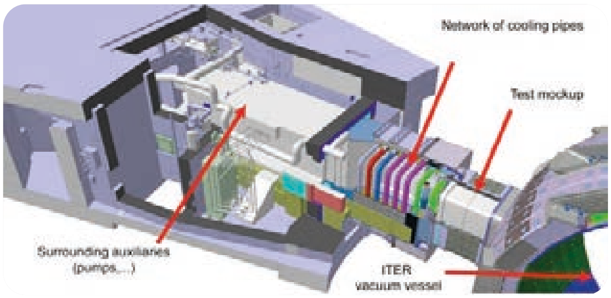


Figure 1: TBS system in the ITER environment

The various Test Blanket Systems (TBS) will be installed and used inside the ITER machine and regularly replaced after the test campaigns (approximately every two years). CEA has mobilized three operational divisions (DRF, DES and DRT), coordinated by IRFM, to design and validate the integration and replacement scenarios under a contractual agreement with ITER Organization. A first series of studies[1] completed mid-2022 enabled the TBS design to be adapted to the integration and maintenance constraints and particularly to the principle of minimizing worker dose exposure.

The TBS integration and replacement operations involve removing and reinstalling multiple elements (Figure 1) in a restricted environment under radiological control, for which the occupational exposure to radiation must be as low as reasonably achievable (ALARA[2]). ALARA principles must be integrated very early in the design process to identify and implement appropriate means, including design adaptations, to minimize the workers' exposure to radiation.

CEA thus uses its engineering skills in complex nuclear systems to address the various elements driving the design: requirements, architecture, technical options, configurations, environment and operations. To validate certain options, several individual feasibility evaluations are carried out to support the TBS systems engineering study. In collaboration with a CEA laboratory in Pierrelatte, the flange function options for the various nuclear piping configurations are being studied as an opportunity to substantially reduce worker exposure to radiation. At Nano-INNOV-Saclay, digital assistance and robotics are being investigated to reduce the time spent by workers in controlled areas. At CEA Cadarache, an "eXtented Reality XR-Lab" has been developed to validate by simulation the feasibility of certain interventions and to study human factors on a scale 1 digital mockup. This platform verifies the feasibility of the TBS replacement sequence and provides justifications for the TBS design options via an integrated approach on a digital mockup (Figure 2).

Figure 2: Illustration of a session in the CEA-IRFM XR-lab to check the feasibility of a dye penetration intervention.

The implementation of the ALARA optimization process with digital models in the early engineering phases of a TBS component was highly successful. The optimization process was tested on a pertinent example and the results were integrated in the design of the TBS component by ITER Organization. The digital model can thus integrate the results of physical tests being carried out in various CEA laboratories.

The XR-Lab facility, which was used a lot during the TBS design phase, manages a broad range of constraints (ALARA, human factors, work environment, system design options, etc.). The digital mockup is also helping to define the future physical scale 1 mockup of TBS that will be required by ITER Organization to validate certain TBS replacement operations (Figure 3).

Figure 3: Simulation of the installation of a scale 1 TBS mockup in a hall at DRF/IRFM. In the foreground, a heating antenna of the WEST tokamak and its auxiliaries illustrate the scale factor between a WEST component and an ITER component.

Références
[1] For example, see: J.P. Friconneau et al. Transactions on Plasma Science, 50(2022)4481.
[2] ALARA: As Low As Reasonably Achievable

7. Engineering
7.1. First parts produced by additive manufacturing for the internal components of WEST

The main part of a bolometry diagnostic and the central part of the water collector to cool the components of the WEST divertor were the first two elements produced in November 2020 using metal additive manufacturing techniques. The experiment campaign that followed their production subjected these components to tokamak operating conditions and formed the starting point for the integration of new innovative design elements using this method.

Metal additive manufacturing (see Figure1 for the principle) consists in producing a part, one layer at a time, either by material deposit or by laser powder bed fusion with minimal precision of around 0.1mm.

While the principle is simple, actual implementation is relatively complex. However, this innovative manufacturing process opens new perspectives and overcomes certain technological obstacles, particularly for the actively cooled parts of tokamaks.

It is now possible to produce monoblock parts with highly complex geometries, including equally complex cooling channels, very close to the external walls of the machine.

For WEST, mechanical studies carried out at CEA-IRFM regularly use this technology.

The first study concerned a bolometry diagnostic whose main component was produced by additive manufacturing. This choice was made on the basis of the complex geometry of the component and the hydraulic performance required. The actively cooled part has two separate cooling networks.

This first network, providing thermal protection of the diagnostic, is supplied by a first cooling loop (circuit B30: 30bars - 200 °C). The second network is supplied by a circuit (STEFI: 10 bars - 20 °C) and is designed to protect the measurement sensors. These networks are optimized to take up all the available space inside the component (Figure 1.1).

Aside from the hydraulic aspects, the decision to use additive manufacturing encouraged the integration of essential functions, thus reducing the number of parts and welds that would have been required if the assembly had been produced by more traditional machining methods.

The second study concerns the supply collector of the WEST divertor sectors (Figure 1.2). The problem was to optimize the hydraulic distribution while keeping load losses to a minimum. A molded solution met the requirements, but at a higher cost and with a longer production time than those of additive manufacturing. Only the central coupling of the collector was produced by additive manufacturing.

The tubular parts were produced by traditional boilermaking techniques. This type of collector has already been installed and is currently in operation in the WEST vacuum vessel.

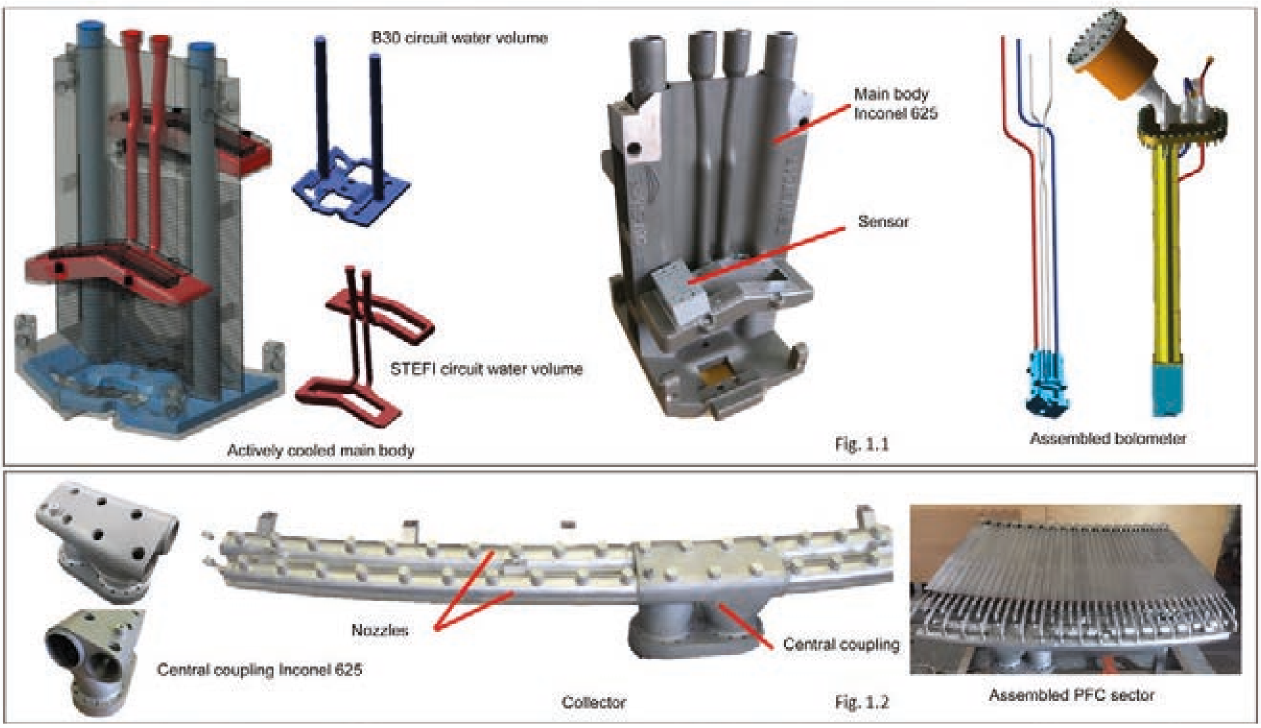


Figure 1: WEST parts produced by additive manufacturing.

However, this technology requires the implementation of non-destructive inspection protocols as well as the usual tightness and hydraulic tests.

Tomographies or gammagraphies (Figure 3) are appropriate and reliable methods for checking component integrity throughout the manufacturing process (presence of porosities, powder residues that might block the cooling channels, etc.).

Infrared thermographies (Figure 3) can be used to corroborate these results.

The first successful integrations of additive manufacturing components in WEST also open up possibilities for new design methods for complex tokamak components.

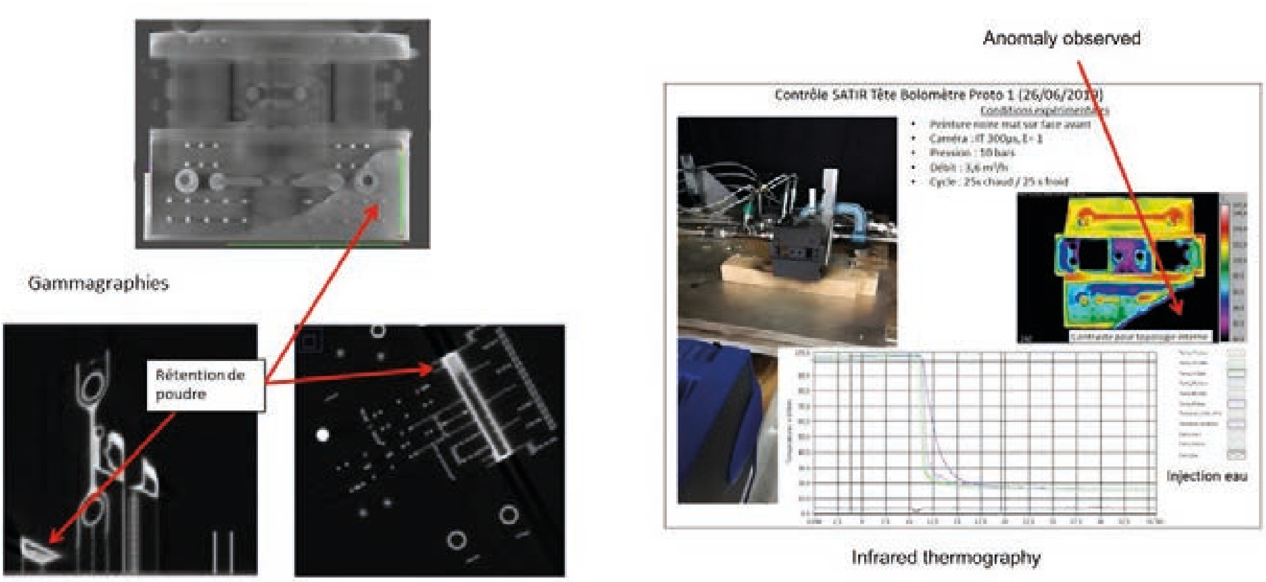
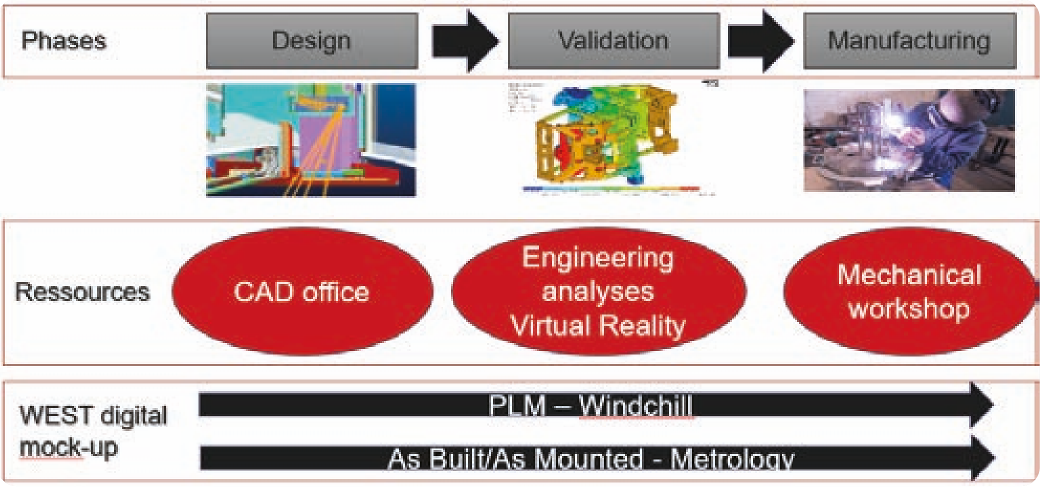


Figure 3: Gammagraphy and infrared thermography carried out on a component

7.2. Engineering activities for WEST, ITER and DEMO projects

The design office of IRFM is involved in most of the WEST enhancements ranging from ancillary systems, to the internal components of the tokamak, as well as new diagnostics. Additionally, it contributes to projects for other tokamaks: ITER, JT60-SA and DEMO. The design office has the expertise to handle complex mechanical objects and to ensure their compatibility with the severe environment of a tokamak (vacuum/pressure, electromagnetic and thermal loads). Design validation relies on engineering analyses (electromagnetics, fluid dynamics and thermo-mechanics) alongside virtual reality simulations and prototyping taking advantage of a mechanical workshop.



The design office is also in charge of maintaining the digital mock-up of the WEST tokamak. Leveraging a PLM (Product Lifecycle Management) software designed for 3D CAD design, CAD designers can access and make modifications or conduct integration checks on the machine's internals and auxiliaries. Presently, the 3D digital mock-up encompasses over 400,000 components (Figure 1).

Figure 1: West digital mock-up

The WEST tokamak is equipped with numerous measurement systems of plasma physical parameters. In particular, a complex Thomson scattering diagnostic is in its final development stage, aimed at characterizing density and electron temperature profiles from the central region to the periphery of the plasma. The design office has been extensively involved in designing both the edge (Figure 2) and central lines for several years. At the start of the year 2024, the central line has just provided its first measurements thanks to the interaction of the laser beam and the nitrogen injected for this purpose. In 2024, all the major components of the edge line have been manufactured and are ready for assembly. The full operation of the diagnostic is scheduled for the C10 campaign in the third quarter of 2024.

Figure 2: Endoscope of the edge line manufactured by March 2024.

Another notable achievement of the design office is the development of the infrared/visible Wide Angle Viewing System (WAVS) for ITER, a project spanning several years under a F4E contract. This system is composed of opto-mechanical parts designed to transport photons over a twelve-meter distance from the vacuum chamber to detectors located at the rear of the port-cell, taking part of the protection of the ITER machine. The key components, located in the vacuum vessel close to the plasma, must withstand extreme electromechanical loads, plasma heating and neutron flux. Therefore, all parts, included mirrors are water-cooled. They feature specific functionalities, such as a shutter to close the line of sight aperture and insulated mirrors, capable of acting as electrodes to initiate local plasma discharges for in-situ cleaning. The primary components, namely First Mirror Units (Figure 3) and Dog-Legs, can be remotely replaced in hot-cell if maintenance is required. The final design review for the three lines of sight of the equatorial port 12 was completed in July 2022. The design office is now focused on twelve lines distributed across three other equatorial ports.

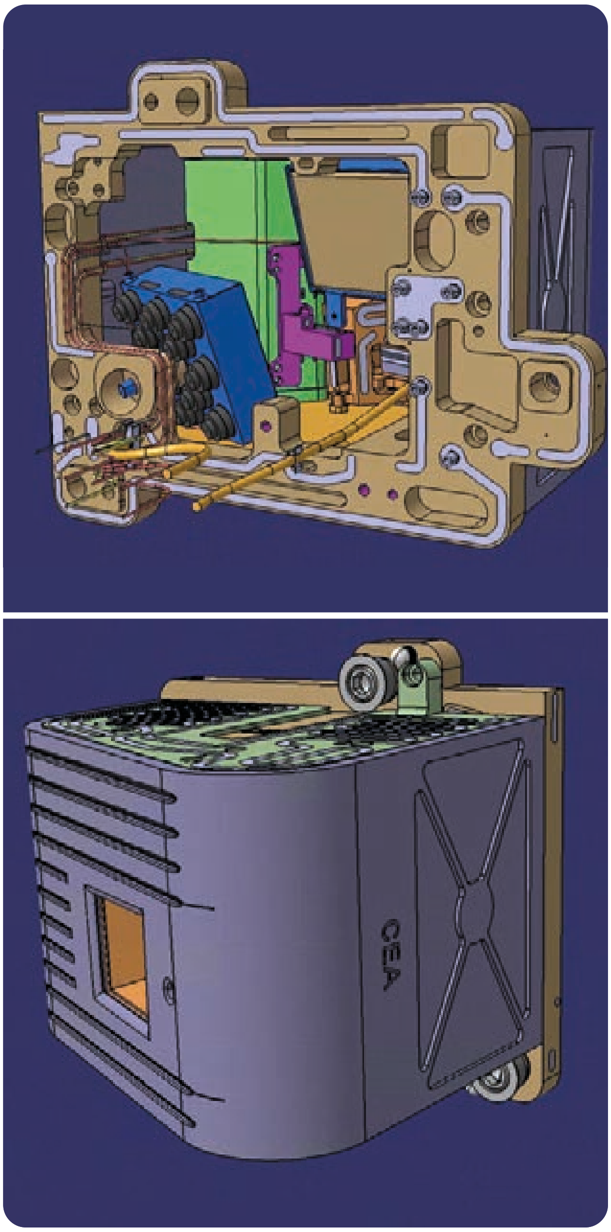


Figure 3: First Mirror Unit of the WAVS

The Virtual Reality platform of the design office consists of two well-equipped rooms, with one large enough to accommodate an ITER port-cell at scale one. This platform is extensively used to review the design of components whose installation on WEST is under consideration. In this case, a dedicated assembly plan is developed and played in the virtual room to assess the feasibility of each step. For instance, the assembly plan of the WEST divertor was developed to train operators prior to any maintenance inside the tokamak (Figure 4).

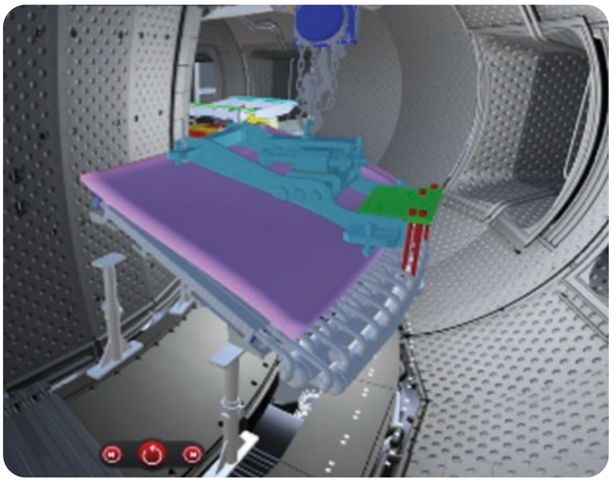


Figure 4: WEST divertor dismounting immersive sequence.

The VR platform is also involved in shaping the sequence of operations for the replacement of ITER TBM sets through a contract with the ITER Organization. The team actively contributes to the replacement sequence and to the design of the requisite tools. This sequence aims at minimizing radiation exposure as much as possible. To achieve this goal, each step of the sequence undergoes thorough analysis and optimization with the virtual platform. This process allows for adjustments to operational conditions as necessary (Figure 5).

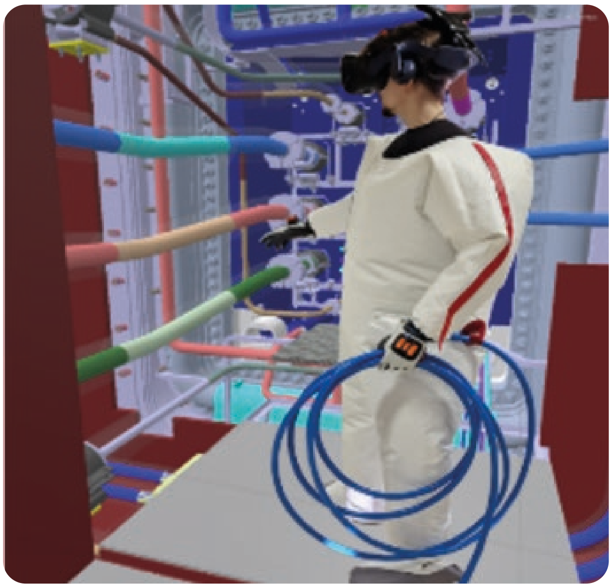


Figure 5: Simulation of TBM replacement with air-fed suit.

V Teaching, Communication

CEA-IRFM is deeply involved in many communication actions. The subject of fusion as a potential source of energy for the future is a societal issue that results in many different inquiries, including requests for interviews from journalists, visits to the site, etc. To explain the research carried out in the field of fusion and specifically at CEA-IRFM, our researchers also participate in events designed for the general public (Fête de la Science) and schools (ITER Robot, for example). CEA-IRFM also arranged a three-month exhibition produced by the EUROfusion consortium in 2021 in Marseille. CEA-IRFM organizes or helps to organize several fusion schools throughout the world and initiated the "Fusion pour Tous" (fusion for all) school in France. Finally, CEA-IRFM published the last WEST newsletters and three issues of the Sciences en Fusion journal between 2020 and 2023.

"Fête de la Science" Marseille 2022

Institute visits

All visits were canceled during the lockdowns in 2020, and were only resumed gradually in 2021 (about thirty visits, representing approximately 400 visitors). In 2022, roughly 70 groups were received, representing 1200 visitors and 90 groups with 1600 visitors came in 2023. The visitors are mostly students, many from abroad. Some overseas university professors come with new students every year for an in-depth visit of the institute. CEA-IRFM is also an active participant in the "Rencontres Enseignants chercheurs" event for researcher professors. Groups of professors are invited to participate in various workshops on the themes of fusion and energy.

CEA-IRFM participated in the CEA Cadarache center open day in 2022, receiving approximately a thousand visitors that day.

Visit of students from Singapore

Communication actions outside CEA-IRFM

CEA-IRFM took part in an original initiative in summer 2020: "Opération Sciences en Tongs" (science in flipflops). The idea was to reach out to young people, sometimes with their parents, at the camp sites and day camps around Cadarache. CEA-IRFM was involved in a session for a group of very curious children at the Tour d'Aigues day camp.

Every year, CEA-IRFM participates in the Fête de la Science. Although the festival was canceled in 2020, CEA-IRFM put on the "Fusion, Power to the People" exhibition in Marseille as part of the Fête de la Science event. The innovative exhibition was set up at Docks Village from October 8 to December 19, 2021. Open seven days a week, it was visited by almost 6500 people, mostly passersby who knew nothing about CEA or fusion. It served as an introduction to this subject and the visitors had plenty of questions for the scientists present to show them around. This communication action was an important event for CEA and the inauguration was attended by the manager of EUROfusion, the European Commission, the deputy mayor of Marseille and the director of CEA-IRFM, along with several journalists and around sixty guests.

In 2022 and 2023, ten CEA-IRFM researchers greeted several hundred visitors, many of whom were children, over three days at the Marseille science village.

CEA-IRFM has a strong commitment to the "ITER Robots" competition organized by CEA's ITER France agency. This operation, supported by the national education system, involves around 500 students for the final (from elementary to high school). They must get their robots to perform motion sequences, transport parts and cooperate with one another, as best they can. The students work all year on their robots, supported by CEA-IRFM engineers who are involved in the project reviews and participate in the juries for the final round.

European exhibition in Marseille « Fusion, power to the people »

Fusion schools around the world...

CEA-IRFM organizes a number of fusion schools around the world. The ASEAN fusion school (Association of South-East Asian Nations) "School on Plasma and Nuclear Fusion" is organized annually in Thailand. The year 2023 saw its 8th edition. The school aims to train students from throughout South-East Asia: from Thailand, obviously, but also from Indonesia, Vietnam, India, Singapore, etc.

The Plasma Physics and Fusion School of the MENA region (Middle East and North Africa) is a more recent initiative. It takes place in Hammamet, Tunisia. The first edition was held in February 2020, with the second and third editions in 2022 and 2023. Tunisia is the leading Arab-African country in the field of innovation and trains many engineers and scientists every year. CNSTN (National Center for Nuclear Sciences and Technologies) is developing its skills in fusion and plasma physics, which is why it organized the third edition of this school. Participants came from Tunisia as well as Algeria, Morocco, Lebanon, Mauritania and Senegal. The school was also organized by the Chinese laboratory ASIPP.

Finally, the first edition of the JT-60SA International Fusion School (JIFS), founded and directed by G. Giruzzi (CEA-IRFM) and Y. Kamada (ITER), was held in Naka (Japan), September 4-15, 2023. The twenty students selected (10 Europeans and 10 Japanese) took classes proposed by twenty-four professors from Europe and Japan, on a range of subjects related to fusion plasma physics, and tokamak technology and operation.

JT-60SA International Fusion School (JIFS)

...and in France

CEA-IRFM has been organizing a theory festival every two years since 2001. This international science event takes place in July every odd year in Aix-en-Provence. Due to the pandemic, the 11th edition in 2021 adopted an online class format. The 12th edition ran for four weeks, from July 4 to 29, 2022. As for the 2017 and 2019 editions, the first two weeks of the congress were devoted to the presentation of new interdisciplinary results, covering magnetic confinement fusion, astrophysics and geophysics, and classes for young researchers. Over the remaining two weeks, research projects were initiated by carefully selected brilliant young researchers, supervised by senior experts. A conference-debate, a fringe event for the public, was organized at Aix-en-Provence city hall, attracting around a hundred attendees.

CEA-IRFM organized the second Tritium school in a digital format in 2021. The school was organized as part of the TRANSAT (TRANSversal Actions for Tritium) project, a European Horizon 2020 project coordinated by CEA. The third edition, with physical attendance, is planned in Marseille in 2024.

In 2022, CEA-IRFM started the "Fusion pour Tous" school. This school aims to provide people directly or indirectly involved in the development of fusion energy, regardless of their prior knowledge, with a solid base to be able to grasp the very broad range of topics related to fusion. The school will enable better team integration and facilitate the processing of the multiple interfaces specific to fusion. It takes place over a week and alternates between theory classes in the morning and visits in the afternoon. Although the first edition was reserved for CEA personnel, the second edition, in 2023, was open to the public. As with the first edition, the feedback from participants was very positive.

Finally, in parallel to these Fusion schools, CEA-IRFM is also an active contributor to the Masters 2 program on fusion, hosting students for a month. After a three-year interruption, the first group of M2 students since the Covid pandemic came together from February 13 to March 10, 2023. The students from the national Masters 2 degree in fusion (Université de Lorraine, Aix-Marseille Université and Paris-Saclay) worked in small groups on projects for two weeks, supervised by physicists and engineers from IRFM. Over the next two weeks, M2 students from the European FUSION-EP Masters program worked on similar projects. On the last day, their first-year colleagues joined them for a visit of IRFM and attended their oral presentations.

"Fusion pour tous" school

Conference organization

The 28th IAEA Fusion Energy Conference (FEC 2020) was organized jointly by CEA-IRFM and ITER Organization. Initially planned for October 2020 in Nice, it was postponed until May 10-15, 2021, and finally held in an all-online format because of the restrictions due to the health crisis. The organizers welcomed 840 participants and almost 2000 "observers" to a virtual congress hall. For the 60th anniversary of the largest international conference on fusion, this rather unusual format, which had become normal since mid-2020, was a huge success, with a high rate of attendance at the various scientific presentations (an average 500 people) and numerous questions in spite of some participants (and speakers) being in different time zones. With this digital format, IAEA chose to open the conference to all those interested in fusion: anyone could watch the presentations, but not ask questions. Just under 2000 people logged on to attend the scientific presentations or events organized at the same time. An interesting fringe program proposed virtual visits of various facilities (WEST, ITER, DIIID, RFX, etc.), webinars (from Tore Supra to WEST, Ansaldo activities, Air Liquide activities, etc.) and events such as "Women in Fusion" or FEC's 60th anniversary celebration.

CEA-IRFM also helped with the MT28 (Magnet Technology) conference in 2023, along with ITER Organization. This conference took place from September 10-15 at the Grand Théâtre de Provence in Aix-en-Provence, with parallel sessions at the Conservatoire and Pavillon Noir; venues usually reserved for dance recitals or concerts. The conference brought together around 1,000 participants, prefiguring the SOFT conference that CEA-IRFM will be organizing in similar conditions in 2026.

FEC 2020 - Virtual convention center.

WEST Newsletters

A newsletter was published by CEA-IRFM throughout the WEST project to keep people abreast of the progress achieved. Once the tokamak started operation, the newsletter was published less frequently. The last three issues were published in 2020, 2021 and 2023 for edition no. 30, the final edition of this long series of newsletters that began in 2013.

Sciences en Fusion

The Sciences en Fusion journal was created in 2018 under the impulse of CEA-IRFM and the French Federation of Research on Magnetic Fusion-ITER (FR-FCM), which comprises several higher education establishments. It aims to provide information on scientific progress in the field of fusion to a broad audience, while highlighting the synergies with other themes, such as astrophysics, for example.

Issue no. 3, "Interfaces" on plasma-wall interaction was published in 2020; the next edition in 2022, entitled "Simulation" concerned digital simulation in the field of magnetically confined plasma research. The latest edition, "Ondes" (waves), written in 2023, is all about electromagnetic waves and their applications for fusion plasmas.

An important ceremony for WEST

The WEST tokamak started operating in its nominal configuration with a full solid tungsten divertor in 2023. This new phase in the machine's life cycle began with an experiment campaign full of promising results for ITER and future fusion machines. On June 6, 2023, under a star-studded ceiling, four key players from the WEST project told the story of the divertor to some 400 people who had come to celebrate this major milestone in the life of CEA's tokamak. The ceremony opened with a sound and light show about the WEST tokamak, accompanied by a video to illustrate a brief description of the divertor's history. Jérôme Bucalossi, director of IRFM, greeted the participants and explained the importance of this component for tokamak-type fusion machines, before handing over to Christophe Bourmaud, the director of CEA Cadarache, then Mr. Mamis, the right hand man of the Regional Prefect and Secretary General for regional affairs, who recalled the regional council's support for IRFM's projects. Christel Fenzi, manager of IRFM partnerships, spoke about the importance of the WEST tokamak in the international landscape of fusion machines. Pietro Barabaschi, Director of ITER Organization, closed the ceremony with a statement about the unique role played by WEST in support of the ITER project.

L'IRFM in the media

More and more, there is significant media interest in fusion. Advancement of the ITER project, announcements concerning new start-ups, progress achieved by other laboratories elsewhere in the world are just some of the reasons for journalists to contact CEA's specialists for an explanation of the scientific challenges of fusion. In 2022 and 2023, multiple articles were written in France about the record set by the European JET tokamak and the launch of the JT-60SA tokamak in Japan and CEA-IRFM researchers were asked to comment. Several journalists were present at the WEST ceremony on June 6: the event was reported on regional TV and several articles were published over the course of the following week.

VI Collaborations

The CEA-IRFM is the central player in France for all research activities on magnetic confinement fusion aimed at the industrial production of fusion energy. It fulfils its mission by relying on a structured collaboration network aligned with its priority research areas, covering tokamak operation, plasma experimentation, plasma numerical simulation and fusion engineering as main fields of activity. This collaboration network is threefold, including the French Federation on Magnetic Fusion Research at national level, the EUROfusion Consortium at European level, and a number of bilateral collaborations at international level in particular with the USA, China and South Korea (see the IRFM collaboration network map illustrated in Figure 1).

Despite of the Covid-19 pandemic and the still very complex international context, the Institute was able to maintain a good level of interaction with its partners over the period 2020 – 2023 with significant progress in the activities, keeping the preparation to ITER operation and scientific exploitation as main structuring and priority element. WEST being the single superconducting and actively cooled tokamak operating in tungsten environment in Europe, it offers unique and attractive assets to the international fusion community. The completion of its full actively cooled ITER-grade tungsten divertor installation in 2022 has unequivocally strengthened its position in the international landscape of superconducting tokamaks alongside EAST and KSTAR, with potential new synergies in support to ITER needs where a tungsten first wall (instead of beryllium) is now considered.

Figure 1: Mapping of the IRFM collaboration network

CEA-IRFM at the center of a national network

The national academic research in magnetic confinement fusion is coordinated since 2007 through the "French Federation on Magnetic Fusion – ITER" (FR-FCM), created on the initiative of the CEA[1] and the CNRS to organize the national effort and prepare the community for the scientific exploitation of ITER. The FR-FCM is an agile and multidisciplinary federative structure, which now brings together expertise from more than 40 academic laboratories, from various research institutions or higher education establishments (see Figure 1).

Figure 1: The FR-FCM network

While the FR-FCM community is mainly spread over three sites (Aix-Marseille, Paris area and Nancy), the Aix-Marseille site plays a central role with the geographical proximity of the WEST tokamak (and later ITER). This role has been further structured around the creation of the Institute for Fusion and Instrumentation Sciences in Nuclear Environments (ISFIN)[2], in which the IRFM and the CEA have been strongly involved, and which came on stream in 2020. In the meantime and at the request of the French Higher Education Research and Innovation Ministry, the FR-FCM has drawn up a national strategy proposal shared by all national players, and proposed a national roadmap for 2020 – 2030 based on two axes: experimentation on WEST and digital simulation. The monitoring of the activities carried out by the FR-FCM and discussion on new prospective activities were on the agenda of the Fifth Prospective Symposium of the FR-FCM held in May 2022 in Montauban, which was the first face to face meeting following the Covid-19 pandemic, gathering more than 70 participants from 25 French laboratories.

CEA-IRFM participation in the EUROfusion programme

Since 2014, the fusion community in Europe is well and strongly structured around the EUROfusion Consortium[3] which currently consists of 30 research organizations, and behind them 152 associated entities including universities and companies, from 25 European Union Member states plus the United Kingdom, Switzerland and Ukraine as associated Members. EUROfusion aims at supporting and funding fusion research activities on behalf of the European Commission's Euratom programme, with a roadmap [4] towards the realization of fusion energy. The Consortium Members participate in the programme activities through competitive calls. The CEA-IRFM has been leading the French Beneficiary [5] participation in the programme as third contributor (behind Germany and Italy) since 2021, under the 2021 – 2025 Euratom programme under the Horizon Europe framework. Its contribution is mainly focused on the following sub-programmes:

- WEST operation and scientific exploitation, which complements the substantial effort on tungsten related issues ongoing both in tokamaks (AUG, JET) and plasma wall interaction devices (Magnum PSI, Judith, Gladis, PISCES ...) by providing an integrated test environment for the ITER divertor operation, but also allows for the exploration of long pulse operation scenarios over relevant plasma-wall equilibrium time scale in metallic environment, the assessment of full tungsten wall conditioning techniques in a superconducting metallic long pulse tokamak, or the development of tools for safe operation of actively cooled metallic plasma facing components to a high reliability and adaptability using AI;
- JT60-SA commissioning and scientific exploitation preparation (the current European experimental team leader being from IRFM), and the first international JT60-SA Fusion School organization;
- digital simulation, with the lead of four out of 14 simulation projects aiming at developing European standard numerical tools on the following high priority topics:
  - boundary plasma modelling towards reactor relevant simulations;
  - impurity sources, transport, and screening;
  - dynamics of runaway electrons in tokamak disruptions;

- reliable prediction of plasma performance and operational limits in tokamaks.
- divertor R&D studies for DEMO, JT60-SA, DTT and W7X;
- breeding blanket R&D studies;
- specific activities on synthetic infrared diagnostic and AI tools development, neutronics, neutral beam test facility in Padova, in the preparation of ITER operation and scientific exploitation.

Bilateral cooperations outside the EUROfusion framework

CEA-IRFM is also involved in numerous bilateral collaboration activities outside the EUROfusion framework, including GEM tomographic system exploitation and W transport studies with IPPLM, Soft and hard X-ray tomographic methods development with IF JPAN, and infrared radiation bolometer design studies for WEST with the American University of Beirut (AUB). In addition, the IRFM is deeply involved in the coordination of the DTT [6] Research Plan. The collaboration agreement signed with the DTT Consortium in 2022 also covers a number of supporting activities, including in particular the participation and training on WEST operation (advanced plasma control strategies for long pulse plasmas, ICRF heating system), scenario and physics simulations with CEA-IRFM tools, and an IMAS data base system development.

France-China cooperation largely covered by SIFFER

the four-party SIFFER agreement (Sino French Fusion Energy Center) involving ITER-China, CNNC/SWIP, CAS/ASIPP and CEA/IRFM, has covered most French/Chinese cooperation projects in the field of fusion since 2017. This cooperation includes the development of fusion technologies and physics R&D, relying on joint activities in the scientific operation of French and Chinese fusion machines (WEST, EAST, HL-2A, HL-3), and R&D activities supporting design studies for future fusion machines (ITER, CFETR, etc.). From 2020 to 2023, these activities concerned seven SPAs (Specific Project Agreements) under a strong cooperation approach between the three laboratories SWIP, ASIPP and CEA-IRFM on the subjects of integrated fusion plasma simulation, the ITER-grade divertor for WEST (supervision of manufacturing, installation and operation), infrared technology for monitoring plasma-facing components, MHD control, development of actively cooled Boron/CuCrZr protective elements for the RF antennas, and IMAS deployment for data management.

SIFFER
FUSION TOGETHER

EAST and WEST for BEST

A collaborative action specifically concerning the BEST project[7] was launched in 2022 with ASIPP. Focusing on different scientific and technical topics (cryogenic systems and superconductive magnets, evaluation of heat flow, advanced materials, nuclear safety, RF heating), the collaboration also includes joint experiments on EAST and WEST to prepare for the operation of BEST. As part of this initiative, the CEA-IRFM site has already hosted two ASIPP employees for several months.

[1] Within CEA, the IRFM carries out all research activities on tokamak plasma physics, and interacts with other institutes in the field of cryotechnology, nuclear engineering of fusion systems (for issues related to the design of nuclear components, materials under neutron irradiation, tritium cycle, safety and decommissioning of fusion facilities), heterogeneous materials assembly and robotics for nuclear maintenance.

[2] The ISFIN fosters training and research in fusion sciences, nuclear instrumentation, and the mechanical characterization of materials and structures for fission and fusion, with an interdisciplinary approach, including societal aspects. It involves strong collaboration with many laboratories from Aix-Marseille University and other academic institutions of Marseille, with synergies between fusion and fission.

[3] https://www.euro-fusion.org

[4] https://www.euro-fusion.org/eurofusion/roadmap/

[5] The French Beneficiary consists of the CEA-IRFM as representative, plus participants from other CEA institutes (in particular form the Energy Division), as well as some twenty French academic laboratories from the FR-FCM involved as linked third parties.

[6] The tokamak DTT, currently under construction at the ENEA/Frascati site, is expected to start operation at the end of the decade. The main goal is to test and demonstrate innovative solutions for the heat and particle exhaust for a fusion reactor. The project is supported by national funds, with a contribution from EUROfusion in the divertor design studies and manufacturing.

[7] The BEST tokamak (Burning plasma Experimental Superconducting Tokamak) currently under construction in Heifei, will demonstrate control of burning DT plasmas. Operation is due to begin in 2027.

Collaboration between WEST and KSTAR

The perspective of commissioning the actively cooled tungsten divertor KSTAR triggered the launch of a new collaboration between CEA-IRFM and KFE in 2023. It mainly concerns the operation and control of plasmas over long durations in a tungsten environment, and the challenges of plasma / plasma-facing material interactions. This collaboration with South Korea also includes the use of an ECE imaging diagnostic (ECEI) in WEST with UNIST and POSTECH (the first ECEI measurements were obtained during the 2023 WEST experiment campaign), and operation of a VUV spectroscopy diagnostic to study the transport of impurities with KAIST.

A Joint Research Center at NTU, Singapore

Following the NTU Chair Professorship established in 2023 with IRFM, aimed at supporting new research programs and training skilled workers to advance relevant technologies in the field of magnetic fusion, the SAFE (Singapore Alliance with France for Fusion Energy) Joint Research Center was created under a new CEA-NTU Framework Agreement on fusion energy. SAFE is supported by a five-year grant from the National Research Foundation, Singapore. Its objectives include the theory and modeling of magnetized plasmas using first-principle simulations and artificial intelligence tools, the development and implementation of gamma spectrometry diagnostics in WEST supported by an extensive modeling program on disruptions.

SAFE Signature

Collaboration with very dynamic American laboratories

Since 2020, six projects supported by DOE have been launched with PPPL, ORNL, MIT, University of Tennessee (Knoxville) and University of Illinois (Urbana/Champaign), in collaboration with IRFM, on a certain number of topics that are particularly relevant to WEST, such as the study of tungsten sources and transport, tungsten as a first wall material, wall conditioning, RF heating and the use of HF models for wave coupling, the effects of the plasma-wall interaction, current generation. The projects also include the supply of X spectroscopy diagnostics (soft X-ray, hard X-ray, compact XICS, etc.), a boron powder injector, a spectroscopy diagnostic to measure electric fields in front of the antennas, and development of an emissive probe to measure the magnetic field generated by the ICRF waves.

Visit of the DOE

VII Awards

Samuele Mazzi, double winner at the 47th EPS Conference of Plasma Physics

Samuele Mazzi, AMU-CEA PhD student, obtained two awards, the «PhD Student Poster Prize» and the «Itoh Prize» at the last EPS conference.

The PhD Student Poster Prize is awarded to the best poster displayed at the conference by a PhD student. The prize is sponsored by the Plasma Physics and Controlled Fusion (PPCF) journal, the European Physical Society (EPS) and the International Union of Pure and Applied Physics (IUPAP). There is strong competition for this prize because most of the PhD students presenting a poster at the conference apply for the award. The jury is composed of members of the conference program committee.

Samuele Mazzi won the prize in July 2021 with his poster "Suppression of ion scale turbulent transport by MeV range fast ions at JET."

The Itoh prize, also awarded during the conference, is for PhD students who have carried out excellent research in plasma turbulence, transport, confinement or any other related subject. The competition is judged by a panel of experts, chaired by Professor Akihide Fujisawa from Kyushu University. Samuele Mazzi will have the opportunity to visit Kyushu University during a one-week, all expenses paid trip to Japan.

For his thesis, Samuele Mazzi analyzed the interaction between the very high energy ions generated by the ICRH antennas (up to 2 MeV) that destabilize the very high frequency plasma fluctuations called Alfvén modes (AE for Alfvén Eigenmode), and the turbulent transport caused by fluctuations related to ITGs (Ion Temperature Gradients). Non-linear gyrokinetic simulations using high performance computing (HPC) clearly show that the improvement in thermal ion confinement achieved at JET in such conditions is due to a multi-scale interaction between AEs and ITGs, which leads to elimination of the turbulent transport. These results suggest that for ITER, where the very high energy alpha particles generated by fusion reactions could destabilize the Alfvén modes, the confinement will be improved by these modes rather than damaged, as has been suggested in the past.

Samuele Mazzi is completing an AMU-CEA thesis at PIIM laboratory (supervisor: David Zarzoso), with joint supervision from CEA-IRFM (Jeronimo Garcia) and joint funding from INSTN (Institut National des Sciences et Techniques Nucléaires, French school for nuclear energy and health technology).

Xavier Litaudon: Centrale Marseille prize for 2021

This prize showcases a former student whose career has been successful and serves as an example for the future generations of students at Centrale Marseille engineering school. The Board of Directors of Centrale Marseille Alumni votes for the winner and the prize is awarded at the General Assembly of the school.

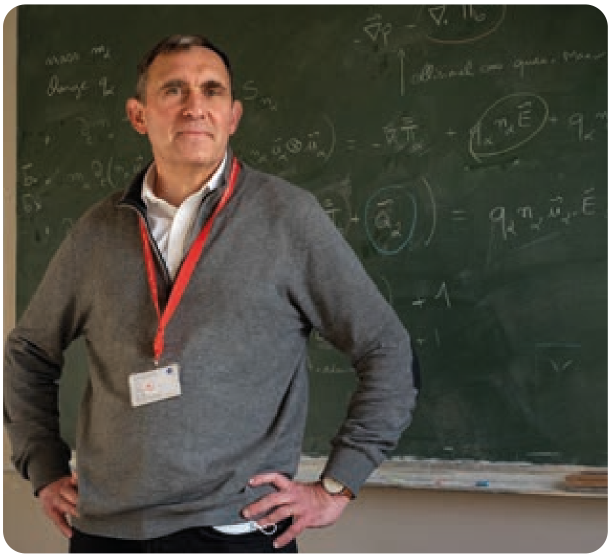
Xavier Litaudon won this prize in 2021 for his career in cutting-edge research and innovation in the field of nuclear fusion.

Xavier Garbet, winner of the European Physical Society's Alfvén award 2022

Xavier Garbet was rewarded for his major contribution to the theory of mesoscopic dynamics of magnetically confined fusion plasmas, winning the Alfvén prize traditionally presented at the European Physical Society plasma physics conference.

Winner of the CNRS silver medal in 2010 and the Holweck prize in 2019, Xavier Garbet has received another prestigious prize in recognition for his work: the Alfvén prize from the European Physical Society. Named after the Swedish astrophysicist who won the Nobel prize for physics in 1970, this prize was created by the EPS division of plasma physics in 2000 to promote results that have already shaped the field of plasma physics or shown potential for doing so in the near future.

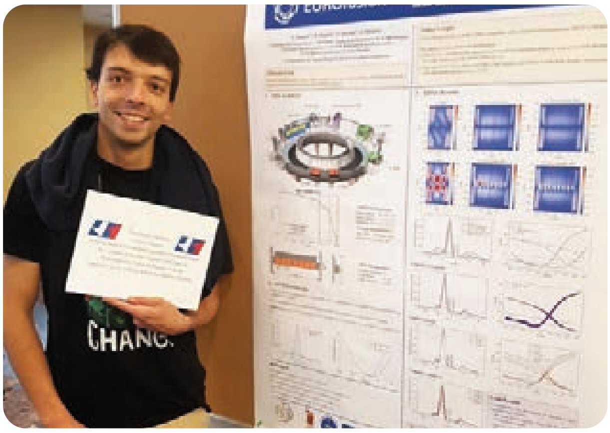
The prize was awarded for his contribution to the theory of mesoscopic dynamics of magnetically confined fusion plasmas and particularly for new insight into the spreading of turbulence, and for his work on flux-driven gyrokinetic simulations, transport barriers and edge plasma instabilities.



Vincent Maquet prize for the best poster at the RF 2022 conference in relation to the TWA antenna project for WEST

The scientific committee of the RF 2022 conference, held in late September 2022 in Annapolis, USA, awarded the prize for the best student contribution to Vincent Maquet from ERM laboratory (Brussels), for his optimization of a TWA-type antenna for WEST.

TWA (Traveling Wave Array) antennas for heating plasma offer many advantages in terms of performance and mechanical simplification, provided they can be installed in the vacuum vessel, because they take up a lot more room than is generally available in the ports. Although this is not a new kind of antenna, it was only recently proposed by the ERM laboratory (Brussels) for the heating of plasma ions (ICRH). A TWA antenna prototype was successfully tested on CEA-IRFM's TITAN test bench and the results were presented at the RF 2022 conference by Riccardo Ragona (DTU/ Denmark, formerly ERM).



Rebecca Riccioli, winner of the Paul Caseau prize 2022

Rebecca Riccioli received the Paul Caseau prize for the scientific excellence of her research work.

The Paul Caseau prize was created in 2012 in memory of Paul Caseau, founding member of the National Academy of Technologies of France (NATF) and former R&D director at EDF. It is sponsored by NATF, the Institut de France and EDF through their science and teaching program. It rewards particularly remarkable thesis work in the field of energy.

Rebecca won the prize in the "modeling and digital simulation" category for her thesis entitled "Mechanical modeling of superconducting cables for fusion under cyclic electromagnetic and thermal loads."

Robin Varennes, winner of the thesis prize from the Doctoral School of Physics and Sciences of the Matter

Robin Varennes was awarded this prize from Doctoral School 352, Aix-Marseille Université, on June 16, 2023.

Robin Varennes made a presentation at Doctoral School 352 (ED352)'s scientific day on Friday June 16, at the ceremony for his thesis prize 2022, accompanied by other winners. The subject of his work is "Flow drive in tokamak plasmas: competition and synergies between turbulence and collisional effects."

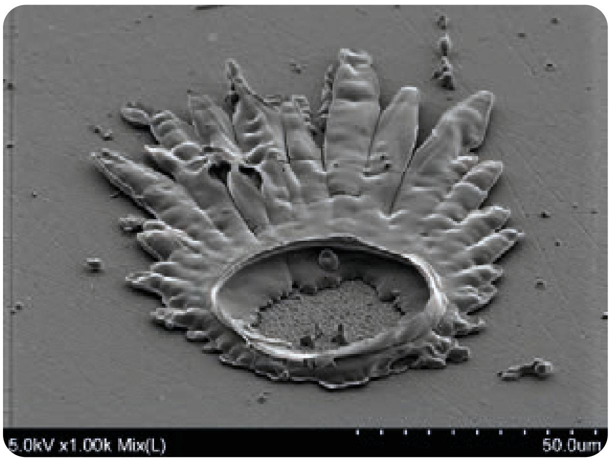
The two best projects, including his, are also in the running for the Marseille city thesis prize and the Aix-Marseille Université thesis prize.

First prize for "Tungsten corolla to star power" for Elodie Bernard

In 2023, this picture by Elodie Bernard won the jury's first prize: the impact of a melted particle on one of the samples placed on the first wall of the WEST tokamak from the collaboration project involving CEA-IRFM and NIFS that was renewed in 2023.

Tungsten samples are prepared in Japan then placed in sample holders fastened on the first wall of the tokamak. They experience all the events of the campaign, from conditioning to the plasma discharge phases. Analysis by a scanning (in this case) or transmission electron microscope enables in situ examination of the evolution of the material surface experiencing thermal cycling, boronization and exposure to particles escaping the magnetic field lines as the changes in plasma conditions occur.

Find out more about the 2023 edition of the "科学の幽玄 – Hidden beauty of science" photography competition: *https://concoursyugen.jp.ambafrance.org/*



The "科学の幽玄 – Hidden beauty of science" competition organized by the French embassy in Japan promotes pictures from French/Japanese collaboration projects.

Rémi Delaporte-Mathurin, thesis prize from the Chancellerie des Universités de Paris

Rémi has been rewarded for his thesis work on the transport of hydrogen in tokamaks.

Rémi, whose thesis was in collaboration with CEA-IRFM and the Process and Materials Sciences laboratory (Université Sorbonne Paris Nord), received the Thesis Prize 2023 from the Chancellerie des Universités de Paris. His thesis, defended in October 2022, was on the "Transport of hydrogen in tokamaks: estimation of the retention of the ITER divertor and influence of the presence of helium."

Every year, the Chancellerie rewards the scientific, literary and artistic excellence of some forty young doctors from fifteen universities and six leading higher education institutions in the Ile de France area.

The winners are selected by a two-stage process: a first selection by the institutions themselves, then the final decision is made by one of the 18 juries specialized in each discipline for each award.

VIII Acronyms

Acronyms	Signification (FR)	Meaning (EN)
AE	Alfven Eigenmode	Alfven Eigenmode
AIA	Articulated Inspection Arm	Articulated Inspection Arm
ALARA	As Low As Reasonably Achievable	As Low As Reasonably Achievable
AMU	Aix-Marseille Université	Aix-Marseille University
ASEAN	Association des nations de l'Asie du Sud-Est	Association of South-East Asian Nations
ASIPP	Institute of Plasma Physics of the Chinese academy of sciences	Institute of Plasma Physics of the Chinese academy of sciences
AUB	Université américaine de Beyrouth	American University of Beirut
AUG	Asdex UpGrade	Asdex UpGrade
BED	Boucle d'Eau Décarbonaté	decarbonated water loop
BEST	Burning Experiment Superconducting Tokamak	Burning Experiment Superconducting Tokamak
CAS	Académie des Sciences chinoise	Chinese Academy of Sciences
CFETR		China Fusion Engineering Test Reactor
CFP	Composants Face au Plasma	Plasma-Facing Component
CJEM		CS Joints Environment Mockup
CNNC		China National Nuclear Corporation
CNRS	Centre National de la Recherche Scientifique	National Centre for Scientific Research
CNSTN	Centre National des Sciences et Technologies Nucléaires	
CORIA	COmplexe de Recherche Interprofessionnel en Aérothermochimie du CNRS - Université et INSA de Rouen	COmplexe de Recherche Interprofessionnel en Aérothermochimie du CNRS
CS	Solénoide Central	Central Solenoid
DA	Agences Domestiques	Domestic Agencies
DEMO	DEMOnstration (Power Plant)	DEMOnstration (Power Plant)
DES	Direction des EnergieS	Division of Energies
DOE	Département de l'Energie	Department of Energy
DPC	Département de Physico-chimie	Department of Physicochemistry
DRF	Direction de la Recherche Fondamentale	Fundamental Research Division
DTT	Divertor Tokamak Test facility	Divertor Tokamak Test facility
DTU		Technical University of Denmark
EAST	Experimental Advanced Superconducting Tokamak	Experimental Advanced Superconducting Tokamak
ECE		Electron Cyclotron Emission
ECEI		Electron Cyclotron Emission Imaging
ECRH	Electron Cyclotron Resonant Heating	Electron Cyclotron Resonant Heating
EDF	Electricité De France	French multinational electric utility company
EPS	European Physical Society	European Physical Society
ERM	Ecole Royale Militaire	Royal Military Academy

Acronyms	Signification (FR)	Meaning (EN)
F4E	Fusion for Energy	Fusion for Energy
FAIR	Findable Accessible Interoperable Reusable	Findable Accessible Interoperable Reusable
FBG	Fibres à réseau de BraGg	Fiber Bragg Grating
FEC		Fusion Energy Conference
FESTIM		Finite Element Simulation of Tritium In Materials
FR-FCM	Fédération de Recherche Fusion par Confinement Magnétique	Fédération de Recherche Fusion par Confinement Magnétique
GEM		Gas Electron Multiplier
HADES	High heAt LoaD tESt facility	High heAt LoaD tESt facility
HFS	Côté à fort champ	High Field Side
HL-2A		Huan-Liuqi-2A
HL-3		Huan-Liuqi-3
HPC	Calcul haute performance	High-Performance Computer
HTS		High Temperature Superconductor
ICRF		Ion Cyclotron Resonance Frequency
ICRH	Ion Cyclotron Resonance Heating	Ion Cyclotron Resonance Heating
IMAS	ITER integrated Modelling and Analysis Suite	ITER integrated Modelling and Analysis Suite
INSTN	Institut national des sciences et techniques nucléaires	French school for energy and health technology
IO	ITER Organization	ITER Organization
IPPLM		Institute of Plasma Physics and Laser Microfusion
IRESNE	Institut de REcherche sur les Systèmes Nucléaires pour la production d'Energie bas carbone	
ISFIN	Institut Sciences de la Fusion et de l'Instrumentation en Environnements Nucléaires	Institute for Fusion and Instrumentation Sciences in Nuclear Environments
ITER USDA	L'Agence Domestique ITER des Etats Unis d'Amérique	ITER US Domestic Agency
ITG		Ion Temperature Gradient
IUPAP	Union Internationale de Physique Pure et Appliquée	
JET	Joint European Torus	Joint European Torus
JIFS		JT-60SA International Fusion School
JT-60 SA	Japan Tokamak 60 Super Advanced	Japan Tokamak 60 Super Advanced
KAIST		Korean Advanced Institute of Science and Technology
KIT	Karlsruhe Institute of Technology	Karlsruhe Institute of Technology
LH	Lower Hybrid	Lower Hybrid
LIBS	Laser Induced Breakdown Spectroscopy	Laser Induced Breakdown Spectroscopy
LIST	Laboratoire d'Intégration des Systèmes et des Technologies	
LPP	Laboratoire de Physique des Plasmas	Laboratory for Plasma Physics
LSPM	Laboratoire des Sciences des Procédés et des Matériaux (Paris)	
MEMENTO	Metallic Melt Evolution in Next-step TOkamak	Metallic Melt Evolution in Next-step TOkamak
MENA		Middle East and North Africa
MHD	magnétohydrodynamique	Magneto-HydroDynamics
MIFI	Magnet Infrastructure Facilities for ITER	Magnet Infrastructure Facilities for ITER
MIT		Massachusetts Institute of Technology
MJT		Multi Jack Bolt Tensioners
MT		Magnet Technology
NBI	Neutral Beam Injection	Neutral Beam Injection
NTU	Nanyang Technological University	Nanyang Technological University
OIS		Outer Intercoil Structure
ORNL		Oak Ridge National Laboratory
PCS	Système de Contrôle des Plasmas	Plasma Control System
PFPO		Pre-Fusion Plasma Operation

Acronyms	*Signification (FR)*	*Meaning (EN)*
PFU	Plasma-Facing Unit	Plasma-Facing Unit
PLM		Product Lifecycle Management
POSTECH		Pohang University of Science and Technology
PPPL	Princeton Plasma Physics Laboratory	Princeton Plasma Physics Laboratory
RF	Radio Fréquence	Radio Frequency
SAFE		Singapore Alliance with France for Fusion Energy
SELFIE	SELf FIEld joints test facility	SELf FIEld joints test facility
SIFFER	SIno French Fusion Energy centeR	SIno French Fusion Energy centeR
SPA		Specific Project Agreement
SWIP		SouthWestern Institute of Physics
TBS	Test Blanket System	Test Blanket System
TCV	Tokamak à Configuration Variable	Variable Configuration Tokamak
TITAN	Test-bed for ITer ion cyclotron ANtenna	Test-bed for ITer ion cyclotron ANtenna
TRANSAT	TRANSversal Actions for Tritium	TRANSversal Actions for Tritium
TWA	Travelling Wave Array	Travelling Wave Array
UNIST		Ulsan National Institute Science and Technology
VUV		Visible Ultra-Violet
W7-X		Wendelstein 7-X - réacteur expérimental à fusion nucléaire de type stellarator, construit en Allemagne par l'Institut Max-Planck de physique des plasmas
WAVS	Wide Angle Viewing System	Wide Angle Viewing System
WEST	W (tungsten) Environment in Steady-state Tokamak	W (tungsten) Environment in Steady-state Tokamak
XICS	X-Ray Imaging Crystal Spectrometer	X-Ray Imaging Crystal Spectrometer
XR	eXtented Reality	eXtented Reality

Institut de Recherche sur la Fusion par confinement Magnétique

Direction de la Recherche Fondamentale
Centre de Cadarache
13108 Saint-Paul-Lez-Durance

irfm.cea.fr