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Editorial

New Challenges in Nuclear Fusion Reactors: From Data Analysis to Materials and Manufacturing

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Abstract: The construction and operation of the first generation of magnetically controlled nuclear fusion power plants require the development of proper physics and the engineering bases. The analysis of data, recently collected by the actual largest and most important tokamak in the world JET, that has successfully completed his second deuterium and tritium campaign in 2021 (DTE2) with a full ITER like wall main chamber, has provided an important consolidation of the ITER physics basis. Thermonuclear plasmas are highly nonlinear systems characterized by the need of numerous diagnostics to measure physical quantities to guide, through proper control schemes, external actuators. Both modelling and machine learning approaches are required to maximize the physical understanding of plasma dynamics and at the same time, engineering challenges have to be faced. Fusion experiments are indeed extremely hostile environments for plasma facing materials (PFM) and plasma-facing components (PFC), both in terms of neutron, thermal loads and mechanical stresses that the components have to face during either steady operation or off-normal events. Efforts are therefore spent by the community to reach the ultimate goal ahead: turning on the first nuclear fusion power plant, DEMO, by 2050. This editorial is dedicated at reviewing some aspects touched in recent studies developed in this dynamic, challenging project, collected by the special issue titled "New Challenges in Nuclear Fusion Reactors: From Data Analysis to Materials and Manufacturing".

Keywords: nuclear fusion; plasma physics; plasma diagnostic; plasma facing components; machine learning; neutronics

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1. Introduction

Fusion energy is expected to play an important role towards a sustainable system of energy production. Fusion does not produce greenhouse gases and its fuel is practically unlimited. Although the primary reaction does not produce radioactive materials, the components facing the plasma become radioactive due to the flux of fusion generated neutrons. However, with a proper choice of materials, radioactivity decays in a time interval of approximately a century [1], a much smaller time window than nuclear fission wastes. By providing a baseload electricity, fusion can be well integrated with renewable Energy Sources (RES) that have the limitation of being intermittent [2].

During the last fifty years, Magnetically Confined Nuclear Fusion Power Plants (MC-NFPs) have become the main focus of research. Nowadays, the largest magnetic confinement fusion experiment is JET [3], which has successfully concluded his second deuterium and tritium campaign in 2021 (DTE2), managing to sustain a high-power plasma for a full five second pulse by producing roughly three times more energy than in his first deuterium tritium experiment 1 (DTE1) in 1997 [4]. However, further steps have to be done towards the construction of a demonstration fusion power plant (DEMO). ITER [3] will demonstrate the scientific feasibility of fusion by achieving a fusion gain $Q = 10$ by producing fusion power 10 times larger than the power provided to the plasma by the external heating system. On a longer time scale DEMO [5] will produce net electricity to the grid of the order of hundreds MW [5]. To achieve the goals set for both ITER and DEMO, there are challenges to face, related also to not yet completely understood plasma mechanism. The improvements of diagnostics coupled to progresses in theoretical modelling, data driven and statistical approaches and methods, are key features toward such comprehension.

At the same time, engineering challenges are still to be tackled. A primary challenge is related to heat loads on the Plasma Facing Components (PFC) and to the handling of the exhaust plasma. Indeed, ITER will withstand severe steady-state thermal loads up to about 20 MW m^{-2} combined with transient ones up to GWm^{-2} due to edge-localized modes (ELMs) [6]. Even larger heat loads are expected in DEMO with the addition of effects such as the erosion of the plasma-facing material (PFM), the change in material properties due to neutron irradiation together with transmutation and activation. To face this challenge a dedicated facility, the Divertor Tokamak Test facility (DTT), is under construction to develop innovative solutions for the heat exhaust based on advanced magnetic configurations and new plasma facing materials [7]. In this framework, it has also to be mentioned the relevant contribution to the development of steady state and high performing ITER pulses provided by the EAST [8] KSTAR [9] and JT-60SA [10] devices.

The just illustrated background allows then to contextualise the main outcomes illustrated in the following section.

2. Overview of the Issue

One of the main issue affecting fusion experiments is related to the occurrence of disruptions [11] which are catastrophic events due to the loss of the plasma confinement in a very short time window of the order of milliseconds. Such events stresses the conductive walls, including the PFCs, with forces which magnitude grows non-linearly with the main plasma current [12]. At JET, considering a plasma current on axis of 3 MA, forces of about 3 MN, that are comparable to the weight of a F15, have been experienced. ITER is expected to operate at 15 MA, therefore, considering the non-linear dependence on the plasma current, unintentional disruptions could cause serious damages to the device. It is straightforward then to require that such occurrences have to be avoided, at least during high power discharges.

Since their initial occurrence, scientists began studying how to mitigate the impact of a disruption on the main vessel, to prevent damages. Nowadays, reliable algorithms monitoring in real time the discharge, are available for mitigation purposes [13]. The next natural objective is represented by avoidance, as well as on the need of developing algorithms capable of detecting different chain of events leading to a disruption, to trigger therefore external interventions in order to correct the plasma disruptive path and guarantee the safe continuation of the discharge. Such approach is still an open issue, but it is evidently a necessary step to be achieved for stable and safe operations [14].

The best results achieved on disruptions have been obtained by following the nowadays well established and quite popular approach based on machine learning (ML) techniques [13]. In Aymerich et al. [15] a systematic comparison of the most widespread and used algorithms is performed. A Multilayer Perceptron Neural Network (MLP-NN), a Generative Topographic Mapping (GTM) and a Convolutional Neural Networks (CNN) algorithm have been considered. Authors have demonstrated the capability of producing early warning times and, considering the MLP-NN and the CNN, a reduced number of false alarms. The GTM has been highlighted because of its capability of providing easier to interpret models w.r.t the other two, and it is shown how it can properly track a discharge from a not disruptive to a disruptive area.

Authors have also highlighted the importance of bolometric data in their analysis. It has also to be stated in fact that radiative events play a very important role, since they can influence the initial phases and the growth of MHD instabilities [16] which then can lead to a disruption.

What stated above then, stresses the importance of the work done by Mariano et al. [17], devoted to the acceleration of a Maximum Likelihood (ML) algorithm for bolometric tomographies in a ITER compatible environment. The ML algorithm has been implemented and validated on JET [18], and more recently on AUG [19] also. The code has the unique feature of providing the confidence intervals on its estimates and therefore, on the quantities that can be derived from the reconstructed emission patterns. Bolometric Tomography (BT) is an important tool in any fusion experiment indeed. Reliable estimates of specific quantities are used on a daily basis also as inputs of modelling codes such as TRANSP [20]. At the same time BT are relevant for power balance and transport studies, including impurity profiles estimates and impurity screening [21,22]. Mariano's et al. work have shown how their first released version of the accelerated algorithm, obtained using an ITER fast controller platform compatible with either Ubuntu 18.04 or the ITER Codac Core System distributions (6.1.2), is faster than a factor ten w.r.t the unoptimized MATLAB version, being therefore compatible for a JET intershot application. Improvements are expected to be achieved in a future release to reach the goal of 25 ms per iteration, compatible then for an ITER real-time implementation.

Disruptions are probably the most dangerous events preventing a safe and the steady state operation of a tokamak. However other designs of magnetically controlled fusion experiments exist, which might represent alternatives to the tokamak path. The most advanced competitor is the stellarator one, characterized by having field coils properly bended toroidally, shaping directly the magnetic field which confines the plasma. Such design allows avoiding the need of a toroidal current to induce a poloidal component as for tokamaks. Stellarators have a series of advantages, among which the most impressive one is indeed the almost absence of disruptions, due to the combination of the lack the toroidal current and the stabilizing effect of the magnetic shear on MHD instabilities, such as 2/1 and 3/1 modes [23]. However, stellarators have both a challenging engineering design, as well as physics issues. Considering the latter, the most important one is due to the fact that at high electron temperature, i.e., low collisionality, the neoclassical transport losses are much higher than the tokamak ones [23]. This is due to the fact that particles can be trapped in banana orbits in tokamaks, because of the axisymmetric magnetic field, while in stellarators particles are trapped in the so-called magnetic ripples, formed by the shape of the coils. Since the poloidal variation of their trajectories is small, such particles can be lost quite easily. However, W7-X has recently shown that it is possible to reduce such losses by working in a optimized neoclassical regime, reaching high temperature plasma conditions in the experimental campaigns conducted in 2017–2018 [24].

The recent developments in the stellarator community, stress the relevance of dedicated studies, such as the Murari et al. [25] article, devoted at deriving alternative scaling laws for extrapolating the confinement time of the new generation of devices. The purely data-driven approach described, makes use of the Symbolic Regression technique which allows scanning a wider range of mathematical functions to describe the relationship between a dependent quantity (the confinement time in this case) and not dependent ones (the engineering and physical quantities considered as relevant). Specifically, the authors have shown how the use of a renormalization factor that has been historically used for the extrapolation of the confinement time, is not strictly mandatory. Alternative scaling laws are competitive with the widely accepted one by the community, called ISS04 [26], both for the shear and for the shear-less devices.

Both tokamaks and stellarators are also affected by the so called Edge Localized Modes (ELMs) [27]. ELMs are periodic, radial events typical of the High confinement mode (H-mode), and are strictly connected to the pressure profile at the edge. Either peeling modes, or edge ballooning modes or coupled peeling-ballooning ones are the most rated ideal candidate MHD instabilities to originate ELMs [6]. ELMs cause large transient heats and particle loads on the PFCs. Extrapolating the energies associated to each ELMs to ITER [6], values up to twenty times higher than the benchmarked limits of considered materials are expected. Energy densities up to 11 MJ m^{-2} are indeed supposed to impact on the first wall.

Therefore, methodologies to either suppress, such as Resonant Magnetic Perturbations, or mitigate, such as ELMs pacing have been tested by scientists in the last decades. Among the latter, ELMs pacing techniques have been extensively tested also on AUG [28], JET [29], DIII-D [30] and EAST [31] is thought to be one of the main tools for ELMs mitigation on ITER.

Pacing would in fact reduce the heat and particle heats on PFCs wetted areas, lowering the loads to manageable levels, by triggering ELMs at the pellets pacing rate. The physics understanding behind the triggering of ELMs with pellets is still progressing, but simulations with JOREK [32] support the idea that high pressure and density plasmoids, generated by the injection of the frozen pellets at the edge of the plasma, lead to a helical perturbation centered on the field line of the injected pellet causing then an ELM.

Considering what stated above, it emerges how a proper estimate of the triggering efficiency and pacing characterization is fundamental.

The article written by Rossi et al. [33] focuses on such topics. It is devoted at estimating the efficiency of ELMs pacing by using a Spectrogram Cross-Correlation based algorithm combined to a k-means approach for classification. The developed methodology allows assessing which ELM is triggered by a pellet as well as which pellet actually triggered an ELM; the triggering efficiency in terms of percentage of triggered ELMs and in terms of triggering pellets as well as the triggering time delay and its distribution. The work has also shown how a relatively new type of diagnostic [34], based on a synthetic diamond sensor can be used to infer dynamical quantities related to ELMs. The resolution of such diagnostic allows in fact to derive a sharper triggered time distribution w.r.t the Be line signal that is used on a routine basis. Estimates using the diamond diagnostic are furthermore more in line with JOREK simulations of the expected triggering time [35].

The previous paragraphs have highlighted the importance of data analysis for both modelling and for diagnostic assessments and improvements, aimed at guaranteeing the operational safety, stability and feasibility of the next generation of devices.

Another key feature that has emerged and that is strictly connected with the previous one, is represented by the manufacturing and by the study of the thermomechanical properties, the interaction with the plasma with the neutron fluxes of both the plasma facing materials (PFM) and of the plasma facing components (PFC).

Thermally induced erosion of the (PFM) and the damage of the joints between the PFM and the heat sinks, are indeed to be considered for the integrity and operational safety of a device.

Extensive research campaigns aimed at studying the PFM interaction with the plasma have been held by different facilities [36–38] and by many laboratory experiments [39–48]. In such context, it is worth to recall the occurrence of the melting failure event during the plasma campaign in 2019 on the EAST tokamak [49–52].

Xiang Zan et al. [53] have studied, by metallography and SEM+EBSD technique, the microstructure of the upper divertor components of EAST, consisted of the CuCrZr cooling components, protected by tungsten armor, that have been damaged because of the just mentioned melting failure event occurred during the 2019 campaign. The recrystallization of the rolled tungsten components has been observed and the range of the recrystallization had been determined via heat flux distribution. Intergranular cracks in the recrystallized and in the rolled zones of tungsten monoblocs as well as interface deboning in Cu/CuCrZr interface of the cooling components were found.

Tungsten is actually the main candidate of the PFMs to be used in the locations of the device with the highest thermal and plasma particles loads, such as the divertor, to protect the cooling system components [54–59]. The joining of W with the heat sink materials is a challenging task due to the W refractory property. The plasma spraying (PS) of W on the heat sink materials with an appropriate interlayer is an established and recommended technique that shows good adhesion and thermomechanical proprieties [60–63]. The optimal plasma spraying process parameters, interlayer and substrate characteristics are the object of many research campaigns [64–68].

E. Pakhomova et al. studied the effect of the grain orientation on of the hardness gradient in the FGM interlayer between CuCrZr cooling component and W coating deposited by plasma spraying in Ar-H2 atmosphere. The preferred grain orientation as a result of nucleation and growth processes during the impact of droplets and solidification give rise to the observed hardness gradients [67].

The contribution of Dose et.al [69] is dedicated to such line of research as well, following a new approach. A thermomechanical engineered interlayer design, made with a functionally graded material, of a PFC is described in his work. The excellent mechanical properties of such solution are discussed and it is shown how the suggested solution can effectively reduce the thermal stresses due to the mismatch of the coefficient of thermal expansion between the armor and the heat sink material.

As stated above, the structural components of the future fusion devices will need to withstand high neutron fluxes and have to show a low radiation activation. Considering neutron loads then, Noce et al. [70] performed a detailed comparative neutronic analysis, regarding the nuclear load's spatial distributions on the Divertor PFCs targets of the EU-DEMO1-2017 design. Authors have assessed the main nuclear loads in terms of radiation damage, nuclear heating density and spatial distribution, He production and comparison in different types of materials (W-monoblock, Cu/CuCrZr and PFC-CB in EUROFER97). Such important contribution confirms even more the importance of material sciences for the design and realization of the next generation of devices.

Considering the low activation properties, the good weldability and mechanical performances in the range of temperature between 350 °C and 550 °C, the RAFM steels are widely considered as a candidate for nuclear fusion application [71,72].

To extend the upper limit above 550 °C, EUROFER97 steel can be reinforced by dispersion of oxide (ODS). The improvement of the mechanical performances of EUROFER97 without compromising its ductility by the refinement of microstructure is under study [73–79].

G. Stornelli et al. presented preliminary results of the microstructure and mechanical properties characterization of the cold-rolled EUROFER97 steel plates fabricated by cold rolling with different cold reduction ratios (CR) and heat treatment's temperatures. The effect of heat treatments on the mechanical proprieties of the samples deformed with the greater CR ratio (80%) is also shown. The EUROFER97 steel could be strengthened without its ductility compromising were evidenced by this work. To conclude, the most effective identification of the process parameters is under study [80].

3. Conclusions

This editorial has followed the thin thread connecting physics modelling, data analysis, diagnostic aspects and engineering issues of actual nuclear fusion experiments at a particular historical moment of transition to a new generation of devices. The topics covered in this issue are interdisciplinary, with a strong interpenetration of engineering and physics aspects. Fusion experiments are indeed multidisciplinary projects, were physical goals such as MHD instabilities and ELMy controlled scenarios, schemes for disruption prevention and avoidance, comprehension of the role of fast ions, impurities, and isotopes mixtures on the plasma stability, its confinement regime and on plasma transport, are examples of open issues to be addressed in the coming years.

Thermally induced erosion of the plasma-facing material (PFM), damages of the joints between the PFM and the heat sink are to be considered as well. Material irradiation with ions, impurities' particles and neutron fluxes are expected to induced degradation of the wall mechanical properties, transmutations, and activation. Therefore, the appropriate choice of the PFM, the design and the joining technique of PFC are indeed examples of the engineering challenges of nuclear fusion research as well. In conclusion, the editors hope this Special Issue to stand as a reference collection of valid projects and studies, and to contribute then to the development of the dynamic and challenging nuclear fusion community.

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