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European effort towards consolidating the design and manufacture of the ITER in-vessel components

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A B S T R A C T

The design of the in-vessel components represents one of the greatest technical challenges of the ITER machine. Europe has always played a very important role in supporting the design of the divertor and its high heat flux components, the first-wall the shield blanket and in developing and testing essential technologies and strengthening reliable manufacturing processes. This paper highlights the main research and development activities in progress in the European Fusion Programme in preparation of the start of ITER construction, to qualify all the technologies to be used, in particular, for the manufacture of the ITER plasma facing components and to demonstrate their compliance with the design requirements. It also describes some of the most outstanding problems that are still at issue in the design especially of the first-wall and that must be resolved to achieve the performance and machine availability goals and minimize the associated technical risks. The design and R&D priorities which are foreseen to achieve these aims are briefly mentioned.

1. Introduction

The ITER first-wall/shield blanket and the divertor have been designed in great detail in the past and a large part of the underlying research and development (R&D) work has been carried out in Europe [1,2]. The whole process has been long, arduous and resource intensive and the available technical solutions are presently being optimised for industrial production. Mock-ups and prototypes have been produced and have survived exposure to high heat flux and fatigue tests at levels exceeding in some cases the ITER design requirements. Reliable joints to bond the armours to the heat-sink materials and acceptance criteria for the manufactured components have been developed and demonstrated [3]. Neutron irradiation tests of the materials and joints have been carried out and the effect of irradiation on the performance of the components examined.

In spite of this remarkable progress, there remain a number of critical technical issues related to plasma facing components (PFCs) design and integration, which were recently analysed in the context of the ITER design review [4]. The outstanding problems include: (i) the uncertainties of the power loads and the demanding requirements on design and power handling, especially during transient events; (ii) the damage effects arising form Edge localised Modes (ELMs) and disruptions (e.g., material ablation and melting) and the influx of impurities into the plasma, which are only partially observed in existing devices due to the higher plasma power and energy content in ITER than in today's machines (hundreds MJ vs. few MJ); (iii) the need to strictly control erosion products and, in particular, tritium co-deposition that affects safety and machine availability; (iv) the unprecedented difficulty of maintaining the PFCs with remote handling (RH). These topics have been discussed at length in the past (see for example Refs. [5–7] and references therein).

The organisation of the paper is as follows: Section 2 reviews the main technological achievements with respect to the manufacturing and testing of divertor and first-wall/blanket mock-ups in view of forthcoming start of procurement. Section 3 discusses the outstanding technical issues that are still open. Finally, a summary is provided in Section 4.

2. Review of achievements

Europe has played a pivotal role in the past in supporting the ITER design process and in establishing reliable manufacturing processes, developing and testing essential technologies and complex in-vessel components, especially those that involve the divertor and the first-wall/ shield blanket (see for an overview Ref. [8]). Europe is strongly committed to the procurement of ITER PFCs and some of the most recent achievements in these areas are briefly discussed below.

2.1. Divertor

The procurement sharing of the ITER divertor assigns to the EU the responsibility of the manufacturing of the inner vertical target and cassette bodies and the integration of divertor PFCs and related

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diagnostics. In the past years Europe has made a significant contribution to the development of technologies for PFCs. These technologies have been validated mainly at Plansee (Austria) and Ansaldo Ricerche (Italy) by manufacturing a large number of mock-ups and prototypes. These components have been high heat flux tested in EU test facilities such as FE 200 (CEA-AREVA ANP) in France and in Judith at the Forschungszentrum Juelich (Germany) at heat fluxes well above the ITER requirements [9]. Carbon-Fibre-Composite (CFC) prototypes have been successfully tested up to 1000 cycles at 23 MW/m^2 (ITER design target 300 cycles at 20 MW/m^2) and tungsten prototypes up to 1000 cycles at 10 MW/m^2 (ITER design requirement for the divertor baffle is 3 MW/m^2). Recently, another medium-scale vertical target prototype (see Fig. 1) manufactured at ENEA Frascati by the 'hot radial pressing' technology has been high heat flux tested in the FE200 electron beam facility. The mock-up sustained up to 2000 cycles at 20 MW/m^2 and 15 MW/m^2 on CFC and W part, respectively, without failure. An experimental critical heat flux of 35 MW/m^2 was determined for the CFC monoblock part [8].

The reproducibility of the industrial processes for the CFC armour to heat sink joining is one of the issues that were addressed by the European R&D programme for the divertor. In fact, although the available manufacturing techniques allow for the fulfilment of the divertor requirements, the presence of randomly distributed defects at armour-heat sink interface cannot be excluded. The defects are also connected to the intrinsic features of the CFC composites such as orthotropy and porosity and to the real impossibility to guarantee constant properties among different material batches. Therefore, there is a potential risk that manufacturing defects can reduce the high heat flux performances. As a results of the manufacturing of several mock-ups and prototypes it is now clear that these defects cannot be fully eliminated. Therefore, nondestructive testing methods have been specialized for divertor PFCs and set up of suitable acceptance criteria has been undertaken. The activities on acceptance criteria are being carried out in collaboration with EU Associations (e.g., CEA Cadarache and CIEMAT) and in agreement with the ITER International Team. More than one hundreds mock-ups with calibrated artificial defects have been manufactured and underwent ultrasonic inspection and SATIR (*Station d'Acquisition et de Traitement InfraRouge*) infrared thermography [10] to both check the defect size and the detection capability of the two techniques. Then, the mock-ups have been high heat flux tested to know the minimum defect size able to limit the requested performances and subjected again to ultrasonic and infrared inspections to provide evidence of defect propagation. An analysis is ongoing to draw suitable acceptance criteria.

The manufacturing of the divertor qualification prototypes has started in line with the ITER procurement schedule. These prototypes include all the key features of the vertical targets and have CFC and W armoured parts. Two European companies have been

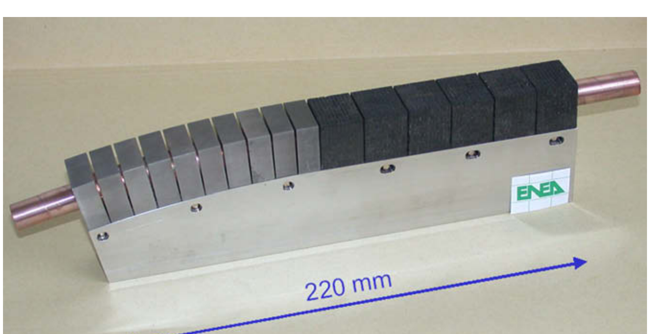

Fig. 1. ENEA vertical target prototype.

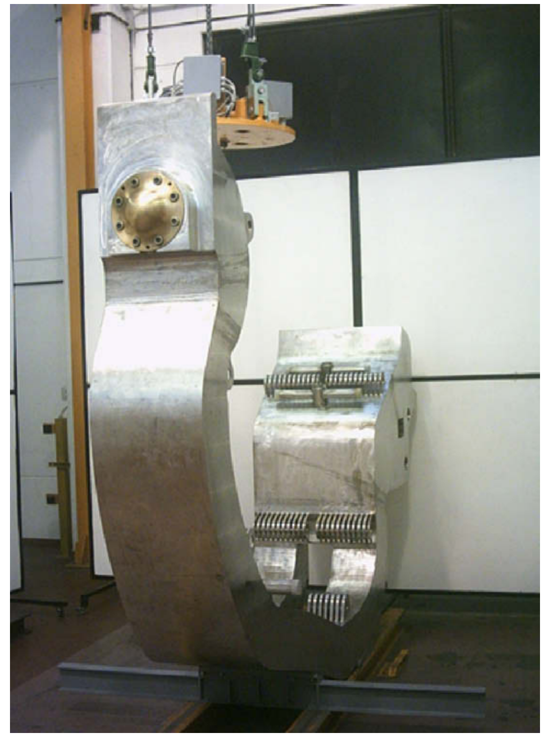

Fig. 2. Full size cassette prototype.

appointed to manufacture three qualification prototypes, namely one full monoblock and one mono-flat tile (Plansee); one full monoblock (Ansaldo). The prototypes have been manufactured by means of two different technologies: hot hydrostatic pressing (Plansee) and hot radial pressing (Ansaldo).

In addition, the manufacturing of a full-scale cassette prototype (Fig. 2) and a complete set of divertor PFCs with dummy armour (inner and outer target and dome) has been completed at Ansaldo Ricerche. The objectives of such activities were: (i) to verify that the cassette and PFCs could be manufactured within the specified tolerances; (ii) to validate the procedures to assemble the PFCs onto the cassette body including both the mechanical attachments and hydraulic connections; and (iii) to verify by testing the hydraulic design and to define proper procedures for draining and drying. The prototype manufacturing phase has been completed successfully and the integration of the PFCs onto the cassette body is in progress.

2.2. The ITER first-wall and the blanket-shield modules

The design of the first-wall (FW) panels of the blanket modules currently consist of a bi-metallic structure made of a 20-mm thick CuCrZr alloy heat sink layer bonded to a 40-mm thick 316L(N)-IG stainless-steel (SS) backing plate. A 10-mm thick beryllium (Be) layer is used as the protective plasma facing material and is bonded to the CuCrZr alloy layer in the form of tiles. An extensive development work programme performed in Europe has allowed to produce very good Be/CuCrZr alloy joints by HIPping [1]. Performances achieved with representative first-wall mock-ups exceed the present ITER design requirements (30000 cycles at 0.5 MW/m^2 peak heat flux plus transient events up to 1.4 MW/m^2). Detachment of Be tiles are observed at 3 MW/m^2. A neutron irradiation programme is still in progress to complete the full characterisation with irradiated mock-ups. Brazing has also been considered as an alternative technique, potentially cheaper, for joining Be tiles to the CuCrZr heat sink. A development work programme has been launched to develop a fast induction brazing technique to minimize the holding time at high temperature and consequently retain enough mechanical properties of the CuCrZr alloy. Induction brazing tests were done using the only appropriate silver free braze alloy presently available in the market, the STEMET 1108 procured from Russia. It was found that this braze alloy had poor wetting properties and the quality of the product was changeable. Difficulties were met for brazing Be tiles of dimensions representative of the Be tiles of FW panels. A few FW mock-ups were fabricated with inductively brazed Be tiles but showed thermal fatigue performance well below HIPped mock-ups, with detachment of Be tiles between 1.5 and 2 MW/m^2. This result has been considered not satisfactory. The development work on fast brazing equipment has been stopped in Europe and the effort is being concentrated on the development of a new silver free braze alloy.

The shield block of the blanket modules consists of a 316L(N) stainless-steel (SS) massive structure of typically 1.5 m long, 1 m high and 0.4 m thick. Conventional fabrication techniques were first considered for the manufacture of the shield block made from SS forgings. This fabrication route resulted in a very large number of welds, increasing the risk of water leakage inside the vacuum vessel during ITER operation, and in a lot of expensive machining and welding operations. An alternative design and fabrication route have also been developed to increase the reliability of the components e.g., by minimizing the number of the seal welds exposed to the vacuum. This alternative design is based on the experience gained during the ITER EDA by fabricating a shield blanket prototype by powder HIPping [2]. An advantage of this technique is the possibility to fabricate near net shape complex parts at lower cost by minimizing welding and machining operations. The difficulty however, is to predict the deformations due to the shrinkage of the powder during HIPping. The larger the amount of powder, the larger is the deformation. An easier approach has therefore been to reduce the amount of powder to a minimum, i.e., to use it only where it is beneficial such as for the complicated parts at the rear side of the modules. This also offers a greater design flexibility since the rear side, made from SS tubes imbedded into SS powder can easily be designed to accommodate the design requirements for any location in the blanket segmentation. It is particularly attractive for special modules with complex shape such as those next to the Neutral Beam Injector openings. Also HIPping powder on the top of water headers offers the additional advantage of having a double containment for the water coolant and therefore reducing the risk of water leakage inside the vacuum vessel. A detailed description of this fabrication route is presented in [11]. A full-scale shield block prototype representative of a standard module, HIPped at 1100 °C and 140 MPa for 4 h, have been completed and demonstrates the feasibility of the proposed shield concept. This shield prototype is shown in Fig. 3. Details of the development work and of the manufacture of this shield block prototype are presented in Ref. [12]. In parallel to this fabrication development work, material characterisation has continued. 316L(N)-IG SS powder material and joints with oxygen content lower than 100 wppm, HIPped under representative conditions have shown mechanical properties, including impact properties, similar or better than those of forged 316L(N)-IG SS.

The ITER blanket system will be procured by six parties (i.e., China, Europe, Japan, Korea, the Russian Federation and the USA). Some of them have limited experience on key technologies selected for the manufacture of the blanket-shield modules such as e.g., HIP or Be technologies. Therefore, a proper qualification programme is being implemented by the ITER Organisation (IO), starting with the first-wall, and each Participant Team involved in the procurement must demonstrate their capability to manufacture in time the due components with the required quality. The qualification is split into two stages: stage 1 aims at qualifying the technology proposed by each Participant Team through a test programme using mock-ups of dimensions 240 (L) $\times$ 80 (w) $\times$ 81 (H) mm. Several mock-ups have been manufactured by each party involved and is undergoing a test programme covering the specified ITER FW operation conditions [13]. The mock-ups are being tested in two test facilities located respectively in the NRI of Rez (Czech Republic) and in Sandia National Laboratories at Albuquerque in the United States. The detailed test conditions and acceptance criteria are still under discussion and should reflect the recommendations expressed during the ITER Design Review. The Participant Teams who will successfully pass this stage 1 shall undergo a stage 2, which will address the manufacture and testing of semi-prototype FW panels. Only those who will pass this stage 2 will be allowed to produce the FW for ITER.

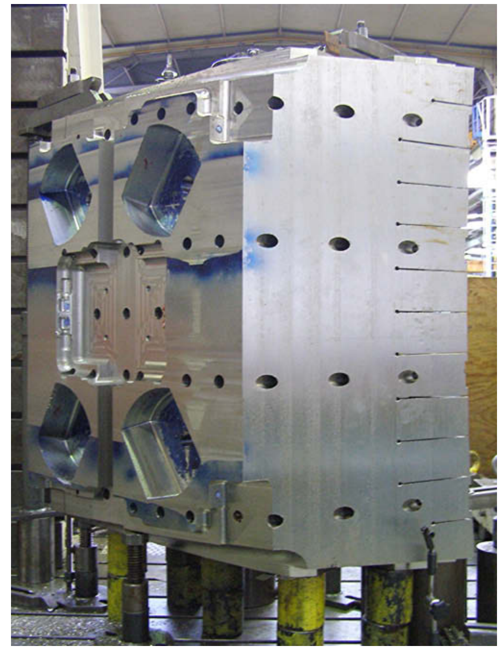

Fig. 3. Blanket-shield prototype.

3. Outstanding problems and uncertainties

The overall design of the ITER PFCs is the result of extensive work, both in the field of engineering, plasma edge physics and plasma-wall interactions (PWIs) and in the supporting technologies, by many teams, resulting in some case in well assessed and successful designs. However, the ITER Design Review [4] and the following EU assessment [14] have shed light on a number of weaknesses of the present design. They include:

- unprotected leading edges resulting form installation misalignments and inability to withstand modest levels of conducted power, especially in some areas (e.g., inner-wall and top of the machine;
- inability to survive without damage transient events (e.g., even mitigated disruptions could lead to melting Be surfaces);
- complex and partially undemonstrated feasibility of partial/total first-wall exchange; and
- divertor plasma material change-out strategy for various phase of operation.

3.1. Power handling and damage of first-wall components

Recent experimental evidence from divertor tokamaks indicates that significant plasma particle fluxes will impact on the ITER first-wall during steady-state operation and significant plasma energy and particle fluxes will reach the main wall during transients as well (e.g., ELMs [15,16]). The total power conducted to the wall during steady-state is expected to be not very large (~5 MW) and this could be taken in some areas like the inner-wall with some minor design optimisation (e.g., by simply minimizing misalignment and shaping to avoid leading edges). However, the upper X-point region need to be re-designed to withstand higher steady-state loads up to 5 MW/m^2 for standard single-null (upper X-point region) The problem is much serious for transient power depositions (e.g., ELMs, runaways, VDE, etc.), where the power is expected to be much higher but the duration of the event is typically very short. Similarly, slow (up to few seconds) plasma transients in ITER can cause plasma displacements so that the plasma separatrix can contact or closely approach the first-wall, leading to power loads of $\sim$100s MW m^{-2} for timescales of 1 s or longer, for instance following a sudden H–L transition. The duration of this contact phase depends on whether some of the poloidal field coils (including the central solenoid) are in current saturation or not. Such loads are near or beyond the limit of what can be sustained by a properly designed wall (field line angle of incidence of few degrees), particularly if the contact phase is several seconds in duration.

Thus, it is clear that the poloidal field coil system in ITER must adequately designed to minimize the length of these plasma-wall contact phases. Because of the potential for first-wall damage by VDEs, the ITER vertical stability system must be also designed to minimize their occurrence over a large range of plasma conditions and operating scenarios. Given that type 1 ELMs will be unacceptable in ITER, it is critical that reliable and robust ELM avoidance techniques are identified as soon as possible and implemented. Several design solutions are being investigated by the ITER Organisation, but for sake of space limitation they cannot be discussed here. A possible approach, similar to that used in today's tokamaks, would be that of protecting the main part of the wall with protruding limiter structures, which can localise the engineering problems giving better overall power handling performance and exchangeability. This would also make the performance and material choice for the recessed areas of the wall between the limiters much less critical. The problem of the start-up could be solved by the use of dedicated limiters. More work is clearly needed to confirm the attractiveness of each of these solutions.

3.2. Improve maintainability of first-wall components

A description of the maintenance systems of ITER is given at this conference by A. Tesini [17]. In ITER the unprecedented difficulty of remote maintenance of its components is due to: (i) the complexity and variety of the RH sequences, affecting in depth the design of many PFCs and interfaces; (ii) the hostile radioactive environment after the start of the deuterium phase with limited in-vessel viewing, inspection and cleaning capabilities, and manned intervention totally precluded; (iii) the scale and weight of components against the millimetric installation accuracy required; (iv) the fact that all PFCs and the vacuum vessel must be actively cooled (due to the long pulse duration – few 100 s in ITER vs. few seconds in most of existing machine) requiring remote cutting, re-welding, and inspection of water cooling conduits.

An aggressive development programme has been conducted by Europe in the area of divertor maintenance. Similar efforts would need to be devoted for the first-wall. Whilst some progress has been made in the past the main concerns include the complex and partially undemonstrated feasibility of first-wall module exchange and the cutting and re-welding of water pipes in areas of restricted access. A full review of the requirements and solutions for improving remote maintainability is underway in the ITER Organisation.

3.3. Risks arising from plasma facing material selection

A mix of different materials is currently proposed in ITER [7] to optimise the requirements of areas with different power and particle flux characteristics (i.e., Be for the first-wall, CFCs for the divertor strike point tiles and W elsewhere in the divertor). However, it is well known that this option suffers from recognised shortcomings (e.g., limited power handling and susceptibility of melting of Be, uptake of large amount of tritium by C, and the formation of material mixtures, whose behaviour remain uncertain) and sufficient engineering margins and flexibility must be available together with design and operational provisions to recover from undesired situations.

It must be noted that to date, no fusion device uses the same material mixture as foreseen for ITER and only some limited experience exists for the simultaneous use of different materials. The results of experiments from ASDEX Upgrade, which has became a fully clad with W [18], and those planned in JET which will install a Be wall and W divertor [19] should indeed help answer some key questions including the control of ELMs and disruptions, the magnitudes of erosion and tritium co-deposition, dust formation in the vessel, the ease of tritium removal from mixed-materials, as well as operational aspects (e.g., of using beryllium on the first-wall and of operating a full W machine).

The choice of the wall material for the first phase of operation is closely linked to the exchangeability of the first-wall. In fact, given the present decision to begin the physics phase with a low Z wall, the feasibility of exchanging the first-wall is mandatory, both to exchange damaged modules and, because to fulfill the goal of preparing the way for a demonstration fusion reactor, ITER will have probably to operate later on with a high Z wall.

Concerning the material to be used at the divertor target plates, a change from carbon to tungsten tiles is foreseen in the deuterium phase, and at the latest before start of DT operation. This is being debated and if the decision can be made to start operation with tungsten tiles, this would avoid programmatic delays that would arise from the lengthy shutdown to change to tungsten, the difficulties in cleaning up the carbon dust and flakes and the need to re-qualify the operating scenarios with tungsten.

4. Concluding remarks and further work

This paper discusses the progress made in Europe, as the major contributor to ITER, and under the technical coordination of EFDA before and now of Fusion for Energy, to continue to strongly support procurement of PFCs where the EU has a strong stake. Work has continued to consolidate the manufacturing R&D in some critical areas and especially to determine the most technically and cost effective fabrication methods.

Despite remarkable advances achieved in the past there are still several areas that require further urgent work and that could otherwise adversely impact the achievement of scientific and programmatic goals. To this extent, the engineering design of the first-wall is being optimised and design changes are foreseen but it is expected that most of the key manufacturing technologies, which have been developed during the last ten years would remain valid. In particular, RH of the first-wall is considered to be one of the areas of highest risk, and reliable maintainability must be demonstrated through R&D and proper test facilities. In common with other complex engineering projects, the final choice should be based on a technical assessment and management of the associated risks and on avoiding show-stoppers.

It is clear that the EU should make its best effort to ensure that all the risks associated with the issues associated with PFC design described in this paper are not overlooked and that they are eventually solved. In this respect, the recent creation of the EU domestic agency in Barcelona, Fusion for Energy, is enabling a new framework that will support the ITER Organisation more effectively.

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