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Status of the EU R&D programme on the blanket-shield modules for ITER

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ABSTRACT

A research and development (R&D) programme for the ITER blanket-shield modules has been implemented in Europe to provide input for the design and the manufacture of the full-scale production components. It involves in particular the fabrication and testing of mock-ups and full-scale prototypes of shield blocks and first wall (FW) panels. This paper summarises the main achievements obtained so far and presents the latest results of this R&D programme. In particular, it reports the status of the shield fabrication development programme with the manufacture of a full-scale shield prototype. It also reports the latest results of high heat flux and thermal fatigue tests of FW mock-ups. It describes the preparation of irradiation experiments of Be coated FW mock-ups. Finally, it presents the outline of a possible qualification programme that each contributing participant teams should pass prior to the procurement of the blanket-shield modules for ITER.

1. Introduction

The ITER blanket-shield concept is a modular configuration mechanically attached to the vacuum vessel. The modules consist of a water-cooled 316L(N)-IG stainless steel (SS) shield block and separable first wall (FW) panels mechanically attached to the shield Block. Two design options have been studied for the attachment system of the FW panels to the shield block. For design option A, developed by the European participant team (EU PT), the FW panels are directly attached to the front surface of the shield with high strength bolts as shown in Fig. 1. For design option B, the FW panels are welded at the rear side of the shield by means of a leg going through the module. Except for this difference, all the interfaces between the blanket-shield modules and the vacuum vessel and between the modules and the remote handling systems have been kept the same. A research and development (R&D) programme for the blanket-shield modules option A has been implemented in the EU to provide input for the design and the manufacture of the full-scale production components. It involves in particular the fabrication and testing of mock-ups and full-scale prototypes of shield blocks and FW panels. This paper summarises the main achievements obtained so far and presents the latest results of this R&D programme.

2. Description of blanket-shield modules

The ITER blanket is segmented into 440 blanket-shield modules, mechanically attached to the vacuum vessel by four flexible radial supports. A detailed description of the modules is reported in Ref. [1]. Typical dimensions of these modules are 1.5 m × 1 m × 0.5 m and their weight does not exceed 4.5 ton. Depending upon the poloidal location, the modules have either 4 or 6 FW panels, with typical dimensions of 1 m × 0.25 m × 0.07 m. They consist of a bi-metallic structure made from a 20-mm thick precipitation hardened copper chromium zirconium (CuCrZr) alloy heat sink material imbedded with 10/12-mm diameter 316L(N)-IG SS tubes.

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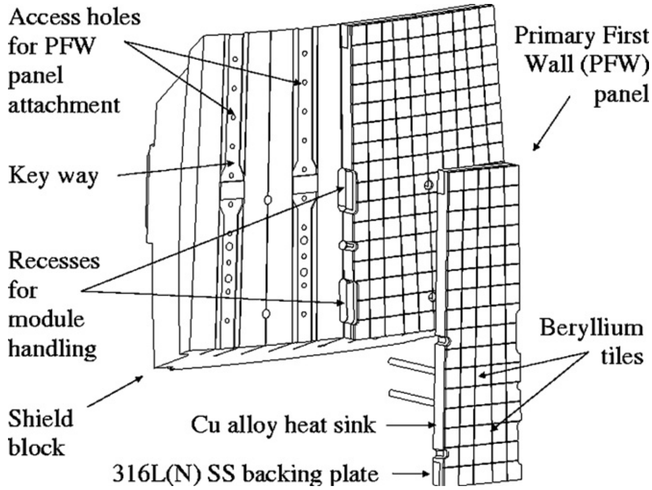

Fig. 1. Shield module (design option A).

The Cu alloy layer is bonded to a 40-mm thick 316L(N)-IG SS backing plate having 12-mm diameter cooling channels. A 10-mm thick Beryllium (Be) layer is used as the plasma facing material and is bonded to the Cu alloy layer in the form of tiles. The FW panels are mechanically attached to the front surface of the shield block using a central key (Fig. 1) inserted in the corresponding key way of the shield block and a poloidal row of studs made from high strength PH13-8 Mo martensitic steel. The FW panels are designed to sustain a peak surface heat flux of 0.5 MW/m^2, a maximum neutron wall load of 0.8 MW/m^2 and an average neutron fluence of at least 0.3 MWa/m^2 (about 3 dpa in steel) for a nominal number of 30,000 cycles. According to the present ITER design requirements, they shall also sustain a limited number of 10 s transient events of up to 1.4 MW/m^2.

3. Shield blocks

From the two fabrication routes considered initially by the EU PT for the manufacture of shield blocks, namely the welded and the hot isostatic pressed (HIPped) methods [1], the latter has been selected as the reference one. The proposed fabrication route is based on the experience gained during the ITER EDA with the manufacture of a shield prototype by powder HIPping [2]. An advantage of this technique is the possibility to fabricate near net shape complex parts at lower cost by minimising 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 shield block. 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 a complex shape, such as those next to the neutral beam openings. Also, HIPping 316L(N)-IG SS powder on the top of water headers offers the additional advantage of having a double containment of the water coolant and therefore of reducing to a minimum the number of seal welds exposed to the vacuum. A detailed description of this fabrication route is presented in Ref. [1]. A full-scale shield block prototype representative of a standard module (no. 11a), HIPped at 1100 °C and 140 MPa for 4 h, is almost completed and demonstrates the feasibility of the proposed shield concept. A picture of this shield prototype is shown in

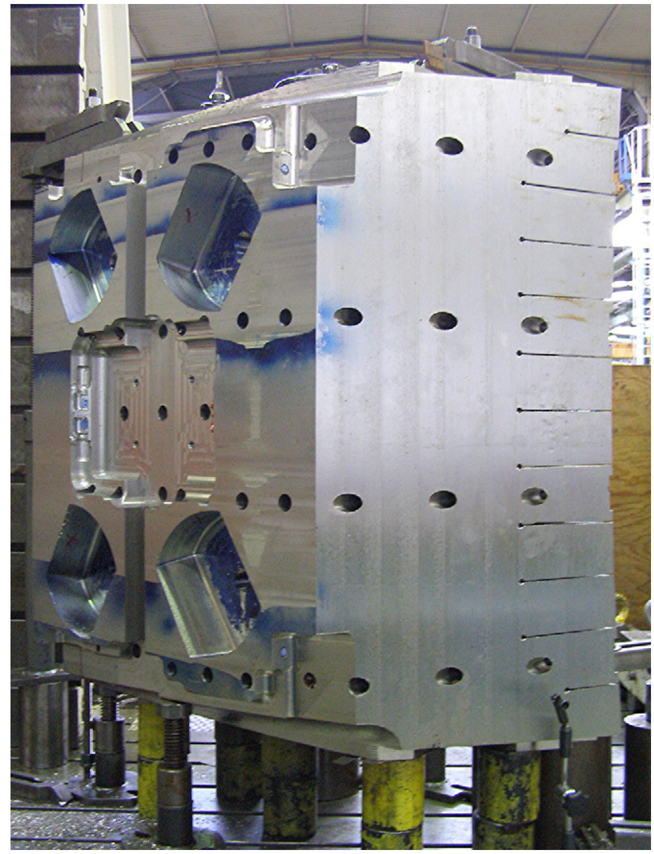

Fig. 2. Full-scale shield block prototype HIPped at 1100 °C and 140 MPa for 4 h (AREVA NP).

Fig. 2. Details of the development work and of the manufacture of this shield block prototype are presented in Ref. [3]. 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 shield representative conditions have shown mechanical properties, including impact properties, similar or better than those of forged 316L(N)-IG SS.

4. First wall panels

4.1. Manufacture of the base structure of FW panels

Two joining techniques are still in competition for the manufacture of the CuCrZr/316L(N) SS bi-metallic structure: solid HIPping and powder HIPping.

For the solid HIP fabrication route, the FW panels are assembled from solid parts. The 316L(N)-IG SS plates, the CuCrZr alloy plates and the 316L(N)-IG SS tubes are joined together with one single HIP cycle at 1040 °C and 140 MPa for 2 h. In order to retain acceptable mechanical properties for the CuCrZr alloy, cooling rates above 50 °C/min must be achieved [4]. To avoid performing HIP cycles only in facilities equipped with fast cooling rate equipment (HIP quenching), the EU has investigated the effect of post HIP solution annealing heat treatments performed at several temperatures ranging from 980 to 1040 °C. Results have shown a slight degradation of the high heat flux (HHF) performances for CuCrZr/316L SS mock-ups HIPped at 1040 °C and then solution annealed at 980 °C or 1040 °C for 1/2 h. They achieved 1000 cycles at 5 MW/m^2 plus respectively 267 and 76 cycles at 7 MW/m^2 in the e-beam test facility FE200 of AREVA at Le Creusot (F). This result may be