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R&D ON W FIRST WALL FOR ITER AND FUTURE FUSION REACTORS

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Abstract

The first wall (FW) is a critical component of fusion reactor blankets, responsible for providing 100% thermal shielding at a heat flux level of 1~2 MW/m² and forming a compatible interface with burning plasma. The research and development on FW have received extensive attention worldwide in several decades. Given the rigorous operational demands, tungsten (W) has emerged as the preferred plasma-facing material (PFM) for FW armor due to its high melting point and low tritium retention, while structural materials like reduced-activation ferritic/martensitic (RAFM) steel, CuCrZr (for ITER) and V4Cr4Ti alloy are selected based on their comprehensive mechanical properties, heat-sink capabilities and irradiation resistance. This paper summarizes the R&D progress on W-armored FW for ITER and future fusion reactors (e.g., CFETR, DEMO), with a focus on contributions from China and Russia. For ITER, China undertakes the manufacture of 10% of ITER FW panels and participates in the ITER Test Blanket Module (TBM) program. The advanced R&D on W-armored FW—developing a one-step HIP process to bond W tiles to CuCrZr heat sinks, demonstrating component durability under 4.7 MW/m² heat load for 15,000 cycles. A 20-finger W-armored FW semi-prototype has been successfully assembled. Russia has qualified fast brazing technology for ITER divertor dome plasma-facing units (PFUs)—a design applicable to ITER W-armored FW—though challenges remain for 12 mm-thick W tiles to withstand 4.7 MW/m² for 30,000 cycles. For CFETR's Water-Coolant Ceramic Breeder (WCCB) blanket, China's ASIPP developed a box-structured W-armored FW which producd the world's largest W-armored FW (1.2×0.95×0.95 m³) with no adverse effects on W/RAFM bonding. Furthermore, a helium-cooled FW configuration that combines RAFM steel and V4Cr4Ti alloy was developed by SWIP to mitigate the coolant-induced corrosion of V4Cr4Ti. HIP-manufactured mock-ups of this design demonstrated durability through 1500 equivalent thermal cycles at a heat flux of 1 MW/m², while TiC layers formed in-situ at the bonding interface were found to lower deuterium permeability by over three orders of magnitude—enhancing the structure's tritium retention capability. Additionally, China investigated FeCrAl cladding on RAFM steel as a tritium permeation barrier (TPB), achieving successful HIP bonding of Fe-Cr-Al/CLF-1 steel, though hydrogen isotope permeation properties require further study. These advancements validate the feasibility of W-armored FW designs and manufacturing processes for ITER and future fusion reactors.

1. INTRODUCTION

First wall (FW) is to provide 100% thermal shielding to blanket at 1~2 MW/m^2 level and to form a friendly interface with burning plasma in future fusion reactors. To reach permanently to acceptable performances under intense 14 MeV neutron irradiation and high heat flux, it is essential to develop a feasible FW structure to remove the heat with reliable materials bonding together. W, a promising plasma-facing material (PFM) with the highest melting-point and low tritium retention, have been decided to replace beryllium as armor material for ITER FW as well. Reduced-activation ferritic/martensitic (RAFM) steel will be used as structural material for near-term DEMO reactor and V-base alloys for mid-term application relying on their strong neutron irradiation resistance and high-temperature strength, while ITER FW has to use CuCrZr as heat-sink material due to its much higher heat flux up to 4.7 MW/m^2 at plasma limiter operation phase. fter two-decades investigations by qualification of various material bonding technologies for ITER FW, the production process qualification has been completed by manufacturing full-scale prototypes (FSP) in China, EU and Russia. Recently R&Ds focused on enhanced heat flux (EHF) tungsten FW, mock-ups and full-size fingers have been made with reachable thermal fatigue performances. A semi-prototype with 20 fingers was assembled on a full-scale central beam - hich was machined from a 316L(N)-IG forging through deep drilling, cover welding, and pipe connection. Tolerable gaps and height deviations were designed between the fingers to mitigate magnetic forces and avoid sharp leading edges, respectively.

As a candidate blanket for CFETR, the Water-Coolant Ceramic Breeder (WCCB) blanket - proposed and designed by the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP) - adopts a box structure, with RAFM steel as its main structural material and a 2 mm-thick tungsten (W) armor on its first wall (FW); for the FW's fabrication, a one-step hot isostatic pressing (HIP) process was developed, which enables both the bonding of W to RAFM steel and the formation of inner flow channels for small-scale FW mockups. Due to China's largest HIP furnace's limited loading space, only 1/9 - scale FW parts could be produced per cycle, so the full-scale FW preparation route was defined as fabricating nine 1/9 - scale parts via HIP and then joining them into a full-scale component. Besides, FW technologies investigation for future reactors covers helium- and water-cooling RAFMs and V4Cr4Ti ones for various blankets. This work aims to address material issues under severe neutron irradiation and to provide strong tritium barrier by specific-designed and in-situ formed material-bonding interface. The deuterium permeation rate could be reduced by 2 ~ 5 orders at elevated temperatures. For tritium permeation barrier, cladding the RAFM structure with a FeCrAl layer was also investigated. However, the hydrogen isotopes' permeation properties have not been researched yet.

2. DEVELOPMENT OF W-ARMORED ITER FW

ITER enhance heat flux (EHF) FW utilizes a hypervapotron (HVT) heat sink to remove the surface heat. To withstand ultra-strong thermal shock from runaway electrons, the armored W tiles are in thickness of 6~12 mm for various ITER FW panels [1], which provide strong neutron shielding as well. However, the W armor need to be much thinner as 1~2mm for future reactor in order to have more neutrons into the breeding zones of a blanket to produce more tritium for self-sustaining.

China would not only manufacture and supply 10% of ITER FW panels, but also join the ITER TBM (test blanket module) program to develop helium-cooling ceramic blanket for future DEMO. A full-scale prototype (FSP) of Be-armoring EHF FW has been manufactured for ITER, via a armor bonding to CrCrZr heat-sink and CuCrZr heat-sink to 316L(N)-IG water-box back plate by hot-isostatic-pressing (HIP) route, and successfully passed all of the acceptance tests [2]. W-armored ITER FW has the same structural and heat-sink materials as the FSP. R&Ds activities on the W tiles bonding on CuCrZr heat sink, technologies for mock-ups and fingers manufacturing have been started since last year, employing the same HIP diffusion-boning route, with an exception of 0.5mm thicker oxygen-free interlayer being made by vacuum casting on the W tiles, instead of previous PVD coating plus a 0.5mm foil before HIP joining. The capability of the components to withstand 4.7MW/m^2 heat load for 15,000 cycles has been demonstrated.The high heat flux test results of a mockup with tungsten size of 16mm×16mm× 10mm are shown in FIG. 1. It can be found after 5.9MW/m^2 8,868 cycles HHFT (equivalent to 4.7MW/m^2 44340 cycles) there is no overheating zone on the tungsten surface. From the UT results, it demonstrates the bonding interface of both W/OFC and OFC/CuCrZr are very good without large disbanding area after HHFT.

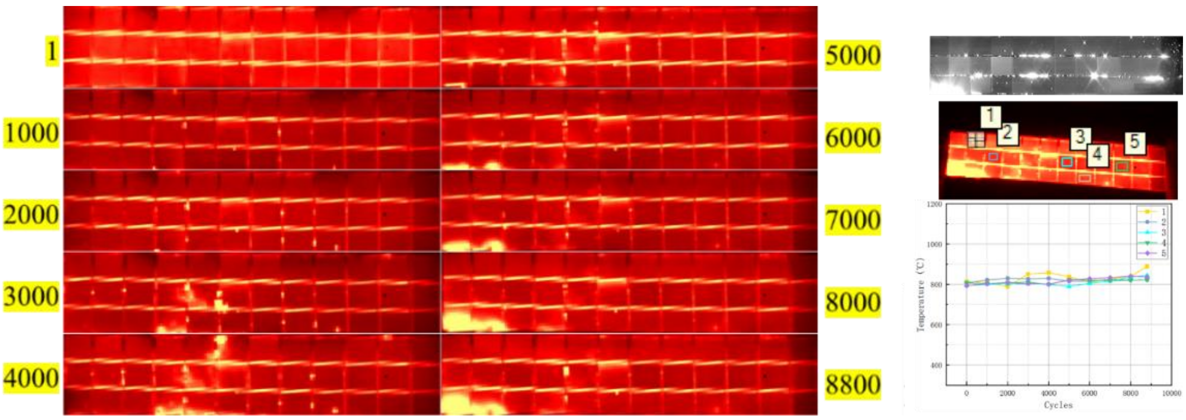

FIG. 1 The temperature distribution during HHFT and ultrasonic testing (UT) results after HHFT

A semi-prototype with 20 full-scal W-armoring fingers mounted on a 316L(N) central beam has been completed with improved welding technologies to connect thin-wall water pipes . The success rate increased from 50% to 90% in pipe connection for fingers pairing and for the pairs to the central beam assembly, by single-path welding of 1.75mm thick 316L tubes in narrow space. After manufacturing, the final factory acceptance test (FAT) includes visual testing, non-blocking test of the cooling channel, hydraulic pressure test, ultrasonic testing, hot helium leak tests and dimensional examination, etc. For non-blocking test, hot water (up to 85 °C) was circulated in the fingers with their surface viewed by an infrared (IR) camera in order to identify uneven temperature-rising to localize the presence of blocked or partially blocked coolant paths. Results are showed in FIG. 2, the panel showed uniform temperature distribution, indicating no blockage of the cooling circuit. At the flow rate of 2.24 kg/s, the temperature of thepanel surface reached stability in 45s, however 70s at the flow rate of 1.20 kg/s, which indicates that the faster flow rate is, the shorter time to get stability.

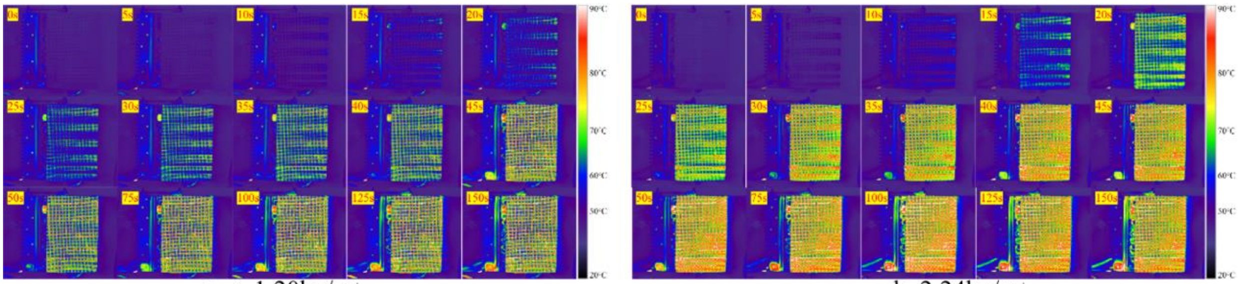

FIG. 2 IR temperature image during hot water test.

However, the heat-sink to withstand 30,000 cycles for a full-operation life is still a challenge under the test condition, needs to be further addressed. Instead of a 4.7MW/m^2 peak surface heat-load at locally small area of the finger surface in actual operation, the high-heat-flux (HHF) thermal cycling test was performed with the peak load unformly appling on a 50×200 mm region, which causes much higher thermal stress in the side-wall of the HVT cooling-channel and resulted in high risk of water leak at the CuCrZr/316L(N) bonding interface before reaching the life time number of cycles. Either heat load profile or the finger design should be improved for a solution.

Efremov institute in Russia has fully developed the technologies for ITER divertor dome plasma-facing unit (PFU) by fast brazing, which was qualified the capability to withstand 5 MW/m^2 heat flux for 1000 cycles [3]. The PFU has almost the same structure and geometry with the ITER W-armoring FW, including the HVT cooling channels. Consequently the technologies are applicable for manufacturing the ITER FW. As the W tile thichness for some of the FW around the top X strike point increases to 12mm, it will be a big challenge for them to handle the 4.7 MW/m^2 for much longer lifetime of 30000 cycles.

For China ITER TBM and other blanket, the technologies to make W-armoring RFAM steel FW mock-ups with sound bonding have been investigated [4]. A soft pure iron layer (0.5~1mm) was taken into the joint as a stress-complaint layer between the two material to match the different thermal expansion coefficient. Based on the TBM design, mock-ups with internally rectangular cooling channel have been made from thick CLF-1 steel plate and 3 mm thick pure W tiles by diffusion bonding. The mock-ups survived from a repeated 1 MW/m^2 surface heat loading for 1000 cycles under active water cooling at 70 ºC, indicative of reliable bonding between the materials. Before the test, the W/RAFMs joint had shown a RT bonding strength in the level of 250 MPa by shear test, despite of an 1μm thick FeW intermetallic compund layer formed by atomic reaction during HIP bonding at 900 ºC under 150 MPa pressure. The strength could be kept even after a post-HIP normalizing at 980 ºC and a tempering at 760 ºC to recovery the microstructure of the RAFM steel to an mormal tempering mantensitic.

3. FW R&D PROGRESS FOR WCCB BLANKET IN CHINA

As a blanket candidate for CFETR, the Water-Coolant Ceramic Breeder (WCCB) blanket has been proposed and designed in the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). Considering the manufacturing, installation and maintenance, the WCCB blanket is designed as a box structure, with RAFM steel selected as the main structural material. As the first wall (FW) of the blanket is located at the very front and directly faces the high-temperature plasma, a 2 mm thick tungsten (W) armor is designed on the surface of the RAFM steel to prevent plasma erosion. In present design of the WCCB blanket module, the area with tungsten is greater than 1 m^2.

In the manufacturing development, hot isostatic pressing (HIP) process has been used to bond tungsten/steel and study the preparation process of the FW small-scale mockups with tungsten tiles. A one-step HIP process has been proposed, which means that the W/RAFM bonding and the forming of the inner flow channel of the U-shaped FW mockup could be achieved in a single HIP cycle. Using this process, the small-scale FW mockup has been successfully prepared. The tests show that the performance of the mockup meets the technical requirements, which proves the feasibility of the process. However, due to the limited size of the HIP furnace, for example, the workpiece loading space of the largest HIP equipment in China is a cylinder with a diameter of 1.25 m and a height of 3 m, only one-ninth of the FW part could be prepared by this process. Therefore, the preparation route of the full-scale FW assembly is determined as follows: first, nine one-ninth FW parts were prepared by HIP, and then nine parts were welded together by Electron-Beam Welding (EBW) technology to assemble a full-scale FW component, as shown in Figure 3. To prove the feasibility of this route, we carried out the R&D activities for the electron beam welding of the FW mockups with tungsten tiles.

In the preparation of the one-ninth FW parts, three workpieces should be bonded together through a one-step HIP process. To achieve this goal, we have designed special welding structures and welding jigs. In the following, we have successfully completed the manufacturing of 9 one-ninth FW parts. At the same time, the processing test of EBW shows that when the penetration depth of EBW is 20 mm, it has no influence on the tungsten/steel bonding near the weld seam of 2 mm. On this basis, all nine FW parts were piece-welded into a full-scale FW component through the design and application of welding tooling. Its overall size is 1.2×0.95×0.95 m^3, which is the largest FW component with W armor at present, as shown in FIG. 3.

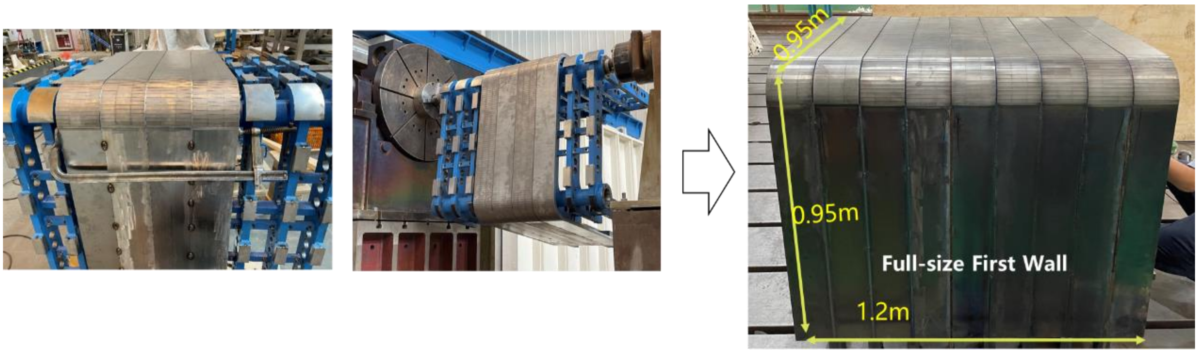

FIG. 3 EB welding of the full-scale FW

4. FW DESIGN AND R&D PROGRESS FOR FUTURE

The neutron irradiation dose in the FW for ITER is merely ~3 dpa but will be 1~2 orders higher in DEMO reactor and fusion power plant, where copper alloy and austenitic stainless steel (ASS) will be excluded because of their high activation issues. Consequently, future FW for DEMO will be a structure with W armoring on low activation structural materials, such as RAFM steel [5] and V4Cr4Ti alloy. RAFM steels, typically 9Cr-2WTaV, are assumed endurable for 20 dpa fusion neutron irradiation, which is not enough for DEMO. V4Cr4Ti could provide stronger irradiation resistance, but cannot be used in He-cooling and water-cooling conditions because of sever corrosion in the coolants. To overcome this issue, a helium-cooling FW integrated with both materials was designed by SWIP with rectangular channel, to prevent vanadium alloy from contact with the coolant, as shown in FIG. 4 (a) and (b). Such helium-cooling FW with V4Cr4Ti alloy as structural materials could provide inherent tolerance to high dose neutron irradiation and high-temperature operation capability up to 700 °C. By cladding a thin RAFMs layer facing the coolant, issue of oxidation could be solved. According to this helium-cooling design, the W-armoured FW mock-up has been prepared by diffusion bonding under HIP , as shown in FIG. 4 (c).

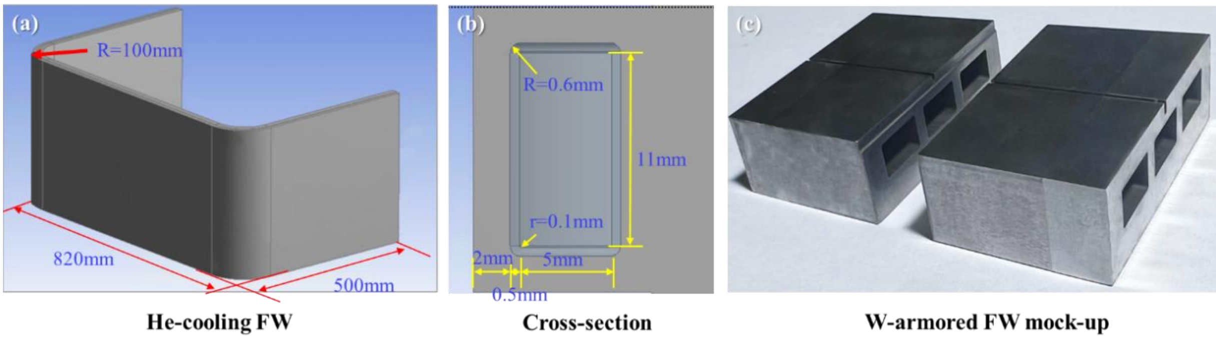

FIG. 4 The He-cooling FW layout, its cross-section showing the materials and cooling channels and the mock-ups

To verify the design indicators of the mock-up and the joining performance of different heterogeneous materials, it is necessary to further conduct high heat flux tests (HHFT) on the mock-up. Since the current existing high heat flux experimental equipment cannot carry out the helium-cooling tests under actual operating conditions, a water-cooling first-wall mock-up for HHFT was designed based on the existing water loop experimental conditions. To achieve the goal of simulating the maximum operating temperature of vanadium alloy under equivalent helium-cooled working conditions, the distance between the upper wall of the cooling channel to tungsten tile was increased in order to ensure the upper surface temperature of the vanadium alloy could be over 700 °C under a steady-state heat flux of 1 MW/m² (see FIG. 5 (a)). The HHFT FW mock-up, with tungsten as armored tiles (with size of 30 mm × 16 mm × 2.5 mm) and vanadium/steel as bimetallic structure materials, was successfully prepared by the hot isostatic pressing (HIP) process under the conditions of 900 °C, 2 hours, and 150 MPa (see FIG. 5 (b) and (c)).

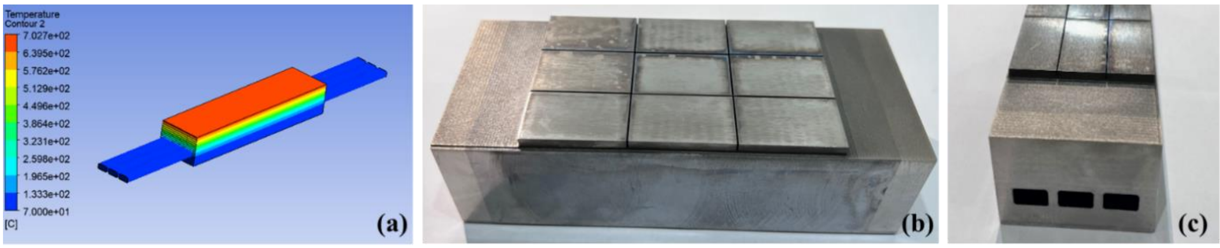

FIG. 5 Temperature simulation and fabricated prototype of the water-cooled mock-up for HHFT

During 500 cycles of high heat flux tests at a power density of 1 MW/m² and 200 cycles of accelerated tests at 1.25 MW/m² (equivalent to a total of 1500 cycles at 1 MW/m² as referring to ITER SDC-IC), the surface temperature of the tungsten armor on the mock-up transitioned uniformly without local overheating (as shown in FIG. 6). After the tests, there was no damage on the surface of the tungsten armor material, and no ultrasonic defects were detected at each bonding interface which can meet the acceptance criteria [6] for high heat flux performance of first walls in ITER. It shows the mock-up have a heat removal capability more than 1 MW/m² for 1500 cycles.

Furthermore, the structure (as shown in FIG. 4) could provide strong tritium permeation barrier. The deuterium permeability of RAFM steel/V4Cr4Ti bonding joints was tested by gas drive permeation device [7] in Southwestern Institute of Physics, at various temperatures in the range of 400~600 °C at pressure of 100 kPa. FIG. 7 shows the deuterium permeability curves of the vanadium alloy (V4Cr4Ti-850) and the RAFM steel/ V4Cr4Ti bimetallic joint samples. It can be seen that the deuterium permeability of the bimetallic samples is reduced by more than three orders of magnitude compared with that of the vanadium alloy material. It could be mainly attributed to the dispersedly distributed TiC carbide layer (see FIG. 8 (b) and (c)), formed between well diffusion-bonded interface of CLF-1 RAFM steel and V4Cr4Ti alloy, which assumed to be able to suppress the diffusion of hydrogen isotopes across the bonding interface.

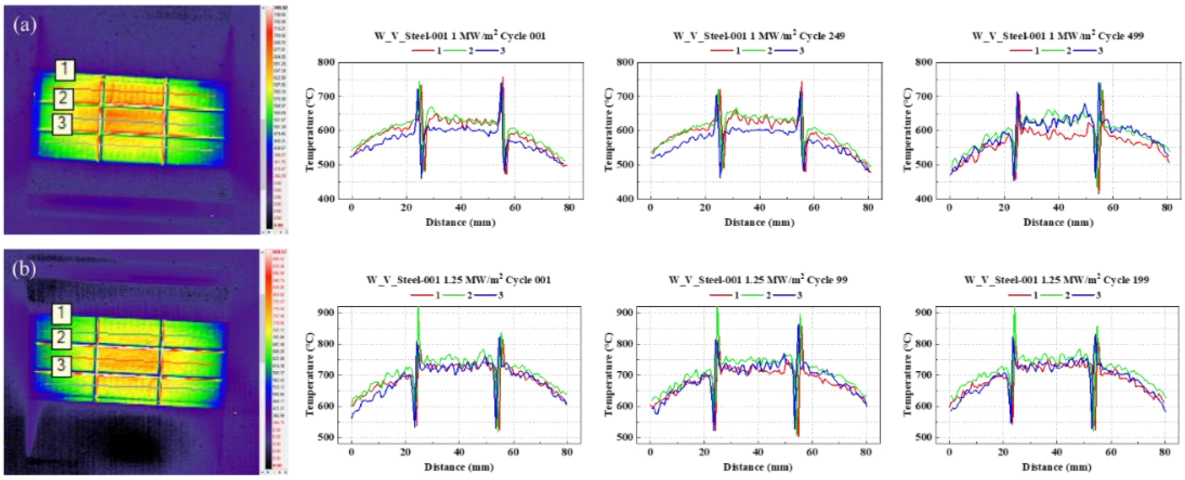

FIG. 6 Tungsten surface temperature distribution and linear variation law under different power densities of HHFT:(a) 1 cycle, 249 cycles, and 499 cycles at 1 MW/m²; (b) 1 cycle, 99 cycles, and 199 cycles at 1.25 MW/m²

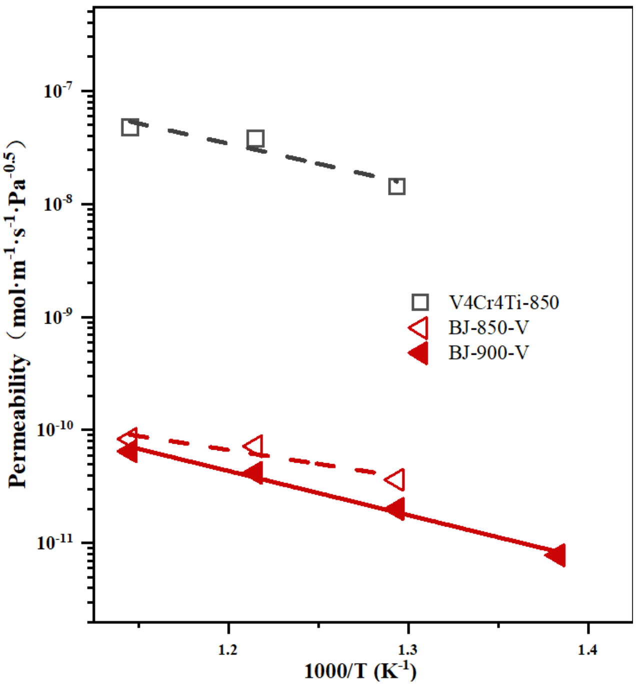

FIG. 7 Arrhenius plots of deuterium permeability of the bonded joint and base material samples

Thermal desorption spectroscopy (TDS) analysis was performed on the specimen (BJ-900-V) after the deuterium permeation test. The results showed that there was a deuterium desorption peak in the temperature range of 700-1100 K (see FIG. 8 (a)), which is very close to the desorption temperature of TiC reported in the literature [8]. This confirms that the in-situ TiC particles formed at the interface provide a large number of irreversible hydrogen traps, which play a key role in reducing the deuterium permeability of the bimetallic material.

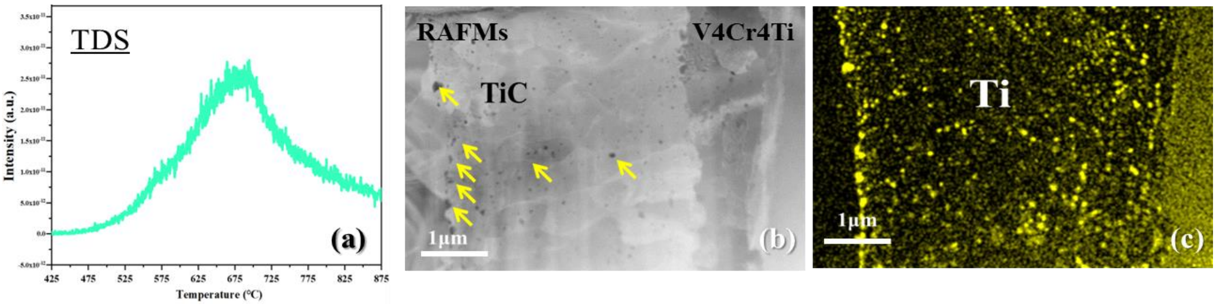

FIG. 8 (a) Thermal desorption spectrum of specimen (BJ-900-V) after deuterium permeation test; (b) SEM-Ti Mapping results of interface of specimen (BJ-900-V)

5. DEVELOPMENT OF OTHER TRITIUM PERMEATION BARRIER IN CHINA

For tritium permeation barrier, cladding the RAFM structure with a FeCrAl layer was also investigated. Institute of Plasma Physics presented that in the internal cooling channel forming of the blanket component for fusion reactor, the combination of Fe-Cr-Al ferritic steel with RAFM steel (reduced activation ferritic/martensitic steel) can form alumina film and then be served as tritium permeation barrier (TPB), which can inhibit the penetration of tritium into structural materials.

Based on it, the bonding experiment of Fe-Cr-Al and CLF-1 steel was carried out. Fe-Cr-Al/Ni/CLF-1 steel has been successfully joined by hot isostatic pressing (HIP) at 1050 °C with a pressure of 60 MPa for 2.5 h [9]. The microstructure and mechanical properties of the joints were analyzed, and the results showed that Fe-Cr-Al/CLF-1 steel interfaces generated some large-size AlN, and the interfacial microvoids are not totally eliminated (as shown in FIG. 9). It has provided a new method for the preparation of Fe based tritium barrier materials with RAFM steel in future fusion reactors. However, the hydrogen isotopes' permeation properties have not been researched yet.

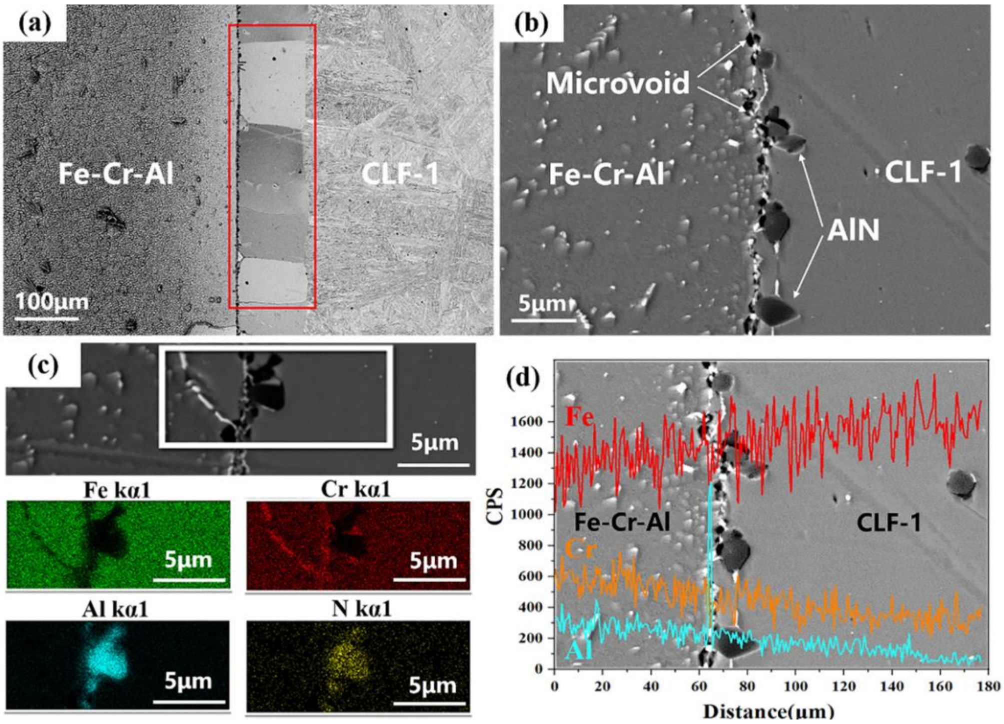

FIG. 9 Microstructure of Fe-Cr-Al/CLF-1 steel joint in (a) Lower magnification, (b) Higher magnification, and EDS scan of Fe-Cr-Al/CLF-1 steel interfaces (c) EDS map-scan results, (d) EDS line-scan results [9]

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