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ACTIVELY COOLED PLASMA FACING COMPONENTS DESIGN FOR W7-X AND JT-60SA IN SUPPORT OF THE ITER DIVERTOR

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Abstract

The present study describes the ongoing developments for plasma-facing components (PFCs) intended for future fusion devices, requiring PFCs to handle high heat loads of at least 10 MW/m² in the divertor region. Tungsten (W) is chosen as the armor material due to its resistance to plasma-wall interactions and high heat flux (HHF). Significant efforts are underway on various fusion devices (WEST, EAST, KSTAR...) to test and optimize water-cooled W-based PFCs. For W7-X, new PFCs are being developed, using W armor material placed at the divertor target region. The design aims to simplify manufacturing, inspection, and installation processes compared to the currently operated actively cooled carbon fiber composite based divertor, while meeting thermal and mechanical constraints. Also, the JT-60SA tokamak plans a transition to metallic PFCs after 2029. The W monoblock concept, successfully used in WEST, EAST and KSTAR and planned for ITER, is being adapted for JT-60SA, and is currently the reference concept. However, some advanced designs exploring cost-saving measures and enhanced heat transfer capabilities are also under development. Manufacturing processes, such as laser powder bed fusion (LPBF) and hot isostatic pressing (HIP), are optimized to ensure reliable and high-density cooling channels for the development of the proposed enhanced design. For these developments, thermo-hydraulic analyses and HHF tests have shown promising results, with surface temperatures being in agreement with models and material temperature limits. In conclusion, the technological and modelling efforts underway are essential for the successful operation of ITER and future fusion power plants, as it will provide feedback from their operation in fusion devices. The integration of advanced materials and

manufacturing techniques, coupled with rigorous testing, is crucial for ensuring the long-term sustainability and performance of fusion reactors.

1 INTRODUCTION

In ITER, plasma scenarios will be developed to achieve durations up to 3000 s [1]. To achieve such durations, plasma facing components (PFCs), which handle the exhaust from the plasma, need to be specially designed. To face heat loads in the divertor region (10 MW/m² as nominal peak stationary power load) and operation duration constraints, water is circulating in the cooling channels of each PFC. Taking into account plasma wall interactions and sustainability under high heat load constraints, tungsten (W) is considered as the best option as an armor material. Up to now, a lot of efforts have been undertaken to understand plasma interaction on W PFCs, mainly in laboratory, but the principal difficulty lies in performing assessments under realistic tokamak plasma operating conditions, meaning simultaneous impact of high fluence, stationary and transient heat fluxes. Until this is achieved, no lifetime assessment is possible for ITER. This statement is also true for a future fusion power plant. As a consequence, to foster a successful operation of ITER and define what is the most appropriate PFC technology, several fusion devices (such as WEST, EAST, KSTAR) are testing W actively cooled PFCs (ITER-grade actively cooled divertor with W monoblock PFC concept) [2]. Further, this research on W PFCs should be also carried on the recent large long pulse devices such as JT-60SA and W7-X. A program within EUROfusion has been launched in 2021 to develop W actively cooled prototypes for W7-X and JT-60SA. Since then, new materials and W PFCs optimised designs have been proposed and successfully tested up to 24 MW/m² in steady state. This paper presents the technological (manufacturing, material development, joining) and modelling (computational fluid dynamics (CFD) and finite element modelling (FEM)) efforts that have been performed in the frame of these developments.

2 DEVELOPMENT OF PLASMA FACING COMPONENTS FOR THE FUTURE W DIVERTOR TARGET OF W7-X

2.1 Guidelines for the design of W-PFCs divertor target for W7-X

W7-X (Greifswald, Germany) is the largest superconducting modular stellarator of the world. It started plasma experiments with a carbon-based water-cooled plasma-facing wall in 2022, allowing long pulse operation [3]. The objective of W7-X is to demonstrate the feasibility of the stellarator concept for a future fusion power plant. One crucial next step to reach this goal is the demonstration of high-performance plasmas with a tritium compatible (i.e., carbon free) wall. Based on the lessons learnt for the manufacturing and installation of the existing divertor a key objective of the design of new plasma facing components for divertor target using tungsten based material (called here after W-PFC) is simplification of: the manufacturing process, the inspection steps, the testing steps and its installation [4]. More specifically, the goal is to minimize the number of welds resulting in a target module made of a single heat sink part (FIG. 1) in which the manifold is integrated and only one inlet and one outlet water connection are required. Some design constraints, as for the currently installed actively cooled divertor [5], are imposed to propose an adapted design [4]:

- Similar curvature of the plasma facing surface
- Design heat load up to 10 MW/m² (no slow transient loadings)
- Use existing cooling water infrastructure (cooling with 1 l/s per 100 mm toroidal length, static pressure 10 bar, pressure drop < 15 bar),
- Size and weight of the target modules must be limited to allow for manual installation (Size < ~A2 size and weight < ~60 kg)

All these constraints, lead to the design proposed in FIG. 1, it is based on circular cooling channel running in parallel (1 channel every 5 mm in the toroidal direction) with a 2.6 to 4 mm diameter. The cooling channel geometry was optimized towards maximum thermal performance assessed by conventional FEM, while manufacturing, mechanical design limits and pressure drop limits are respected.

Thermo-hydraulic analyses were realized using analytical assessment, FEM and CFD analyses. The pressure drop assessed is only 2.65 bar, this has to be compared to the 11 bar of currently installed plasma facing components. Moreover, the heat removal capacity at the edge of the target module towards the pumping gap (curved shape of the target surface) is enhanced to 10 MW/m² allowing for a strike line operated close to the pumping gap.

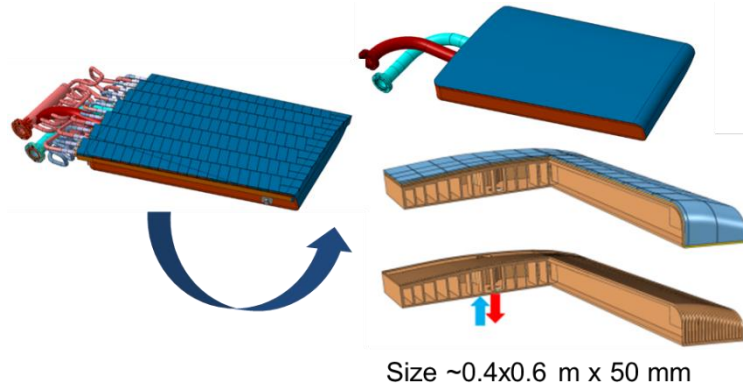


FIG. 1. Simplification of the W7X target module in the view of the tungsten PFC installation, sketch of the: currently used design (left) proposed new design, tungsten is planned to be castellated (right)

2.2 Materials and manufacturing

Based on feedback experience on high heat flux (HHF) testing [6], pure W or W based alloys or composites (Ni, Fe...) are considered as plasma facing material for the W-PFCs of the future W7-X divertor. W based alloys or composites are considered as option since they may represent ductile material or may ease the manufacturing (machinability, manufacturability by coating process for example and joining different materials) of the plasma facing material onto the heat sink. However, the drawback of these materials is to potentially exhibit a lower temperature limit compared to pure W (1300°C) (i.e., 1100 °C (WNiFe)) and significantly lower thermal conductivity. WNiFe demonstrates its ductility, provided the W content is limited to ~95% by weight and a carefully selected annealing scheme is applied. A drawback of WNiFe is the magnetic permeability larger than unity [7], inducing potentially unacceptable changes of the magnetic field, but it has been calculated that the resulting field errors in W7-X are acceptable [4].

Armor materials of up to 4 mm, depending on the armor material type [4] and its final shape facing the plasma, are planned to be used for the present study. W or W based alloys or composites are applied by coating process (as for example with the cold spray process [8]) or using diffusive bonding, employing in this case bulk armor materials. All these proposed processes were already used in the past to produce PFCs using W based alloys or composites as plasma facing material. The purpose of the present development is to check the adaptability of the processes to the particular geometry and related requirement for the W-PFCs of the future W7-X divertor.

CuCrZr is considered as the heat sink material [4] since it presents good thermo-mechanical properties and was considerably used during the series manufacturing of several fusion devices [9] [10]. The properties of this material depend on the manufacturing process and related heat treatments. The heat sink made of CuCrZr is planned to be manufactured using laser powder bed fusion (LPBF) [4]. This manufacturing technology has indeed the advantage to build geometries which may be difficult to manufacture with conventional process (FIG. 2). For the LPBF process, the process parameters were optimized to reach a high density (>99.8%). With these settings, samples were manufactured (FIG. 2) to measure mechanical, thermal and helium leak tightness properties in relation to the heat treatment and manufacturing parameters.

To reduce thermal stresses at the interface between the W-based armor material and the CuCrZr heat sink material, a pure copper interlayer (CuOFHC) of ~1 mm thickness is mandatory. Some processes, such as galvanization, casting [4] and diffusion bonding are investigated to produce the assembled tile (W based alloys or composites with CuOFHC) with the required thickness.

For the complete proposed design, (i.e. including armor, interlayer and heat sink materials), FEM suggests that the proposed design is able to sustain 10 MW/m² in steady-state over the entire plasma facing surface, while design constraints of the existing water infrastructure, available space, weight limitations per module are fulfilled [4].



FIG. 2. CuCrZr samples and heat sinks for plasma facing components manufactured using laser powder bed fusion process

2.3 Testing

The final involved technologies have to be qualified. For this, first, uncooled samples including the armor material and the CuOFHC interlayer are manufactured. If visual observations reveal no important crack at the interface, the quality of the interface is checked with ultrasonic testing (UT) [11]. Finally, HHF tests are performed at GLADIS facility [12] to analyze the ability of the joint material to sustain the required temperatures. Then, the evolution of interface quality after the HHF tests is checked again with UT (FIG. 3). With this, as no defect is emphasized during and after the HHF tests, one concluded that the quality of the CuOFHC interlayer assembled to WNiFe via diffusion bonding process provides promising results. The testing of other alternatives of joining (galvanization, casting...) is underway and should lead to the choice of the technologies involved for the series manufacturing of the W-PFCs of the future W7-X divertor.

To qualify the ability of the proposed cooling concept to sustain the required heat loads, HHF tests of small scale CuCrZr heat sinks (146x57x26 mm) implemented with the desired cooling design were also performed at JUDITH-2 facility [13].

The results of the HHF tests are promising since no damage was observed on the tested components. Moreover at 10 MW/m² (mass flow rate of 0.5 l/s) the measured surface temperature is below 370°C, which is in good agreement with FEM calculation (~370°C) and the admissible temperature use window of CuCrZr (<450°C). Further, this small-scale mock-up was tested up to 24 MW/m² without leading to critical heat flux.

Moreover, the pressure drop of the small scale heat sink was measured. It reveals a good agreement between modeling (1.7 bar at 0.5 l/s) and experiment (1.6 bar at 0.5 l/s) which emphasizes the possibility to rely on the CFD modeling for assessing the pressure drop and provides good feedback on the LPBF process to produce reliable cooling channel quality.

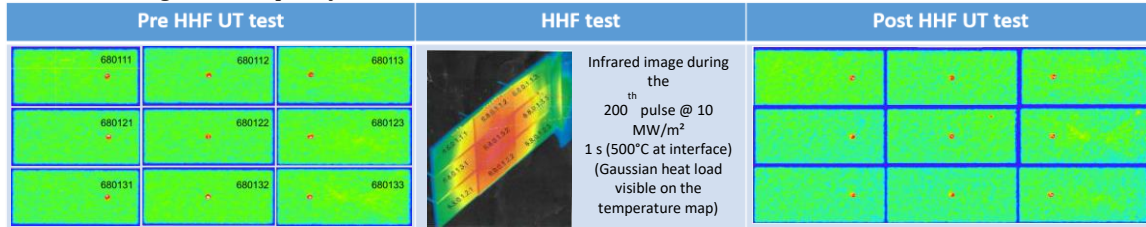


FIG. 3. Strategy of testing for the qualification of the joining processes (Ultrasonic testing (UT) before and after high heat flux tests)– The strategy is applied here to check the quality of nine samples using diffusion bonding process to assemble armor material (WNiFe) to soft copper. On the UT maps, the green areas means that the joining between WNiFe and pure copper is existent, while the red areas are representative of machined holes at interfaces used for ultrasonic testing calibration

3 DEVELOPMENT OF PLASMA FACING COMPONENTS FOR THE FUTURE W DIVERTOR TARGET OF JT-60SA

3.1 Guidelines for the design of W-PFCs divertor target for JT-60SA

On its side, the tokamak JT-60SA has been constructed and will be operated in the framework of the broader approach with strong European support. A transition to a metallic device is foreseen after 2029 for the so called “Integrated Research Phase II”. For the divertor region, PFCs have to sustain up to 10 MW/m² in steady-state [14] [15]. PFCs based on the W monoblock concept, as the one used for the divertor region of the ITER divertor [16], have been chosen as the reference. Indeed, such concept proved to sustain heat loads in the range of 10 MW/m² in steady state and up to 20 MW/m² for slow transient regime [17][18] [19]. Moreover, the series production of

components based on the W monoblock concept have already been done in the past [20] [10]. The procurement of such components for JT-60SA is currently handled by Fusion for Energy (F4E) in the frame of a contract with industry. For these W-PFCs, as for the W7-X ones, W based alloys or composites are envisaged as armor material while CuCrZr is envisaged as heat sink material.

In addition to the current development led by F4E, some advanced W actively cooled PFCs are currently developed to propose alternative designs, enabling cost saving on the series manufacturing phase and / or sustaining heat loads in steady state higher than 20 MW/m^2 . One way to reduce cost is to use a concept with flat interface surfaces (W to CuOFHC, CuOFHC to CuCrZr), which is easier to manufacture than the ones involved for the fabrication of the reference concept (W blocks using cylindrical interfaces). To sustain heat loads higher than 20 MW/m^2 , one option is to propose cooling design able to increase the heat transfer coefficient between the cooling channel and the coolant by taking advantage of the nucleate boiling. The current sketches of the reference and of the enhanced designs of the envisaged W-PFCs divertor target for JT-60SA are enclosed in FIG. 4. One can note that, to compensate the 90° transition of the component to the manifold, a part of the PFC enclosed a portion with flat W tiles (5 mm) assembled onto flat CuCrZr heat sink. To connect these components to the cassette manifold, a transition to stainless steel (SS) pipes is needed [15] (FIG. 4a right).

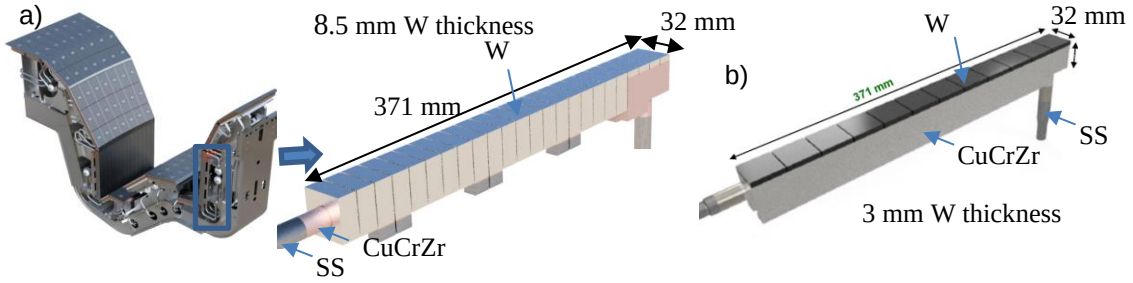


FIG. 4: Sketches of reference (a) and enhanced (b) designs for the JT-60SA W-PFCs divertor target – Tungsten (W), CuCrZr and stainless steel (SS) localizations. A pure copper (CuOFHC) interlayer is placed between W and CuCrZr (not visible on this figure).

3.2 Manufacturing and testing

For the reference and advanced concepts, armor material is planned to be procured along with the tungsten ITER procurement specifications [16]. For the reference concept, CuCrZr material is planned to be in form of forged blocks and seamless tubes while for the advanced concept, as for the development of the W7-X W-PFCs, laser powder bed fusion is envisaged to produce the heat sink. Indeed, this type of manufacturing presents the advantage to improve the heat exhaust capability by producing optimized cooling channel designs.

The complete manufacturing process involved for the fabrication of the reference concept is presented in FIG. 5. For these PFCs, one of the major challenges is to produce W monoblocks with the adequate quality. As a consequence, some mock-ups composed only with W monoblocks have been manufactured. These mock-ups undergone the HHF test protocol enclosed in FIG. 6. For the tested mock-ups, there is a stable surface temperature distribution during cyclic loading with 10 MW/m^2 and 15 MW/m^2 . No degradation of the bonding of the blocks to the tube is indicated, nor any visible damage on the W blocks (FIG. 6) is emphasized. However, it has to be noticed that one of the block presents a different thermal behavior during HHF tests, as also identified with the SATIR test [21]. Considering the correlation of the HHF test and the SATIR tests, it seems that the thermal loadings undergone by the thermal imperfection (probably an interface quality lower than the other blocks) does not impact the interface quality. Then, as the proof of principle of the manufacturing steps for final geometry, two intermediate scale mock-ups were manufactured. These elements also showed promising results under HHF tests.

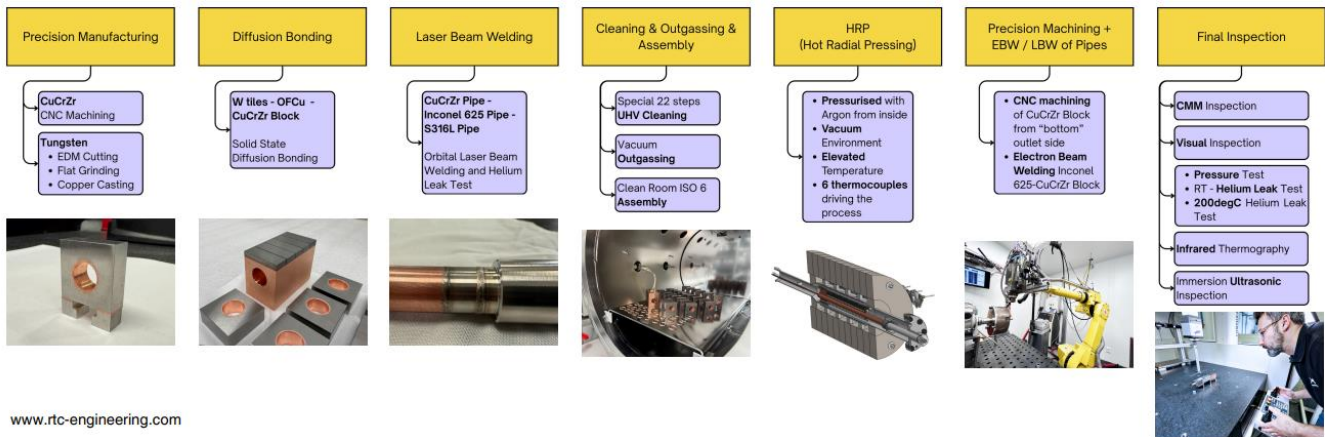
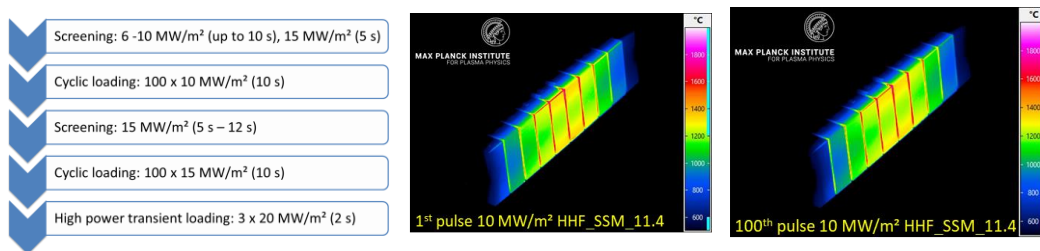


FIG. 5. Involved steps for the manufacturing of W monoblock concept

FIG. 6. High heat flux testing protocol at GLADIS and infrared image during HHF test at 10 MW/m²

For the enhanced concept, it turned out that the use of additive manufacturing technique also drives some constraints in the PFC geometry definition (for example the wall thickness has to be defined to compensate the risk of failure during the manufacturing process....) [22]. Due to the intrinsic characteristics of the LPBF process, namely rapid solidification, directional solidification, and layer-wise build-up, LPBF CuCrZr mock-ups manufactured exhibit residual porosities in the form of lack-of-fusion defects. These defects resulted in a measured relative density of 98.8%. Higher densities may be achieved but it is not essential at this step of the manufacturing since hot isostatic pressing (HIP) is planned to be applied after the additive manufacturing process. HIP was applied in order to close porosities. As high temperature and high pressure are necessary to close them, it was decided to perform this heat treatment with the same temperature and pressure than those qualified for the fabrication of the CuCrZr/SS joint (980°C/2h). Indeed, for the studied components, this type of joining will be needed at the final step of the manufacturing process to join CuCrZr and SS pipes. After the solution annealing (SA) at 980°C for 2 hours, gas quench at 150°C/min and ageing at 580°C for 2 hours, the porosities are eliminated. The microstructure after HIP showed similar microstructure as the as-printed one with only partial recrystallization areas, and a final density of 99.6%, confirming the effectiveness of the HIP process to close the porosities and obtain high density for materials made by powder metallurgy. The influence of thermal treatment (including gas cooling rates) on the CuCrZr mechanical properties was also shown in [22].

CuCrZr mockups with different cooling configurations (chevron, diagonal) were manufactured using LPBF process, followed by thermal treatment ensured by HIP (solution annealing, thermal quench and ageing) (FIG. 7). The definition of cooling channel geometries was studied by CFD and provides the conclusion that such a concept could be able to sustain heat loads higher than 20 MW/m² [22]. We also emphasizes the difficulty to model two-phase flow regimes [23]. This needs to be overcome in order to predict the thermal behavior of PFCs with this type of enhanced cooling regime. One of the solutions proposed in this project is to provide inputs to the CFD models by the visualization of bubbles flowing into the channels [24].

The heat extraction capacity of the manufactured models was tested on the HADES high flux test facility (DRF/IRFM) [25]. Up to 15 MW/m², the components exhibited expected thermal behavior. At 20 MW/m², however, the Diagonal design exhibited higher surface temperatures. Post exposure investigation hint to the fact that this behavior could be due to the presence of residual sintered powder resulting from insufficient internal depowdering after printing. Components were up to 24 MW/m² without leading to critical heat flux

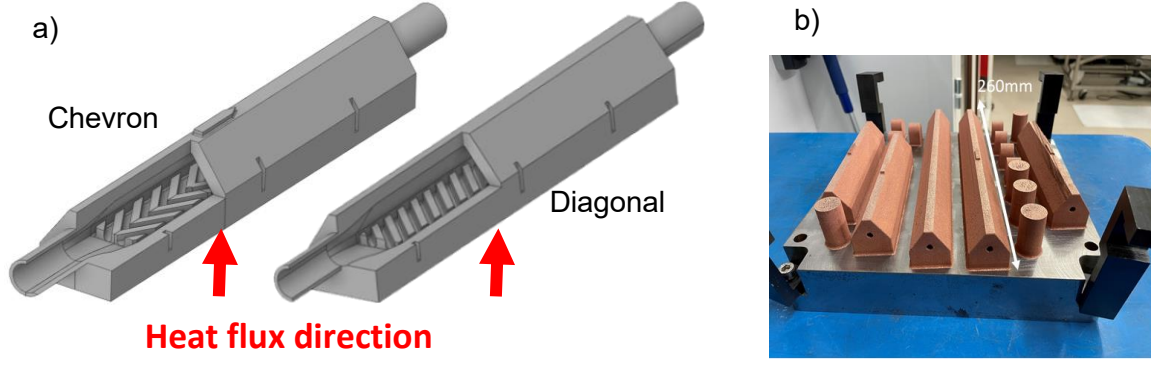


FIG. 7: Mock-ups manufactured for the development of JT-60SA advanced W-PFC with additive manufacturing: sketches showing different cooling channel configurations (a) and examples of manufactured mock-ups (b)

4 CONCLUSIONS

The development of plasma-facing components (PFCs) for future fusion devices such as W7-X and JT-60SA is crucial for achieving long pulse duration, while sustaining high heat loads in ITER. For W7-X and JT-60SA, future divertor targets are planned to use tungsten (W) based materials as armor material. For W7-X, the design of new W-PFCs aims to simplify manufacturing, inspection, and installation processes, while meeting stringent thermal and mechanical constraints. The proposed design features circular cooling channels optimized for thermal performance and manufacturability. Thermo-hydraulic analyses and high heat flux (HHF) tests have shown promising results, with mock-ups sustaining 10 MW/m^2 in steady-state and demonstrating good agreement between modeling and experimental data. Mock-ups were tested up to 24 MW/m^2 , without leading to critical heat flux. For JT-60SA tokamak, the W monoblock concept, proven efficient under ITER conditions, is currently the reference concept for the future actively cooled JT-60SA divertor target, which will use tungsten as armor material. Mock-ups for such application were HHF tested. A stable surface temperature distribution during cyclic loading with 10 MW/m^2 and 15 MW/m^2 were noted. Advanced designs exploring cost-saving measures and enhanced heat transfer capabilities are also studied. Manufacturing processes, including laser powder bed fusion (LPBF) and hot isostatic pressing (HIP), are being optimized to ensure high-density and reliable cooling channels leading to promising results in terms of heat flux handling capability. Indeed, modeling and experiments results show that such enhanced designs may be able to sustain heat loads higher than 20 MW/m^2 . Overall, the technological and modeling efforts undertaken in these developments are paving the way for the successful operation of future fusion devices. The integration of advanced materials and manufacturing techniques, coupled with rigorous testing and validation, will be essential for achieving the long-term sustainability and performance of fusion reactors.

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