Development of the upper divertor for the HL-3 tokamak

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The divertor is a core component in tokamak fusion devices, responsible for managing core heat exhaust and impurity removal under extreme operational conditions, while withstanding intense particle bombardment and high heat fluxes. In 2023, the HL-3 fusion device achieved ion and electron temperatures exceeding 100 million degrees Celsius, with the lower divertor playing a critical role in enabling this milestone. To further improve plasma performance and support China's first deuterium–tritium (DT) fusion experiment scheduled for 2027, the device will operate in a double null configuration, necessitating the development and installation of an upper divertor. The upper divertor employs carbon fiber composite (CFC) and tungsten as plasma-facing materials, CuCrZr alloy as the heat sink, and 316L stainless steel for structural support. After more than one year of intensive engineering development, the fabricated divertor modules successfully passed high heat flux testing under 10 MW/m^2 for 500 cycles, meeting qualification requirements. A total of 82 CFC-based and 2 tungsten-based divertor modules were produced. This paper presents a systematic overview of the manufacturing processes and inspection protocols for the upper divertor, with particular emphasis on the development of domestically produced fusion-grade CFC materials and the joining technologies for dissimilar materials—specifically the integration of CFC and tungsten with CuCrZr alloys.

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Water-cooled STEAM heat shield against extreme steady plasma flux 80-160 MW/m²

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Topics

G. Plasma-Facing Components

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Oral

Abstract

Effectively withstanding extreme surface heat fluxes, possibly exceeding the steady-state limit 10-20 MW/m^2 of the best nowadays shield (ITER-like divertor), continues to pose one of the major unresolved engineering challenges for a practical thermonuclear fusion reactor. Therefore we present a novel heat shield STEAM (Super Thermal Exchange via Advanced Microboiling) aimed for a tokamak or stellarator divertors. It employs high-speed water impingement cooling, enhanced heat transfer area and efficient boiling-driven heat removal. It is tested on a plasma torch under air atmosphere (at Technology Readiness Level 3-4), supported by heat transfer simulations, infrared thermography and calorimetry. Water-tight STEAM sustained intact for over 20 minutes (in 36 cycles) heat flux of 80 MW/m^2 = 31 kW / 3.7 cm2, exceeding thus 5x the ITER-like divertor PFC. Continuous inertial impingement flow suppresses dryout and enables steady heat removal far beyond the classical critical heat flux limits, limited probably only by heat conduction of the thin front plate and the sound speed of the steam+liquid mixture. Laser enlighted fast camera and thermovision revealed a strongly boiling outlet water foam: a million steam bubbles per second and cm2. Thanks to a shape-optimized water injector, it sustained under heat flux rising for 1 minute up to 160±30 MW/m^2 — probably representing the most intensive sustained non-destructive heat shield test ever reported.
Exposure to the 100 kW HADES electron beam in vacuum and gradient coating by tungsten (necessary for divertors) is planned until Summer. We'll also present a conceptual design of STEAM cooling a gyrotron resonant cavity.

Application and damages of ITER-like W/Cu monoblocks used as the divertor targets in EAST

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

In the future ITER, the actively cooled W/Cu monoblocks will be employed for divertor target, aiming to exhaust the 10 MWm-2 or even more high heat loads[1]. They are also the first choice for the next-generation fusion devices, currently. Although the high performance W/Cu monoblocks can pass through the harsh high heat flux tests(1000 cycles 10 MWm-2 and 300 cycles 20 MWm-2) [2], their behavior under actual in-situ tokamak plasma conditions is not completely clear so far. Thus, tests of such kind of W/Cu monoblocks in current tokamaks is still very essential. Especially, the damages on W/Cu monoblocks in current tokamaks are the most effective evidences to speculate about possible future component failures in future ITER.

The EAST, as an full superconducting tokamak with similar magnetic configurations and long pulse capacity like the ITER, was gradually installed actively cooled ITER-like W/Cu monoblocks for its upper and lower divertor targets in the last decade [3]. With the application of such high performance W/Cu monoblocks for high heat load components, many world records, such as the long pulse up to 403 s H-mode plasma disharge(#122296) and 1066 s high electron temperature up to 8.4 keV plasma discharge (#150254), were thus achieved in recent plasma campaigns by the EAST. However, after a long term plasma exposures with peak heat loads in the range of 10 ~20 MWm-2 on the surface of W/Cu monoblocks, many types of damage phenomena, i.e. the cracking of W/Cu joint interface, both macro and micro cracking at leading edges, the net-like cracking on surface, the recrystallization even abnormal W grain growth and so on, have been observed and inspected on W/Cu monoblocks. The morphological feature and distribution pattern of these damages are well described by means of in situ photography and post mortem characterization. And their possible generation mechanisms are deeply analyzed and discussed according to the plasma loading conditions and the corresponding thermal mechanical simulations.

The behavior and mechanism of these various damages on actively cooled ITER-like W/Cu monoblocks as the divertor targets in the EAST are very important references for the future ITER and other tokamaks.

Groove-Induced Vortex Effects on Bubble Dynamics and Heat Transfer in Divertor Hypervapotron Channels

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Managing the extreme heat fluxes in fusion reactor divertors remains a critical challenge. The hypervapotron (HV) is widely favored for its superior thermal performance; however, the fundamental two-phase mechanisms governing its critical heat flux (CHF) limits remain elusive. While groove-induced vortices are conventionally assumed to enhance heat transfer by ejecting vapor into the mainstream, this hypothesis lacks direct microscopic verification. Here, we investigate the vapor-liquid dynamics in HV channels using transient Volume of Fluid (VOF) simulations validated by visualization experiments. Contrary to the prevailing "vapor ejection" model, our results reveal a "vortex confinement" phenomenon, where strong shear-induced centrifugal forces trap vapor bubbles within the vortex core. Consequently, we demonstrate that the enhanced heat transfer stems primarily from a localized phase-change cycle—comprising sustained thin-film evaporation and rapid condensation at the shear layer—rather than bulk bubble removal. Crucially, we identify that CHF is triggered when the vapor generation rate exceeds the shear-layer condensation capacity, causing the confined bubbles to coalesce into an insulating vapor blanket.These findings revise the understanding of heat transfer mechanisms in hypervapotron structures and establish a new theoretical basis for optimizing fusion cooling components.

Neutron irradiation effects in tungsten based materials

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The EUROfusion Consortium is an association of Laboratories across 28 European countries in charge of implementing the European Roadmap to Fusion Electricity. Development and qualification of neutron tolerant materials is one of the missions in the Roadmap, with neutron irradiation and subsequent post irradiation examination representing a key challenge requiring substantial time and financial resources, and long-term vision for the codification of the results into reactor design codes.

In this paper, we provide an overview of the EUROfusion contribution to the investigation of the neutron irradiation effects in different commercial and advanced tungsten grades provided by different partners within EUROfusion as well as via international collaboration programmes, including divertor relevant joints. The presentation summarizes the main results and lessons learned over the 2014-2026 period, and it is sub-divided into five sections covering: (i) commercial ITER-specification tungsten; (ii) advanced tungsten grades achieved by alloying or innovative production methods; (iii) impact of thermal-neutron shielding on the irradiation damage; (iv) performance under very high irradiation temperature; (v) irradiation of W-CuCrZr and W-Cu-CuCrZr joints. The effects of neutron irradiation are assessed in terms of the modification of mechanical properties, and supported by microstructural investigations to explain and interpret the obtained results.

Advanced manufacturing of tungsten for nuclear fusion applications using laser deposition: microstructure and properties

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Tungsten has been widely used in plasma-facing components for first wall and divertor of future nuclear fusion reactors. However, pure tungsten is difficult to process due to its high melting point and brittleness. As components directly exposed to plasma, tungsten faces the risk of burn-out and cracking during high-temperature plasma operation; therefore, surface repair of tungsten has become a key concern. This study aims at the issue of surface repair by employing selective laser deposition process to fabricate pure tungsten. A novel ring and core beam shaping strategy with a single mode core and a multimode annular ring was utilized to achieve effective keyhole formation using core and a wider and homogeneous melt pool with annular ring. The microstructure and properties of pure tungsten were systematically investigated under constant core power but varying ring powers, substrate and powder size. The element distribution and defects in printed tungsten were also analyzed in depth. Results indicated optimal surface quality and properties of tungsten were obtained at laser ring power of 320 W and a powder size of 15-25 μm on RAFM (reduced activation ferritic/martensitic) steel. Cross-sectional microstructural observations revealed that the W-steel interface ultimately formed a three-zone gradient material. Cracks predominantly occurred in the transition zone, primarily due to elemental migration and the formation of hard, brittle Fe-W inter-metallic compound. Grains in this transition zone exhibited a dendritic structure. Hardness distribution data revealed exceptionally high Vickers hardness (650 HV3) in the transition zone, closely correlated with microstructure and the brittle second phase. By using the finest powder size and moderate power, tungsten parts with a purity of up to 97% can be obtained, and the crack rate in tungsten is minimized. This work explored an innovative and promising manufacturing method of pure tungsten without vacuum environment, offering new insights for on-site manufacturing of fusion reactor components.

Investigation of glow discharge cleaning by cryogenic trap assisted mass spectrometry at Uragan-2M stellarator

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The wall conditioning procedure is one of the phases in conducting experiments with high-temperature plasma in thermonuclear fusion research. One of the most often used methods for wall conditioning is glow discharge cleaning (GDC). It is used in devices with magnetic plasma confinement, such as tokamaks and stellarators. This method is also used on the Uragan-2M stellarator. Mass spectrometry (MS) is usually used to control the GDC process. This method has limitations in terms of sensitivity. In addition, in the case of polyatomic molecules generated during wall conditioning, difficulties arise in interpreting mass spectra because the products of molecular dissociation may have the same masses. To overcome these limitations, cryogenic trap assisted mass spectrometry (CT-MS) is used to study wall conditioning discharges. During the discharge, the gases condense in a cryogenic trap, and, after the discharge stops, the trap is defrosted, and the composition of the released gases is analyzed using a mass spectrometer.

This study presents the results of the first GDC by CT-MS investigations at the Uragan-2M stellarator. The cryogenic trap was cooled with liquid nitrogen. Glow discharge cleaning was performed in an argon atmosphere. Experiments were conducted using three scenarios. In the first scenario, without GDC, only the vacuum chamber was pumped, and CT-MS was used. With GDC, in the second scenario, the MS method was used, or the CT-MS method (third scenario) was used instead. When GDC burns, a higher intensity for M=2 (hydrogen) is observed in the mass spectrum than without GDC. This correlates with the intensity of the hydrogen spectral line measured by optical emission spectrometry. A similar increase is observed for masses M=28 and M=44. When the cryogenic trap is heated, a significant increase in intensity, several times over, is observed for the majority of masses. In particular, an increase in intensity is observed sequentially for masses characteristic of carbon dioxide, acetylene, and water. The intensities for M=2 (hydrogen) and M=12 (carbon), products of molecular dissociation, also increase. The results obtained for the three scenarios were compared and analyzed.

Ratcheting-Constrained Design of Rounded-Edge Fin Monoblock Hypervapotrons under 15 MW/m2 Divertor Heat Flux

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Ratcheting induced by cyclic loading imposes a critical limitation on structural qualification in divertor plasma-facing components (PFCs) under ITER-relevant high heat flux. This constraint originates from thermo-mechanical interactions among tungsten armor, interlayers, and CuCrZr cooling structures and is exacerbated by geometric stress concentrations inherent to conventional flat-tile monoblocks. In this study, a rounded-edge fin monoblock hypervapotron design is proposed to mitigate ratcheting. Elasto-plastic finite element analyses are performed to quantify global and local ratcheting behavior, and elastic-plastic compatibility is verified to ensure applicability of the Monoblock Elastic Analysis Procedure (MEAP). At 10 MW/m^2, replacing a rectangular fin with a rounded-edge geometry increases the MEAP Rule-1 index from 0.87 (non-qualified) to above the qualification threshold (≥ 1.0), with a 1.5 mm fin radius identified as optimal. However, the design loses robustness at a higher heat flux of 15 MW/m^2. Thermal optimization through heat transfer enhancement and optimized operating conditions (120 °C inlet temperature, 80 bar pressure, 28 m/s coolant velocity) improves the index by approximately 29%, yet remains insufficient to meet the qualification criterion. A ratcheting-constrained cooling-channel geometry transformation increases the index to 1.00. The introduction of a copper interlayer further elevates the index to 1.06 and maintains qualification under harsher conditions (150 °C temperature and 20 m/s velocity). These results demonstrate that thermal qualification alone is insufficient to ensure structural integrity, and the proposed ratcheting-constrained design strategy enables robust qualification while remaining compatible with existing monoblock manufacturing routes, thus making the concept directly implementable for ITER-relevant divertor applications.

Development and Testing of a Novel FAST-Diffusion Bonding Process for the Joining of Copper Chromium Zirconium Alloys to Tungsten for Plasma Facing Components

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten (W) is a leading candidate for the armour material of plasma-facing components (PFCs) in nuclear fusion reactors due to its high melting point, excellent thermal conductivity, low tritium retention, and high resistance to sputtering. Copper Chromium Zirconium (CuCrZr) alloys are commonly used as structural materials for divertor components and other high- to medium-heat-flux regions due to their favourable thermal conductivity and creep resistance. However, the joining of W to CuCrZr is challenging, due to their differences in melting points, an approximately 360% mismatch in thermal expansion coefficients, and their mutual immiscibility.

Current solutions, including the use of Cu interlayers in hot radial pressing or hot isostatic pressing, have successfully enabled the joining of CuCrZr cooling pipes to W monoblocks. However, despite being well established, monoblock-based PFC concepts require large quantities of W and a high number of individual components. As a result, increasing attention is being given to alternative “tile on heat sink” designs, in which large-area W tiles are bonded to CuCrZr substrates. [1]

This work reports significant progress in the development of a joining technology for large-scale “tile-on-heat-sink” PFC designs. In collaboration with the Dr Fritsch Group, industry-scale field-assisted sintering technique (FAST) equipment was employed to diffusion bond 100 mm × 100 mm × 5 mm tungsten (W) tiles to 100 mm × 100 mm × 22 mm CuCrZr heat sinks. The developed FAST diffusion bonding (FAST-DB) process enabled both direct W–CuCrZr joining and joining using different interlayer concepts. Building upon previous work, similarly sized first wall PFCs were manufactured by FAST-DB of W to Eurofer97 steel. All joints were machined into mock-up components and subjected to high heat flux testing at the JUDITH 2 facility at Forschungszentrum Jülich. The influence of interlayer design on joint integrity and interfacial microstructure was investigated using scanning electron microscopy and energy-dispersive X-ray spectroscopy. Optical microscopy and non-destructive testing were performed at successive machining stages to identify defects and assess manufacturing robustness. The key mechanisms underlying FAST-DB that address the challenges of W to CuCrZr/Eurofer97 joining were analysed using finite element modelling and utilized to increase the bond strength and reduce the residual strain in the W armour.

[1] T. Hirai, et al., Fusion Eng. Des. 125 (2017)

Lessons learned from European ITER-like divertor components installed in the WEST tokamak

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Under the auspices of a collaboration between the ITER Organization, its European Domestic Agency Fusion for Energy (F4E), and the CEA, 20 european ITER-like Plasma-Facing Units (PFU) have been progressively installed in the WEST tokamak.
The design of these actively-cooled PFUs consists of shaped tungsten (W) monoblocks (MB) bonded to a CuCrZr cooling tube, representative of the straight (high heat flux) part of the ITER divertor [1]. Manufactured by four ITER's European suppliers, they include different bonding methods and W grades.
The performance of these PFUs is tested since 2020 under various operating conditions, combining steady-state regimes (high fluence or high power up to 12 MW/m2 at the outer strike point (OSP)), wall conditioning, off-normal events (~2000 disruptions), but also laser cleaning to remove surface layers accumulated during plasma discharges.
This contribution presents the first lessons learned from this “WEST Joint Team” manufacturing and testing program.

An overview will firstly be provided of the manufacturing feedback based on the strict ITER technical requirements for suppliers S1, S2, S3 and S4 (detailed information on supplier will not be disclosed at this stage of the Series production and tender competition).

Secondly, a summary of the PFU pre-characterization and reception tests performed at CEA will be presented. Results include: an inventory of existing surface defects and small dimensional non-conformities, W hardness measurements (conformed to specifications), a comparison between non-destructive tests on the W-Cu bond (showing good consistency between ultrasonic testing, infrared thermography, and High Heat Flux tests [2]).
Following these successful fabrication and installation steps and subsequent exposure to plasma through several experimental campaigns, two important observations were made: 1) as for all other PFUs constituting the WEST Divertor, EU PFUs have developed micro-crack networks on monoblock surfaces in the OSP area (detailed analysis provided in [3]) ; 2) so far, all PFUs demonstrated good heat exhaust capability under WEST conditions.

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

[1] M. Missirlian et al., FED., vol.89, no. 7, pp.1048–1053, Oct. 2014
[2] O. Buxton et al., SOFT 2024
[3] O. Buxton et al., SOFT 2026

A Porous-Medium MHD Model for Pressure Drop Prediction in Liquid Metal CPS for fusion applications

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

In magnetic confinement fusion reactors, the divertor is a critical Plasma Facing Component forming the boundary between the fusion plasma and the rest of the device, tasked with handling power and particle exhaust. Liquid metals (LM) are considered a promising Plasma Facing Material for divertor armor due to their self-healing and vapor shielding effects. However, LM must be confined to prevent droplet emission, splashing, and vacuum chamber contamination caused by plasma-wall interactions, electromagnetic forces, and fluid instabilities. Magnetohydrodynamic (MHD) effects play a significant role in defining the LM-PFC operational space.

A possible approach envisions using a Capillary Porous System (CPS) to confine the LM through capillary force. In case of a CPS filled with flowing LM, as the one that will be developed for the linear plasma generator at the Dutch Institute for Fundamental Energy Research (DIFFER) and the Eindhoven University of Technology, it is fundamental to evaluate the pressure drop given by the porous structure. An accurate estimation of the pressure drop is essential for the proper hydraulic design and dimensioning of the LM circuit. This pressure drop is significantly influenced by MHD effects in the presence of strong magnetic fields.
To this purpose, a numerical tool has been developed to estimate the pressure drop in LM-filled CPS under magnetic field influence. The model extends an existing OpenFOAM solver by incorporating additional momentum sink terms into the momentum conservation equation, based on Darcy–Forchheimer coefficients modified to account for MHD effects. This approach represents the CPS through effective porous-medium coefficients rather than explicitly resolving the porous structure. This limits the computational cost while retaining the key macroscopic flow features, and may facilitate its application to CPS configurations with different geometries with limited modelling effort. This study presents an initial validation of the proposed model against available numerical and experimental data obtained under MHD conditions, assessing its capability to reproduce the pressure drop and overall flow behavior in LM-CPS systems.

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Probabilistic Evaluation Method for Cracks on Divertor Armors in Fusion Reactors Using Direct Current Potential Drop Technique

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The divertor consists of a large number of tungsten blocks (tungsten monoblocks) that protect the heat removal components from intense thermal loads. Previous studies have reported that thermal loading from fusion plasma can cause fatigue cracks on the monoblocks. These cracks reduce the divertor's cooling performance and can result in local melting of tungsten. Therefore, early detection and reliable evaluation of the cracks are necessary.

Eddy current testing is suitable for the rapid inspection of the monoblock surfaces. However, high electrical conductivity of tungsten hampers crack sizing from measured eddy current signals. Ultrasonic testing, which is widely used for crack evaluation, is also unsuitable for the vacuum environment of fusion reactors as it requires a liquid couplant. In contrast, the direct current potential drop (DCPD) technique is a promising method for evaluating the cracks on the monoblocks. However, DCPD signals are sensitive to variations in the geometry of the monoblocks, uncertainty in the location of the cracks, and uncontrollable environmental factors (e.g., noise and thermal contact resistance). These factors cause signal fluctuations that hinder the deterministic evaluation, so a probabilistic approach is required to quantify the uncertainty of the evaluation.

In this study, we develop a probabilistic crack-sizing framework based on a parallel ensemble model, which is used to estimate class probabilities of crack surface length and maximum depth from DCPD signals. The ensemble comprises of multiple shallow classification neural networks. Each network is trained using DCPD signals generated by 3D finite element simulations and augmented with uniform noise at various levels that emulate measurement and evaluation errors. The final output is obtained by averaging the class probabilities across these networks. Thus, robust estimation is achieved under unknown noise levels. The proposed method is validated using test DCPD signals measured from brass specimens containing machined slits designed to mimic monoblock cracks. Brass is selected here because its electrical conductivity is similar to that of tungsten. A direct current is injected through two surface terminals, and the test DCPD signals are measured using ten electrodes across the slit. The results demonstrate that the proposed method successfully estimates both slit geometries and uncertainty as a probability distribution, with the estimated slit geometries being particularly close to those in the specimens for deep slits.

FILMS - Fully Integrated Liquid Metal System for a Fusion Reactor

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Fully integrated liquid metal (LM) plasma-facing and breeding systems offer a promising approach for fusion reactor concepts capable of withstanding extreme heat and particle loads while supporting tritium breeding and power conversion. This paper introduces and evaluates the Fully Integrated Liquid Metal breeding/cooling System (FILMS), a concept that combines three traditionally separate subsystems into a single LM circuit: (1) a flowing LM first wall (FW), (2) an LM breeding blanket, and (3) an open-surface LM divertor. In the FILMS concept, LM is supplied at the top of the vacuum vessel and forms a thin (~0.5-2 cm), high-velocity (~5-10 m/s) open-surface flow over FW structures. This flowing layer is intended to remove the plasma surface heat flux and a portion of the volumetric nuclear heating. Downstream of the FW, the LM flow is divided into two branches: one branch feeds the open-surface divertor region, designed to accomodate the highest heat and particle fluxes, while the second branch supplies the breeding blanket located behind the FW, where only volumetric heating is present. After passing through the divertor and blanket, the LM streams are collected at the bottom of the chamber and directed toward external systems for heat extraction, tritium processing and recirculation. The feasibility of FILMS is governed by multiple coupled engineering challenges, including magnetohydrodynamic (MHD) effects on LM flow, heat transfer under extreme surface and volumetric loads, corrosion and mass transport in high-temperature LM enviroments, tritium transport and control, and the stability of open-surface flows in strong magnetic fields. Of particular concern are MHD-driven instabilities, free-surface wave amplification, and potential droplet formation, which may lead to flow non-uniformity, dry spots, or enhanced vapor flux to the plasma. Materials constraints further limit the allowable operating temperature range and influence blanket and structural design choices. The present work considers liquid lithium (Li) as the working fluid and focuses on scoping-level analyses and integrated engineering assessments of the FILMS concept. While the scientific and engineering challenges are substantial, the potential benefits are significant. Employing a single liquid to perform plasma-facing, breeding, and cooling functions can substantially simplify reactor architecture by reducing internal solid components and intefaces, lowering mechanical complexity, and improving thermal efficiency. The results of this work define preliminary operating windows for flow velocity, layer thickness, temperature, and pressure drop and establish a basis for future detailed 2D and 3D simulations , aimed at assessing design feasibility for application in a Fusion Pilot Plant.

Manufacturing Process for Preventing Cutting Chip Ingress and Contamination in the Cooling Circuit of the ITER Shield Block

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

ITER Blanket Shield Block (SB) is a key in-vessel component for providing neutron shielding to support Vacuum Vessel (VV) and ex-vessel components under severe operating condition such as nuclear, thermal load and electromagnetic force. To absorb the thermal load, SB has structure as Cooling Circuit (CC) with coaxial connector to circulation of coolant, and it is connected with First Wall (FW) panel and manifold by remote handling welding.

In the machining process of the SB, preliminary measure such as caps are implemented to prevent ingress into the cooling circuit. Nevertheless, because the connection between the connector and the pre-machined circuit is formed, there remains a risk of cutting chips entering the interior during the finishing process on the inner surface of the coaxial connector, as a large volume of cutting fluid is continuously supplied adjacent to the cutting tool.

In this study, the cutting oil supply method was improved by transitioning from the conventional approach to the use of the SB cooling circuit. This method completely prevented the chip ingress into the cooling circuit during final machining. Although the cooling circuit was contaminated by cutting oil, this problem was addressed through a revision of cleaning procedure to completely remove residual oil from the cooling circuit. These results are expected to improve the quality of SB in subsequent series production and other ITER components with complex cooling circuit structures.

Plasma-Facing Components in the COMPASS Upgrade Tokamak

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

COMPASS Upgrade (COMPASS-U), a new tokamak currently being built at the Institute of Plasma Physics in Prague, is designed to investigate physics gaps in a fusion power plant environment. It will operate at high magnetic field (5T), high plasma current (2 MA) and with its vacuum vessel and all in-vessel components at temperatures up to 500oC, justifying the choice of Inconel for support structures. The plasma-facing components (PFC) will be made of bulk tungsten (W) tiles in high heat flux regions (~5 m2) and Inconel tiles coated with W by physical vapor deposition elsewhere (~5 m2), providing a 100% metallic coverage of the plasma-wall area and making it ITER-relevant after the 2024 rebaselining.

This contribution presents the COMPASS-U PFC (and embedded diagnostics) design and associated challenges of extreme operating conditions combined to a small size. The inner wall, where plasma discharges will be started and ended, features 8 guard limiters made of clamped W tiles with recessed W-coated tiles in between. The tiles are shaped to maximize the wetted area and to account for a possible narrow component in the scrape-off layer profile. There are 10 discrete, narrow outer wall limiters placed in between large equatorial ports, fully covering the poloidal region between the upper and lower stabilization coils (fully covered by W coated tiles). Two limiters are slightly protruding and equipped with W tiles to protect against runaway electrons. COMPASS-U is equipped with a bottom closed divertor optimized for high power exhaust (≤ 70 MW/m2) and an open, flat divertor at the top with W tile shaping allowing operation in both helicity of the field lines. The closed divertor is made of 24 cassettes fixed together creating a stiff ring to handle huge electromagnetic loads. The challenge for the cassettes is twofold, machining such complex objects and ensuring tight tolerances on cassette-to-cassette radial alignment (0.2 mm). The thermal expansion consequent to plasma heating (+6 mm at low field side) is accounted using flexible supports relying on spherical bearings. Additive manufacturing was tested and validated (2 Inconel cassette prototypes were successfully produced), and represents the preferred manufacturing route because of lower costs and higher flexibility. The closed divertor has an ITER-like poloidal cross-section with toroidally beveled tiles (mr = 0.25 mm) to hide leading edges at vertical targets. A divertor manipulator is able to expose e.g., samples or liquid metal modules, to high heat fluxes without breaking the vacuum.

Industrial Scaling up and Mechanical Performance of Wf/W Composites for Divertor Applications

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten fiber–reinforced tungsten (Wf/W) composites have emerged as a highly promising material system for plasma-facing components in nuclear fusion reactors, particularly for divertor applications, due to their potential to overcome the intrinsic brittleness of monolithic tungsten. Among the available fabrication routes, field-assisted sintering technology (FAST) offers a scalable and industrially viable pathway for producing such composites. This work, carried out within the framework of the SMART-FIT project, focuses on the scaling up and optimization of the industrial production of Wf/W using an industrial FAST system, with particular emphasis on microstructural integrity, mechanical performance, and interfacial stability. Comprehensive microstructural and mechanical characterization of the fabricated composites was conducted to evaluate material quality and fracture behavior. Three-point bending tests were used to better understand the fracture response of Wf/W under mechanical loading.

Specification was given to the influence of industrial binders used in tungsten powders and their associated impurities on the toughness and reliability of the composite. Most industrial binders contain carbon, which readily diffuses into tungsten fibers at high sintering temperatures (1800 °C), leading to fiber embrittlement. Even in carbon-free binder systems, oxygen diffusion into the fibers was observed, also resulting in embrittlement and degradation of mechanical performance. To mitigate these effects, two strategies were investigated: (i) performing the de-binding process in a hydrogen atmosphere to eliminate oxygen contamination, and (ii) increasing the thickness of the yttrium oxide (Y_2O_3) interfacial coating beyond the current standard (>5 μm) creating more efficient barrier against oxygen diffusion. Both approaches demonstrated potential for preserving fiber ductility and improving composite toughness. In addition, the influence of tungsten powder particle size on sintered density and interfacial integrity was systematically studied. While the use of nanoscale tungsten particles significantly increased the relative density from 90.74% to 94.25% under comparable sintering conditions, it simultaneously degraded the fiber–matrix interface, leading to interfacial damage and fiber embrittlement. These results highlight a critical trade-off between densification and mechanical reliability.

From Legacy to Leading Edge: Optimizing Hypervaptron's Cooling for Extreme Heat Flux in Fusion Applications

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Since their development in the 1950s, hypervapotrons have been limited to niche applications, such as the JET neutral beam injector scraper, due to rigid design constraints and the absence of accurate predictive models. Yet, their potential for managing extreme heat fluxes makes them ideal candidates for next-generation fusion devices.

To overcome these historical barriers, EUROfusion launched a dedicated project aimed at rethinking hypervapotron modelling, optimisation, fabrication, and performance by manufacturing and testing divertor plasma facing components, with tungsten as plasma facing material and CuCrZr as heat sink.

This paper presents recent advancements in both modelling and fabrication techniques for hypervapotrons. First, we introduce advanced fluid simulations capable of capturing the onset of nucleate boiling, as well as some initial results on two-phase flow simulations. In parallel a simplified approach coupled with advanced design optimization technique is used to define an evolution from the classic hypervapotron design, less sensitive to the boiling phenomenon.
Regarding the fabrication route, we replace traditional manufacturing with Laser Powder Bed Fusion (L-PBF) additive manufacturing of CuCrZr, eliminating long-standing fabrication limits such as precise milling of internal surfaces, smooth transition from tube to rectangular channel, and cracking susceptibility of CuCrZr welding.
Post-processing via Hot Isostatic Pressure treatment is however required, and has two objectives:
• Full densification and restoration of CuCrZr material properties following ITER requirements (yield strength of 150/125 MPa at 20/250°C, tensile strength of 260/200 MPa at 20/250 °C)
• joining of CuCrZr with 316L stainless steel (cooling tube and mechanical interface) and with pure tungsten tiles (plasma-facing material).

Finally, HIPed LPBF CuCrZr 250x32x40 mm representative mock-ups are manufactured following the aforementioned optimized designs, as well as design comparable to mock-ups experimented in the early 2000's. Results are remarkable, showing a thermal efficiency increased by 50% in similar conditions, allowing more heat flux to be extracted for a given maximal temperature (30 MW/m^2 vs 20 MW/m^2 for 500 °C maximal temperature in CuCrZr heat sink).

These results, validated through high heat flux testing, demonstrate the viability of hypervapotrons as high-performance, scalable, thermal management solutions for fusion applications, marking a significant leap forward in both manufacturability and operational efficiency. W/LPBF-CuCrZr/316L mock-ups are under fabrication by HIP and will be tested in the near future to complete these experiments.

Evaluation of the effect of accumulated HHF on tungsten divertor mock-up manufactured by casting and HRP method

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The tungsten (W) monoblock and Cu interlayer were joined by casting, followed by joining of the W/Cu and CuCrZr heat sink using the hot radial pressing (HRP) method, as applied in tungsten divertor manufacturing industries. A gas pressurizing casting technique between the W monoblock and Cu was developed, which successfully eliminated almost all interfacial defects. Small mock-ups were fabricated by HRP with gas-pressure-cast W/Cu monoblocks and CuCrZr cooling tubes, and subjected to high heat flux (HHF) testing at 10 MW/m^2 for up to 5,000 cycles[1, 2].

Two tungsten divertor mock-ups—one with a defect and one without defects in the diffusion bonding layer—were produced by HRP. The defect, located on the opposite side of the heat-facing surface, measured approximately 36 mm^2. Both mock-ups were tested under HHF loading at 10 MW/m^2 for 5,000 cycles to verify robustness, and no abnormalities were observed.

The quality and reliability of the bonding interface between tungsten (W) armor and heat sink materials represent a critical engineering challenge for divertor plasma-facing components repeatedly exposed to high heat flux. To establish a robust W monoblock divertor fabrication process and validate its performance, mock-ups were systematically manufactured and tested under cyclic HHF loading. The test program was designed to evaluate endurance under repeated thermal stresses and to determine the critical heat flux limit of the fabricated structures. The mock-ups underwent HHF testing at 10 MW/m^2 for 5,000 cycles, will be followed by 15 and 20 MW/m^2 for 500 and 300 cycles each. Finally, critical heat flux evaluation will be conducted by stepwise increasing the applied heat flux in 2 MW/m^2 increments, with 100 cycles at each level, until severe damage occurs in the monoblock structure.

This study aims to validate the established fabrication route, identify the operational limits of W monoblock divertors under extreme thermal conditions, and to investigate the tungsten surface property changes at each heat flux stage, thereby contributing to the development of reliable plasma-facing components for future fusion devices.

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[2] Hyoung Chan Kim, et al., Fusion Eng. And Des. 193, 113800 (2023)

Progress in Scalable Manufacturing of Wf/W Composites via Ultra-High Pressure Sintering

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten is a primary plasma-facing material (PFM) in fusion reactors due to its high melting point and sputtering resistance. However, its inherent low-temperature brittleness and embrittlement at elevated temperatures pose significant challenges for reliable operation. To overcome this, tungsten fiber-reinforced tungsten (Wf/W) composites have been developed. These composites utilize extrinsic toughening mechanisms to improve fracture toughness and mechanical resilience while retaining the excellent high-temperature properties of tungsten. Achieving full density is a critical factor determining the thermo-mechanical performance and engineering viability of these composites. Although high-temperature and high-pressure (HTHP) sintering is an innovative technique capable of producing fully dense Wf/W materials, scaling the manufacturing process from laboratory to engineering dimensions remains a major obstacle.

This work details a successful scale-up strategy for fabricating fully dense Wf/W composites, transitioning from millimeter-sized laboratory samples to centimeter-scale engineering components. The microstructure, including the fiber-matrix interface, fiber distribution, and volume fraction, was meticulously characterized alongside the achieved bulk density. The mechanical properties were evaluated through uniaxial compression and three-point bending tests, while thermal conductivity was determined using Laser Flash Analysis (LFA). The results demonstrate a strong positive correlation between the composite density and its key properties: enhanced density directly leads to superior mechanical strength and higher thermal conductivity. Finally, high-heat-flux testing was employed to qualify the material's performance under fusion-relevant conditions, successfully demonstrating its essential "pseudo-ductile" fracture behavior. This integrated study, encompassing scalable manufacturing, comprehensive microstructural and property characterization, and performance validation under extreme thermal loads, provides a critical foundation for advancing Wf/W composites as a robust and feasible PFM for next-generation fusion reactors.

Effect of Powder Size Variation on the Microstructure and High-Temperature Properties of Hetero-structured Ferritic ODS Alloy

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Oxide Dispersion Strengthened (ODS) alloys, which have attracted attention as structural materials for fusion reactors, are strengthened by dispersed fine oxide particles in the matrix and exhibit excellent high-temperature strength and irradiation resistance. However, ODS alloys face a limitation: poor ductility, which makes them vulnerable to brittle fracture. To overcome this Strength-Ductility trade-off, heterogeneous structures have emerged as a promising solution. In a heterogeneous structure, microstructures with different mechanical properties coexist. During deformation, the interaction between soft and hard zones within the microstructure induces Hetero-Deformation Induced (HDI) strengthening, which is known to improve both strength and ductility simultaneously.
In this study, a hetero-structured ferritic ODS alloy was fabricated, in which ODS alloy powder (Fe-15Cr-2W-0.4Ti-0.5Y_2O_3) and Non-ODS alloy powder (Fe-15Cr-2W) coexist. To confirm the effect of powder size variation, mechanical alloying (MA) was performed under different conditions to intentionally introduce a particle size difference between the ODS and Non-ODS powders. The two powders were then mixed at 50 wt.% each, and bulk specimens were produced through spark plasma sintering (SPS), hot rolling, and heat treatment. The effect of powder size variation on the formation of the heterogeneous structure was evaluated by observing cross-sectional microstructures of the fabricated specimens. Subsequently, tensile tests were conducted at room and elevated temperatures to evaluate the mechanical properties. In addition, high-temperature oxidation behavior was examined by oxidation tests using a muffle furnace.

Design-driven technology R&D for the divertor of a fusion power plant: outstanding issues, strategy and progress

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The divertor is the key component for power exhaust in modern magnetic confinement fusion devices. Historically, the engineering efforts were focused on high-heat-flux (HHF) technologies to cope with the severe thermal loads and surface erosion caused by intensive particle bombardment. Even today, HHF technology remains one of the most critical challenges. Consequently, the technology for HHF target has been often regarded as the synonym of divertor technology per se.

The situation began to change with the ITER divertor. Neutron irradiation effects and the thermal impact of slow transients had to be taken into account. Moreover, design of the steel components such as cassette body or target supports required a proper structural verification against codes as a pressure vessel under nuclear environment. These technical peculiarities drove dedicated technology R&D efforts for the steel pressure components as well as HHF targets. Still, the envisaged nuclear loads and cooling conditions of the ITER divertor are modest allowing a considerable safety margin.

When it comes to the divertor of a power plant scale reactor, the design and technology options are subject to various restrictive high level requirements and technical constraints derived from essentially different loading and boundary conditions, which pose a profound impact on design and technology options for the entire divertor system. In the framework of EUROfusion, the 'Divertor' project (WPDIV) has been addressing these issues for over a decade with its comprehensive design and R&D activities taking the European demonstration reactor (EU-DEMO) as reference.

Recently, a substantially revised divertor design was created on the occasion of the new EU-DEMO baseline. The underlying design features were motivated by the high level requirements to ensure technical compatibility with DEMO scale reactors and enhance viability towards power plant applications. Technology gaps were identified and prioritized. Extensive R&D items were derived from the design and implemented into the WPDIV program where dozens of European laboratories and companies are involved.

In this contribution, starting with the unique challenges and outstanding design issues raised specifically for the divertor of a tokamak-type fusion power plant, we present potential power exhaust scenarios, the current status of design study based on the European approach, the latest R&D progress for the entire divertor system, remote maintenance scheme and estimated pumping performance. Finally, the implications for future R&D endeavors are discussed.

Fusion?relevant neutron irradiation of tungsten: hardening, embrittlement, and recrystallization under diverse irradiation conditions

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten is the primary plasma-facing material selected for ITER's divertor and first wall. A set of exceptional properties makes it ideal for withstanding the extreme conditions in a fusion environment. Nevertheless, significant challenges must be addressed. In a fusion power plant, the primary neutron source is the D-T reaction, emitting 14.1 MeV neutrons. These high-energy neutrons will first interact with plasma-facing components and gradually lose energy through successive collisions with structural materials. Neutron irradiation induces microstructural changes such as formation of voids and dislocation loops and precipitation of transmutation products. These changes lead to swelling, exacerbating tungsten's intrinsic brittleness by increasing the ductile-to-brittle transition temperature and degrading thermal conductivity.

Therefore, accurate characterization of tungsten alloys under conditions approximating those expected in operation is essential. However, in the absence of a dedicated fusion-relevant neutron source, irradiation studies rely on fission-based material testing reactors. Irradiation in the BR2 reactor has been one of Europe's assets, with typical irradiation conditions of tungsten-based materials encompassing 400 °C – 1100 °C and 0.1 dpa – 1 dpa of neutron damage dose. This study focuses on the comparison between data from previous irradiation campaigns at the BR2 reactor, and two irradiation campaigns with new conditions never imposed before at BR2.

The first new irradiation campaign used a gadolinium shielding to cut the thermal neutrons of the BR2 spectrum, thus being closer to a fusion dedicated irradiation experiment. Three irradiation temperatures have been selected: 400 °C, 800 °C and 1100 °C, the dose and transmutation rate were 0.84 dpa and 0.5 at. % Re/dpa respectively. Meanwhile, the second irradiation campaign uses the full BR2 neutron spectrum, with irradiation temperatures of 1200 °C and 1500 °C providing a unique glimpse into the recrystallization and irradiation damage interaction. The target dose was 0.23 dpa with a transmutation rate of 2 at. % Re/dpa on this 2nd campaign.

The focus of this work is on the irradiation hardening and embrittlement of pure tungsten and tungsten alloys after irradiation under a diverse set of fusion relevant conditions. Mechanical characterization is performed relying on Vickers hardness measurements and uniaxial tensile testing. Further analysis of the fracture surface is done by scanning electron microscopy, in order to identify fracture features such as cleavage facets, cracks, necking of individual grains, and intergranular fracture patterns as well as recrystallization features such as anomalous equiaxed grains or enlarged grains.

#12901

Helium Gas-Cooling of Divertor with Uni-directional Porous Pipes

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Helium gas cooling has been investigated as one of cooling technologies for divertors in nuclear fusion reactors. For instance, the cooling technologies using HEMJ and metal foams have been proposed in the past. This study discusses the cooling potential of helium gas flow in unidirectional porous copper tubes fabricated by explosive welding technology. First, heat transfer tests using nitrogen gas are conducted, and the effect of flow velocity on the heat transfer coefficient is discussed. Subsequently, considering the change in physical properties of helium gas under pressurized conditions, the helium gas cooling performance using unidirectional porous copper tubes under a 10 MPa pressure condition is predicted. The results demonstrate that under a 10 MPa pressurized condition, the heat transfer coefficient reaches 30,000 W/(m^2·K), proving it possesses sufficient potential as a cooling technology for fusion reactor divertors.

Development progress of plasma facing components upgrade for HL-3 device

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

After the plasma operation of ions and electrons at over 100 million degrees for HL-3 device, its plasma facing components will be upgraded again to satisfy the operational requirements of the deuterium tritium experiments, including the first wall and the divertor. However, carbon-based material is still employed for the plasma facing materials of first walls and divertors, which have strong tritium retention characteristics. Therefore, in order to reduce the adsorption of the material, it is heated to 200 ℃ during the D-T plasma operation. Due to the first wall components without fluid medium channels, the mineral insulated cables are selected as the heating method, and the heating wires are located between the graphite tiles and the back-plates. While the divertors with cooling channels, hot nitrogen gas at 200 ℃ could be used for heating.

Meanwhile, to meet the D-T plasma operation requirements and reduce the peak thermal loads shock of the advanced divertors that located at the lower of the vacuum vessel, the double-null divertor configuration is designed. So the first wall of the upper region of the vacuum vessel will be replaced by divertors, and a maximum heat load of upper divertor would be designed more than 10MW/m2. The material facing the plasma is the same as that of the lower divertor, and it's a closed divertor configuration, which could complements the advantages of the open-type lower divertor. Their engineering design and analysis of the above plasma facing components have been finished, and are currently in the manufacturing stage.

Ref.
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[2]Y. Lu, et al. Fusion Engineering and Design, 195(2023)113963.
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Improving the Sinterability of Wf/W by Alloying

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten fiber-reinforced tungsten (Wf/W) composites are considered promising candidates for plasma-facing and structural components in future fusion reactors due to their extremely high melting point and intrinsic pseudo-ductile fracture behavior introduced by fiber reinforcement. However, the practical application of Wf/W composites is still limited by the difficulty in achieving sufficient matrix densification without compromising the ductility of tungsten fibers. Under conventional high-temperature sintering conditions, this challenge is particularly significant, as excessive thermal exposure often promotes fiber recrystallization and leads to embrittlement. By alloying tungsten with specific elements such as vanadium (V), a potential strategy can be developed to effectively enhance the sinterability of tungsten-based composites while preserving their excellent mechanical properties.

In this work, vanadium was incorporated into the tungsten matrix to enhance the densification of Wf/W composites while maintaining the inherent ductility of the tungsten fibers. The W-V alloy powders with V contents ranging from 0 to 4 wt.% were synthesized via mechanical alloying and subsequently consolidated using field-assisted sintering technology (FAST/SPS). The effects of V content, sintering temperature, and holding time on densification behavior, microstructural evolution, elemental diffusion, and fracture behavior were systematically investigated. Relative density was determined using the Archimedes method. Microstructural characterization was carried out by scanning electron microscopy combined with energy-dispersive X-ray spectroscopy (SEM/EDX), with particular emphasis on matrix morphology and vanadium diffusion from the matrix into the fibers. The mechanical properties and fracture behavior of the composites were further evaluated through three-point bending tests combined with fracture surface observations.

Density measurements indicate that appropriate vanadium addition significantly enhances matrix densification, enabling a reduction in sintering temperature and holding time while achieving a maximum relative density of approximately 98%. Three-point bending tests reveal that under optimized parameters, Wf/W-V composites exhibit significant plastic deformation and stable load-bearing capacity, reflecting effective load transfer between the matrix and the fibers. Fracture surface observations confirm predominantly ductile fiber fracture, characterized by extensive fiber pull-out and plastic deformation. Conversely, higher sintering temperatures or longer holding times promote the diffusion of vanadium into the tungsten fibers, leading to a significant shift in fracture behavior toward brittle fracture.

Overall, this study reveals a synergistic strategy that successfully overcomes the trade-off between densification and ductility in tungsten-based composites by combining controllable W-V alloying with optimized FAST/SPS parameters. The identified process window enhances matrix densification without sacrificing fiber ductility, providing practical guidance for manufacturing reliable tungsten-based composites suitable for nuclear fusion-related applications.

Design-by-Analysis of Tungsten Fibre-Reinforced Tungsten for Divertor Plasma-Facing Components

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten is the baseline armor material for divertor plasma-facing components (PFCs) in fusion devices; however, its intrinsic brittleness, low fracture toughness, and susceptibility to cracking under cyclic high heat flux loading limit component lifetime. Tungsten fiber-reinforced tungsten (WfW) composites have been proposed as an advanced armor concept with improved damage tolerance and pseudo-ductile behavior. This work applies a multiscale modelling framework to evaluate the suitability of WfW as an alternative divertor PFC material.

At the microscale, WfW is represented using homogenized material models to estimate effective elastic and thermal properties and damage initiation behavior. Standard homogenization techniques are applied based on constituent properties, accounting for the influence of the yttria-based fiber coating on thermal conductivity and the effect of matrix porosity, which significantly reduces mechanical strength compared to fully dense tungsten. The resulting orthotropic properties show reduced directional strength relative to monolithic tungsten, indicating that WfW requires design-by-analysis rather than direct substitution.

These effective properties are transferred to the mesoscale, where the armor is modeled as a layered composite. This enables systematic variation of fiber stacking sequences and orientations, allowing tailoring of anisotropic thermal transport and stress redistribution through composite architecture.

At the macroscale, WfW configurations are implemented in finite-element models. Sequential thermal analyses include manufacturing cooldown, coolant operating conditions, and steady-state high heat flux loading. For an ITER-type monoblock geometry, replacing tungsten armor with WfW increases peak temperatures by up to 300 °C at 15 MW compared to solid tungsten. High heat flux experiments on monoblocks with [0/90] stacking confirm this trend. The temperature rise is primarily attributed to matrix porosity and reduced transverse thermal conductivity caused by the Y_2O_3 fiber coating.

Mechanical analyses show that failure is governed by transverse tensile stresses, strongly influenced by fiber orientation, as transverse strength is significantly lower than axial strength. These findings motivate alternative PFC concepts, such as hypervapotron and microchannel flat-tile geometries, where reduced stiffness and fiber bridging may enhance joint strength and mitigate thermally induced stresses. The presented framework supports optimization of such designs and guides future high heat flux validation experiments. These alternate designs will be analyzed and optimized using parametric modeling in order to exploit the strength of

WfW material with the end goal of testing an optimal design at high heat flux test facility.

#12968

CHF and Boiling HTC Enhancement in One-Sided Heated Cooling Channels Using Structured Inserts and Fin-Shaped for Fusion Divertor Applications

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Fusion plasma-facing components (PFCs) are exposed to severe one-sided heat fluxes approaching ~20 MW/m^2, making simultaneous enhancement of critical heat flux (CHF) and boiling heat transfer coefficient (HTC) essential for reliable cooling design. This work investigates one-sided heated subcooled flow boiling in cooling channels incorporating manufacturable structured inserts and fin-shaped internal features, and establishes a comparative database for fusion-relevant screening and model input.

Both circular and rectangular geometries are considered. Circular tubes are tested with swirl-type structured inserts, including single swirl tape, double swirl tape, V-groove swirl tape, and an airfoil-type swirl tape. Rectangular channels are evaluated with fin and groove-based structures, including an HV 45° fin, V-groove, and airfoil-type configurations. These concepts are selected to intensify near-wall mixing, increase liquid renewal at the heated surface, and reduce vapor accumulation in the near-wall region mechanisms that are particularly important for delaying local dryout under one-sided heating. In addition, the inserts are assessed from a practical standpoint, focusing on geometric simplicity and applicability to PFC cooling-channel fabrication.

Experiments are conducted in a subcooled boiling loop equipped with a one-sided Joule-heated test section capable of supplying heat fluxes up to 10 MW/m^2. Heat flux is increased stepwise until CHF occurs, while the effects of inlet subcooling, system pressure, and mass flux are quantified. Boiling regimes spanning partial nucleate boiling to fully developed nucleate boiling are identified using surface temperature response, and boiling HTC is extracted over the full range of heat-flux conditions prior to CHF. The resulting CHF/HTC dataset provides a consistent basis to rank enhancement concepts under one-sided heating and supplies experimentally verified thermal performance inputs for subsequent PFC thermal analyses, improving wall temperature prediction and design-margin assessment under high heat flux operation.

CFD Simulation of Liquid Metal Heat Pipes as Plasma-Facing Component for Fusion Applications

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Managing asymmetric transient and steady-state thermal loads on plasma-facing components (PFC) remains a significant challenge in magnetic confinement fusion applications. Conventional cooling concepts, such as monoblock chain, utilize copper cooling channels with coolant flow beneath the tungsten armour. However, this configuration is limited by the relatively small effective heat transfer area and restricted heat removal capability associated with geometric limitations. Improving thermal spreading efficiency through increased cooling surface and extremely high thermal conductivity makes liquid metal heat pipes (LMHP) a promising alternative candidate for the conventional cooling system. However, a qualified LMHP specifically designed for fusion applications has not yet been fully developed. In the framework of a research project at Forschungszentrum Jülich [1], a plasma-facing component concept based on LMHP is being developed. The proposed LMHP must withstand the relevant conditions, including extreme heat fluxes up to 20 MW/m2 and inclined orientations, therefore, design parameters of LMHPs – such as material, diameter, length, wick properties, and filling ratio – must be carefully tuned and optimized. Iterative experimental investigations on optimization of LMHPs are costly and time-consuming due to complex manufacturing process. In this work, a design-by-analysis approach is proposed for an LMHP-based PFC integrated with a secondary cooling system with the focus on the design and numerical analysis of the LMHP. Although analytical models for conventional heat pipes operational limitations are available in the literature, their validity and applicability to high-temperature LMHPs under extreme conditions has not been established. Existing analytical models for the LMHPs are mostly limited to specific materials and configurations, limiting their direct applicability for conceptualizing a LMHP-PFC. Therefore, a two-dimensional thermal resistance model is developed to determine the temperature distribution on the evaporator and condenser regions and calculate the thermal conductivity of the LMHP under thermal loading of 20 MW/m2. Moreover, computational fluid dynamics (CFD) simulations with volume of fluid (VOF) method have been conducted to investigate the effect of the wick porosity, filling ratios and inclination angles on LMHP thermal performance. The simulations methodology is applied to a commercially available stainless-steel-sodium heat pipe. Finally, experimental investigations are performed, and the results are used to validate the numerical models. The LMHP design, developed based on the simulation results, will be used in future manufacturing processes.
[1] Y. Mao, et. al., Integrated Development of High-Temperature Heat Pipe-Based Plasma-Facing Components (HPPFC): Modeling, Fabrication and Experimental Validation, 34th Symposium on Fusion Technology (abstract submitted; under review), 2026

High temperature irradiation of diamond and thermal conductivity measurements for applications in fusion plasma facing components

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Carbon was discarded as plasma-facing material in fusion for its high level of erosion and tritium retention. However, carbon as Chemical Vapor Deposition (CVD) diamond has low direct retention, chemical erosion and unlike bulk tungsten (W), has outstanding performances under thermal shocks [de Temmerman NF 2011] because of its very high thermal conductivity. Unfortunately, its physical erosion properties similar as graphite [de Temmerman Phys. Scr. 2009].

For this reason, in a previous study [C. Guillemaut JNM 2026], tungsten-diamond (W-diamond) test samples made of a ~mm thick CVD diamond substrate for a uniform and fast distribution of heat with a ~10 µm W surface coating for protection against erosion were manufactured and tested under high heat flux. Post-mortem analysis showed intact coatings and metal-diamond interfaces for all test samples under:

- up to 104 cycles of 1 ms at 1.7-2 GW.m-2 (~32-95 MW.m-2.s0.5) to simulate ITER and DEMO-like fast transients such as edge-localized modes, disruptions and runaway electrons,
- up to 20 cycles of 1 s at 100 MW.m-2 to simulate ITER reattachment of the divertor plasma.

Despite these encouraging results, irradiation studies on ITER CVD diamond windows [R. Heidinger FED 2001] showed a strong degradation of diamond thermal conductivity after a fluence of 1022 m-2 of fission neutrons (energy < 0.1 MeV) at room temperature.

We have simulated the effect of 14 MeV neutrons after a fluence of 1022 m-2 on CVD diamond (0.0352 dpa of irradiation damage) using a bombardment of 6.8 MeV carbon ions with a maximum fluence of 4.2x1019 m-2 at room temperature, 400°C and 800°C at the 2MV ARAMIS accelerator at IJCLab, Orsay. The thermal conductivity of the irradiated region (at 3 µm in depth) of the three irradiated samples will be measured at the LOMA, CNRS, University of Bordeaux using a combination of the Time Domain Thermoreflectance (TDTR) and the Frequency Domain Thermoreflectance (FDTR) methods [H. Olson JAP 2019].

At this stage, visual inspection of the exposed samples reveals strong differences in color with a completely black sample after irradiation at room temperature, a yellowish sample with some transparence after irradiation at 400°C and a transparent sample after irradiation at 800°C. This suggests a significant reduction of the impact of irradiation with temperature. This is likely related to the self-repairing capabilities of the material through annealing processes. The TDTRT/FDTR measurements will allow quantifying this effect on the thermal conductivity.

Qualification of the HELCZA facility for HHF testing of tungsten divertor components

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The electron beam based High Heat Flux (HHF) test facility HELCZA, is entering a new phase: the testing of tungsten components. In recent years, the HELCZA facility has focused on an extensive test campaign involving beryllium Full-Scale Prototypes of ITER First Wall and Alternative Design Mock-Ups. This campaign rigorously evaluated the performance of the HELCZA facility in conducting fatigue testing with heat fluxes reaching up to 3 MW/m^2, demonstrating the facility's suitability for such demanding testing scenarios.

To enable the testing of tungsten components while avoiding the health and safety concerns associated with beryllium, a comprehensive decontamination process was performed.

Recently, the facility underwent a successful qualification to test the ITER divertor component, the Inner Vertical Target (IVT). During the qualification, the IVT Test Assemblies (TA) and Plasma Facing Units (PFU) composed of tungsten monoblocks are subjected to heat fluxes reaching up to 20 MW/m^2, with monoblock surface temperatures locally exceeding 2000 °C. This qualification paves the way for HHF testing campaigns of tungsten divertor components for ITER and other fusion reactor facilities at the HELCZA facility in the coming years.

STEP Fusion Power Plant First Wall Heat Load Challenges

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The Spherical Tokamak for Energy Production (STEP) fusion programme aims to achieve net electricity production from fusion through the STEP prototype power plant (SPP) targeting 2040s [2]. A major challenge in attaining fusion power in the STEP fusion power plant is to keep the heat loads to the first wall within engineering limits while delivering net energy, achieving tritium self-sufficiency, and ensuring component lifetimes are compatible with future commercial deployment. As part of the STEP fusion power plant concept design, it is essential to understand the expected heat loads on the first wall, limiters and divertors. The plasma facing components (PFCs) must be designed to withstand these heat loads across all operational scenarios, including steady-state and both normal and off-normal transients.

The work summarises initial heat loads due to radiation [4], thermal charged particles [1], fast particles [6], charge exchange neutrals [5] and neutronics [3], based on load cases from steady-state plasma scenario and vertical displacement events (VDE). The work presents initial development of first wall shape, progressing from a 2D axisymmetric first wall with initial limiter shapes to 3D configuration incorporating discrete limiters to assess first wall protection strategies. At this stage of the STEP fusion programme, not all load cases or heat load components have been analysed. Current assessment covers normal steady-state (flat-top) operation and selected off normal scenarios such as VDE and mitigated disruptions.

The presentation outlines the main assumptions, methodology and a proposed pathway to the heat load specification for the STEP fusion power plant. The preliminary results are presented for PFC design, including peak and average heat flux values that informs power generation and coolant selection. The presentation will also outline further work required, enabling engineers, scientists and researchers to engage with the programme and contribute to addressing identified gaps.

Integrated Development of High-Temperature Heat Pipe-Based Plasma-Facing Components (HPPFC): Modeling, Fabrication and Experimental Validation

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Future fusion reactors require plasma-facing components (PFCs) capable of withstanding extreme asymmetric heat loads, particularly in the divertor region. Conventional monoblock designs with embedded cooling channels in tungsten armor are limited by restricted heat transfer areas, resulting in high local heat flux densities and pronounced thermal gradients. For next-generation fusion power plants with increased thermal demands, alternative heat exhaust concepts with enhanced thermal spreading are desirable. To address this, a plasma-facing component concept based on high-temperature liquid metal heat pipes (HPPFC) is being developed at Forschungszentrum Jülich. The application of heat pipes enables rapid heat extraction and redistribution over a larger area, reducing local hot spots and decreasing heat flux density toward the coolant. This heat radiator concept significantly improves the heat exhaust efficiency and reduces the material challenge.

This contribution reports the integrated progress of the HPPFC project, covering analytical modeling, CFD simulation, tungsten heat pipe fabrication, and preparation for high heat flux validation. A thermal resistance model for a purely radiation-based HPPFC configuration showed that at 20 MW/m^2, lithium sustains vapor temperatures up to 1400°C. Maintaining armor temperatures below 2300°C ensures operation approximately 300°C below the boiling limit. The inclusion of conductive pathways significantly reduces the operating temperature of the heat pipe. To refine the design, CFD simulations incorporating porous media flow and phase change modeling are being conducted to evaluate wick porosity, filling ratio, inclination angle, and evaporator behavior. In parallel, a secondary cooling structure has been designed and analyzed to assess thermal interaction, optimize inter-pipe spacing, ensure adequate coolant distribution, and evaluate boiling onset under high heat flux conditions.

Commercial stainless steel/sodium heat pipes have been procured, and a first working HPPFC prototype based on these units has been developed for experimental validation. The JUDITH-2 electron beam facility is being adapted to enable dedicated high heat flux testing, providing validation data for model calibration prior to finalizing the tungsten heat pipe design.

On the materials side, pure tungsten tubes have been fabricated using CVD sealing and powder metallurgical routes, achieving densities above 95% with good bonding integrity and leak tightness. Porous and grooved tungsten wick structures have been sintered and integrated into tungsten tubes, and capillary pumping performance has been experimentally verified. Commercial tungsten and

tungsten alloy blind tubes have been acquired for prototype development, and the first fully integrated tungsten heat pipe is currently under preparation.

Thermal shock performance of tungsten fabricated by laser powder bed fusion

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Plasma-facing materials in future fusion reactors must withstand extreme operating conditions, including transient heat loads from edge-localized modes and plasma particle fluxes. To meet these demands, various manufacturing routes are being explored to ensure the reliable performance of tungsten as the leading plasma-facing material. In this work, tungsten fabricated through laser powder bed fusion was tested in the PSI-2 linear plasma device under ITER-relevant heat and plasma loading conditions. The material's performance was assessed against ITER-grade tungsten using surface roughness measurements and surface damage mapping. Under low pulse number tests (100 and 1000 pulses) at power densities of 0.19 and 0.38 GWm-2, all samples exhibited good performance. After exposure to a single high-power pulse of 1.6 GWm-2, the ITER-grade tungsten reference showed the lowest roughness, followed by high- and low-density additively manufactured tungsten samples. In contrast, all additively manufactured samples outperformed the ITER-grade tungsten reference in high pulse number tests (10000 pulses) at power densities of 0.4 and 0.8 GWm-2, exhibiting a surface roughness up to almost six times lower. These findings are promising since additively manufactured tungsten maintains performance despite the presence of pre-existing cracks, supporting its continued development as a potential plasma-facing material for future fusion reactors.

Advanced strategy on assessing thermo-hydraulic performance of ITER Divertor High Heat Flux Plasma-Facing Components

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The design of the ITER actively-cooled Plasma Facing Components (PFC), including the Blanket First Wall as well as the Divertor Dome, Vertical Targets and Baffles, is driven mostly by aspects related to their thermo-hydraulic performance against high plasma heat fluxes. Without minimizing the importance of other key design drivers, such as electromagnetic and neutronic loads, the design choices for the cooling channels, hydraulic parameters, or PFC materials are all strongly dominated by: first, high heat fluxes from the plasma; secondly, the thermo-hydraulic performance i.e. the required large Heat Transfer Coefficients (HTC) and margins to Critical Heat Flux (CHF) events.

Due to the complexity of the physics involved on the highly turbulent hydraulic flows in actively-cooled PFC designs, the use of correlations fitted with experimental data is key. All over the last three decades, many high heat flux testing facilities exposed PFC units (PFUs) to relevant heat fluxes magnitudes (from about 1 MW/m2 up to values as high as 20MW/m2) providing confidence on correlations and design robustness, but still not fully representing the actual expected thermo-hydraulic operational conditions in ITER machine.

The proposed paper describes a methodology for predicting by analysis how the PFUs would thermo-hydraulically perform under heat load distributions representative of expected ITER operational scenarios. This is done with confirmed correlations from experimental data and heat load profiles relevant for operation. The study is applied to the particular case of the ITER Divertor Vertical Targets, but can be extended to any other PFC, as it looks in a comprehensive way how the HTC and the CHF margin evolve along the cooling tube while axially diffusing heat across different monoblocs exposed to different heat load levels. The proposed methodology integrates Finite Element results into an analytical model capable to extract key thermo-hydraulic parameters at each cooling section, providing a powerful tool to assess in detail complex heat flux profiles under different plasma operational conditions. The results are expected to provide a more realistic view of the PFU operational performance without unnecessarily challenging a design that has been developed under a reasonably conservative and robust criterion.

Development and Optimization of Mechanically Alloyed ODS Copper

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Oxide dispersion strengthened (ODS) copper alloys are considered as promising heat sink materials for a long-lived EU-DEMO divertor. Although ODS copper has demonstrated its great potential under neutron irradiation, the alloy requires further development until its application. The manufacturing process of ODS copper alloys is based on a powder metallurgical route, utilizing mechanical alloying as the central process step. Since the properties of ODS alloys are highly sensitive to the alloying parameters, a systematic optimization is key. In our study, mechanical alloying was performed using a Simoloyer high-energy attritor mill. We investigated how three key parameters—milling duration, oxide volume fraction, and rotor speed—affected the microstructure, grain structure, and oxide particle distribution. Milling durations longer than one hour resulted in a bimodal grain structure, consisting of grains in the micrometer regime and extensive areas of nanoscale grains. Previous studies on 9Cr-ODS steels show a stabilization of the grain structure beyond a critical particle density. Following this result, ODS copper alloys with varying oxide volume fractions were produced to investigate whether this behaviour is reproducible for ODS copper alloys. Additionally, the impact of oxide volume fraction on oxide particle distribution was evaluated. The rotor speed, as the third key parameter, was assessed regarding its impact on bimodal grain structure formation and oxide particle distribution.

Monoblock Surface Cracking: Impact of Divertor Topology on Plasma Heat Loading & Tungsten Evolution in WEST

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Under a framework agreement between the CEA, the ITER Organization (IO), and the European Domestic Agency, Fusion for Energy (F4E), the "WEST Joint Team" (JT) have installed 20 ITER-like Plasma-Facing Units (PFUs) from ITER's European manufacturers in the WEST tokamak. These actively cooled PFUs, made from tungsten Monoblocks (MBs) bonded to a CuCrZr cooling tube, are installed on one (Q1B) of the twelve sectors of the lower divertor where each sector contains 38 PFUs. In ITER, heat loading of MBs is expected up to 10-20 MW/m2 [1], hence understanding the behaviour of the PFUs under the tokamak environment is essential for proper assessment of divertor component lifetime. This study presents how variations in wetted area and handled power correlates with the extent of a micro-crack network observed on the top surface of PFUs in all sectors installed in WEST.

The wetted area and heat flux were estimated per MB for the Q1B Sector using field-line tracing simulations (PFCFlux) based on the as-installed topology of the sector. Results show that dimensional deviations of the PFU geometry and installation positioning can induce up to ~30% variation in surface thermal loading through a combination of allowable bevel tilt and step deviations. Analysis performed on the EU-PFUs, shows a maximum of 1100W is handled per monoblock and that simulated temperature profiles, using PFCflux and ANSYS are comparable to infrared temperature measurements made on the EU PFUs in WEST. Together these results provide good confidence in both simulation and infrared surface measurement in a metallic environment.

The crack networks, observed on the PFU top surface just below the outer strike point after WEST exposure [2], was quantified from images collected during WEST shutdowns. The extent of cracking on 20 MBs on the EU PFUs correlates strongly with predicted power loading above ~800 W suggesting that cracking initiates above this threshold and increases proportionally with absorbed power. However, the absence of cracking on MBs in higher thermally loaded regions indicate that thermal strains alone cannot explain damage onset. This implies that additional plasma material interaction mechanisms (cyclic fatigue, transient loading, ductile to brittle transitions, hydrogen/helium uptake, microstructural changes…) likely contribute to crack initiation and evolution [3, 4]
[1] J. P. Gunn et al., Nucl. Fusion, vol. 57, no. 4, p. 046025, Mar. 2017
[2] Y. Corre, IAEA FEC 2025
[3] Y. Corre, 2026, submitted to Nuclear Fusion
[4] A. Durif, Proposed Ccommunication, PSI conference, 2026

The technological impact of light ions impurities in the oriented microstructure of tungsten plasma-facing components

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

It is well-established that materials intended for fusion reactors must exhibit optimal structural and functional properties to withstand the extreme operational conditions, including intense heat fluxes, high-energy particle interaction, and plasma exposure. Tungsten, with its exceptional properties such as high melting point, low sputtering yield and tritium retention, and superior thermal conductivity, has emerged as the leading candidate for plasma-facing components (PFCs). However, the interaction of light species like hydrogen isotopes (H/D/T) and helium—introduced by plasma implantation and, for H/He, also generated in-situ via neutron-induced transmutation in W— with tungsten's microstructure remains challenging, as these elements can drive significant diffusion, trapping at defects, and release behaviors that compromise the material's long-term performance, for example, by embrittlement. In the broader fusion context, where neutron irradiation simultaneously produces He/H and progressively transmutes W into Re/Os-bearing alloys, establishing microstructure–sink relationships (grain-boundary/dislocation sink strength versus trapping/release kinetics) is crucial for extrapolating implantation-based observations toward reactor-relevant conditions.

While prior studies by our group emphasized deuterium quantification and ion depth profiling on tungsten using spectrometry and ion beam analysis (IBA) techniques, this research complements those efforts by incorporating two key factors: varying host matrix morphology and, helium contribution. Samples of a polycrystalline, fusion grade tungsten (Iter-like) were subjected to consecutive, high energy He and D ion implantation, achieving ~2 μm penetration depths and fluences of 10^{16} at/cm^2 to replicate PFC damage profiles. Advanced characterization techniques were then employed to assess species detection, migration phenomena (e.g., channeling and release), and microstructural evolution. Given the initial microstructure of W, composed of long columnar grains perpendicular to the surface, the effect of structures' orientation on the diffusion and retention of the implanted light species has been examined. The results provide conclusions on the formation of defects due to the presence of one or the other element, and explanation is offered for expected changes in the functional properties due to the combined presence of the light species.

STUDY OF COMPRESSIVE STRENGTH OF ADDITIVELY MANUFACTURED TUNGSTEN-BASED MATERIALS FOR FUSION APPLICATIONS

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Plasma-facing components (PFC) of fusion reactors are exposed to high thermal loads during normal operation and to extreme loads during transients. These thermal loads can induce significant mechanical stresses, including compressive stresses stemming from steep temperature gradients and thermal expansion constraints. Tungsten (W) is considered a primary candidate material for PFCs due to its exceptionally high melting point, good thermal conductivity, and resistance to sputtering. However, its inherent brittleness and limited ductility suggest development of alternative tungsten-based materials, such as tungsten–copper (W–Cu) composites. Advanced manufacturing processes such as additive manufacturing, specifically selective laser melting (SLM), are a promising method for producing components with the desired structure.

In this study, the mechanical behaviour of pure tungsten and W–Cu composite materials, produced by SLM, was investigated in a temperature range from room temperature up to 800 °C. The materials were tested under uniaxial compression to simulate loading conditions relevant to fusion reactor PFCs subjected to thermal loads. Stress–strain response, compressive strength, and deformation behaviour were evaluated as a function of temperature, material composition and sample build orientation in relation to applied load. Post-test microstructural and fractographic analyses were performed using scanning electron microscopy (SEM).

All tested materials showed a strong dependence of compressive strength on testing temperature, which manifested itself in a decrease in strength with increasing temperature. Samples with grain orientation parallel to the direction of loading withstood higher compressive loads, demonstrating the influence of structural anisotropy. The influence of material composition and manufacturing method on compressive strength is of fundamental importance for potential use in future fusion devices.

Keywords: tungsten-based materials, compressive testing, W–Cu composites, fusion reactors, plasma-facing components

Design challenges for the COMPASS Upgrade closed tungsten divertor

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

COMPASS Upgrade (COMPASS-U) represents a new full metal tokamak featuring tungsten (W) plasma-facing components, currently under construction at the Institute of Plasma Physics in Prague, Czech Republic [1]. The lower closed W divertor should withstand the expected substantial constraints such as high electromagnetic loads (Bt = 5 T, Ip = 2 MA) and high energy flux density at targets (≤70 MW/m2). The large thermal expansion that is expected during plasma discharges combined with its relatively small size (R = 0.9 m, a = 0.27 m), together with tight tolerances and a foreseen operation up to 500°C, makes it challenging to design. In the frame of an Eurofusion Engineering Grant, the COMPASS-U closed divertor final design is presented.

The COMPASS-U closed divertor is based on a cassette concept forming a stiff ring to handle extreme thermal and electromagnetic conditions while fitting into a limited space. The cassettes machining is difficult due their complex and different shapes (due to diagnostics openings) but keeping the needed tight tolerances on their radial alignment (0.2 mm) is a real challenge. Additive manufacturing, which offers flexibility and lower costs, was investigated and validated by successfully producing 2 Inconel 718 cassette prototypes with no defects and good mechanical properties. The divertor has an ITER-like shape and includes toroidally shaped bulk tungsten tiles to hide leading edges, halo current straps to extract up to 190 kA each and thermal anchors connected to the baking pipes for passive cooling between plasma discharges. The challenge for the thermal anchors was to ensure a good thermal conductivity whilst being electrically insulated. To comply with the thermal expansion (+6 mm at low field side), the divertor ring sits on 8 flexible supports relying on spherical bearings. A special jig was designed and constructed for in-house cyclic testing, showing good performances of the bearings under all operational scenarios at room temperature and at 500°C. Finally, a full scale 3D printed plastic mockup of the whole divertor was built to ensure correct assembly, to test diagnostics interfaces and to gain experience in the metrology verification for final installation. This divertor will support the COMPASS-U mission to close gaps in the particle and power exhaust physics, contributing to the EUROfusion roadmap and helping the design of future fusion devices.

[1] P. Vondracek, et al., Fusion Engineering and Design 169 (2021)

Nuclear fusion reactor benchmark models for testing post processing scripts that follow the elastic RCC-MRx structural integrity assessment rules

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Stress analysts within the EUROfusion consortium Divertor Work Package have been developing a finite element analysis post processing script that performs structural integrity assessment under monotonic and cyclic loading following the RCC MRx 2022 code Design by Analysis elastic approach. Most of the RCC MRx rules have been implemented within the post processing script. The script encompasses the best practice of how to perform an elastic structural integrity assessment while reducing errors that may arise from the side of the stress analyst. In order to verify the postprocessing script two benchmark finite element models are now being developed. One of the models is an axisymmetric thick cylinder under the action of internal pressure and temperature gradient across its wall thickness. The other model is more complex and is developed to include most of the geometric features and all the anticipated load cases of a box type divertor suitable for a fusion power plant. This benchmark model is called the Numerical Divertor mock up model and in essence has been designed like a heat exchanger under the action of heat flux, deadweight, cooling water pressure, electromagnetic loads due to plasma disruptions and seismic loads. In order to verify the postprocessing script the benchmark models would require less time and would be more versatile than any prototype divertor model currently being developed. The benchmark models can also be used for training purposes on how to use the post processing script and the RCC MRx code of standard. The benchmark models can be adapted relatively easily to test various functions of the post processing script, as this is further developed. The benchmark models can also be used to develop efficient methodologies of how to apply dynamic loads to fusion reactor components. Such dynamic loads mainly arise from plasma disruptions that induce electromagnetic Lorentz forces. In addition, the benchmark models can be used to efficiently check on the conservatism that the elastic approach imposes on the design, when compared to the elastoplastic approach.

#13089

STEP Fusion Powerplant Off Normal Event Handling Impacts on PFCs

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The Plasma Facing Components (PFCs) for the STEP Fusion Powerplant will be subject to high heat loads during operation. The PFCs consist of several components located at different locations within VIVS (Vacuum vessel and In Vessel Systems) including the inboard and outboard first wall, divertor PFCs and Limiters. An overview of the current status of the VIVS PFCs designs will be detailed. Due to uncertainties in the operations of the STEP Fusion Powerplant, in addition to these loads, there will be off normal events which place additional loads, including heat, mechanical and electromagnetic loads on the PFCs. These additional loads, in particular the high heat loads during an off normal event, will interact with the PFCs and in turn produce additional effects, a notable one of which is secondary Tungsten plasmas.

STEP Fusion Powerplant's design includes a radial port-maintained limiter concept in which a section of first wall is located where transients are expected to land in off normal events and, in the event of damage, be replaceable without entering into a major maintenance cycle.

The STEP Fusion Powerplant limiter design features a plasma facing component and, during a high energy transient event, such as a disruption, direct contact between the plasma and the limiter can lead to the PFC's material melting and vapourising. Understanding the effects of this interaction, including vapour shielding and secondary radiation, is vital to designing a limiter capable of performing its functions.

To facilitate design of the limiters and their integration, the effects caused by these off normal events, and interactions with surrounding components, must be understood along with understanding the integration challenges and trade offs with other functions and subsystems of the tokamak including with the maintenance strategy, power cycle and infrastructure and breeder blanket design.

Here we will present an overview of the considerations for the STEP Fusion Powerplant limiter design including a summary of off normal events, the secondary effects that may occur, the integration considerations and future plans.

Thermal and structural analyses of the Tungsten based divertor for Wendelstein 7-X

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The stellarator Wendelstein 7-X (W7-X) has started experiments with a water-cooled CFC divertor towards a long pulse operation. Meanwhile, a tungsten based divertor concept is under development as an alternative option to the CFC module for a reactor relevant environment. According to the experience of previous experiments, the W-based divertor has an improved geometry to accommodate the plasma heat loads and new manufacturing technologies like additive manufacturing. Advanced W alloy and functionally graded materials are also considered for better thermal and structural performance. Therefore, thermal and structural analyses have been performed to find the impact of these variants.

The analysis is performed in the normal W7-X operation scenario, which includes a steady-state peak heat flux of 10 MW/m2 and a peak coolant pressure of 25 bar. The efforts consist of three parts: 1) to optimize the cooling channel structure, parametric models are employed to study the influence of the cooling channel shape, dimension and coolant velocity; 2) a rounding edge of the target module in the toroidal direction is proposed to minimize the leading edge effect, this paper also shows it has thermal and structural advantage to the straight edge design; 3) a functionally graded interlayer will realize a smooth transition from the W coating to the CuCrZr heat sink, which could reduce the thermal stress greatly than the OFE-Cu one, and the analysis gives a quantitative comparison. In the end, thermal-structural calculationed and assessments are performed on an A4-size prototype designed according to the aforementioned efforts.

Keywords: Wendelstein 7-X, stellarator, divertor, thermal analysis, structural analysis

Modeling of Two-Phase Subcooled Flow Nucleate Boiling under Divertor Relevant Flow Conditions

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Divertor targets in future fusion reactors are subjected to extreme surface heat fluxes reaching up to 40 MW m^{-2} under high mass-flux cooling conditions. Under such thermal loads, subcooled flow boiling may develop in the cooling channels and strongly influence heat transfer mechanisms. Accurate prediction of the nucleate boiling heat flux contribution is essential not only for reliable wall temperature estimation, but also for dependable assessment of critical heat flux (CHF), which represents a limiting thermal condition for safe divertor target operation. Although dedicated subcooled flow boiling and CHF correlations have been developed for specific geometries such as one-sided swirl-tape configurations, future divertor target cooling channels may be expected to evolve in both shape and size due to ongoing design iterations and emerging manufacturing approaches. In such a context, geometry-specific empirical correlations become insufficient as predictive design tools. Instead, physically based wall boiling models that can be integrated into computational fluid dynamics (CFD) solvers are required to enable geometry-resolved thermo-hydraulic assessments. This work investigates the application of a new generation wall boiling framework, namely the Massachusetts Institute of Technology Boiling (MITB) model, under fusion-relevant conditions. A new two-phase Euler-Euler framework CFD model is developed that implements the MITB model in OpenFOAM, enabling flexible integration of the respective closure models. The results demonstrate that variations in sub-models can significantly affect predicted wall temperatures, heat transfer coefficients, and the overall thermo-hydraulic response. The MITB model showed improved predictive capability compared to generic wall boiling approaches such as the Rensselaer Polytechnic Institute (RPI) model, yielding more accurate temperature predictions under fusion-relevant conditions. Clear differences are observed relative to single-phase predictions, underscoring the need for dedicated two-phase modeling under fusion-relevant conditions. The simulations further show that, under two-phase conditions, the wall heat flux becomes more spatially concentrated. This redistribution of heat flux leads to earlier detection of CHF conditions. Consequently, relying on single-phase assumptions can result in non-conservative CHF margin estimations.

Hybrid additive manufacturing of tailored tungsten preforms on tungsten armor tiles for plasma-facing component application

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Plasma-facing components (PFCs) in magnetic confinement nuclear fusion reactors must withstand high heat flux (HHF) loads and intense neutron irradiation during operation. Established divertor PFC designs comprise a monolithic tungsten (W) armor and a heat sink made from copper (Cu) alloys. The so-called monoblock design is a solid W block joined to a Cu alloy cooling pipe. It has demonstrated high damage resilience and acceptable heat removal capability, but is not readily scalable due to the need to align a very large number of individual W armor blocks. The flat-tile design comprises a bulky Cu alloy heat sink with multiple W armor tiles joined to its plasma-facing side. It is easier to manufacture and uses the W armor material more economically. However, the mismatch in the coefficient of thermal expansion between the armor tiles and the heat sink results in very high thermal stresses, leading to delamination of the armor from the heat sink. This novel concept uses the design freedom of additive manufacturing (AM) processes to create W-Cu metal matrix composites (WAM-Cu MMCs). The design can be tailored to minimize thermal stresses in the PFC while also creating a mechanical interlock between armor and heat sink, resulting in a highly damage-resilient mock-up with excellent heat removal capabilities. The WAM-Cu MMCs use the well-tested ALMT grade 3N-W as armor material, necessitating hybrid AM processes. In hybrid AM, parts are built on functional components rather than plain build plates, and a subsequent removal is not required. This contribution presents an optimized process for hybrid additive manufacturing on a conventional powder-bed fusion machine, involving the addition of a series of precision slots to a standard build plate to hold the functional material, in this case, specially prepared armor tiles. These W tiles are cut from high-purity plate material with their rolling direction aligned with the build direction during the AM process. The resulting armor tiles in the finished parts are precisely aligned with minimal tolerances. Preforms produced in this manner are infiltrated with Cu in a vacuum-assisted melt infiltration process to create small-scale PFC mock-ups, which are then subjected to HHF testing in the GLADIS test facility. HHF tests indicate a very promising thermomechanical performance of the WAM-Cu MMC PFCs.

Assessment of the influence of heating energy expansion on components in the plasma vessel at W7-X

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The aim of the Wendelstein 7-X (W7-X) experimental facility is to demonstrate the suitability of the stellarator for use in power plants. To this end, experiments with long pulse lengths and high heating powers will be carried out in the future. This means that, with a heating energy of 10 MW, the pulse lengths will be gradually increased from 100 seconds at heating energy level 1 (HE1) to 1800 seconds at HE4. During these long experiments, the plasma-exposed surfaces of diagnostics will be subjected to severe thermal stress from plasma radiation.

This work describes the evaluation method that enables the assessment of the effects of this load on the mechanical integrity and influence on the experimental operation of components installed in the plasma vessel. The aim is to create an overview of the various components, which clearly and unambiguously shows the compatibility of the individual components in terms of the heating power and pulse length of the planned heating energy levels at W7-X.

The evaluation process consists of three steps. The first step is an analysis of the temperature fields in the vessel installation to be examined. Depending on the location and radiation exposure, this is carried out in a simplified qualitative or quantitative manner using the finite element method (FEM) in Ansys. The second step is necessary if the first step reveals that temperatures exceed material-specific limit temperatures or that high temperature gradients occur. A more detailed analysis is then carried out. The effects of the increased temperatures and the resulting deformations and stresses are first examined qualitatively and, if necessary, also quantitatively using FEM analysis. These calculation results are then used to make a statement about the suitability of the vessel installation with regard to its suitability for the corresponding heating energy level. In the third step, the assessments of suitability for the various vessel inserts are compiled in a list and colour-coded according to their degree of suitability. This provides a simple and clear overview of the compatibility of the inserts in the plasma vessel for the corresponding heating energy level at W7-X.

Development of a rapid and simple thermal calculation procedure for diagnostic and wall components in the W7-X plasma vessel for long-pulse operation

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The Wendelstein 7-X stellarator is on track to reach an energy level of 18 GJ per pulse, with a power of 10 MW and a pulse duration of 1,800 seconds. A systematic overview of the multiple diagnostics installed recently has necessitated a procedure for the rapid and conservative numerical analysis of the diagnostics under the expected increased thermal loads.

The calculation procedure consists of two steps. In a first step, radiation analysis is carried out. The last closed flux surface (LCFS) of the plasma is modelled as a radiating surface, with its emisivity representing the maximum expected heat load on the vessel walls. A simple surface model is sufficient for efficiently computing the plasma radiative load taking into account the view factors (visibility factors) between the plasma surface and the irradiated diagnostics. Special care is taken to ensure consistent boundary conditions, emissivities, and numerically stable surface modeling.

In a second step, the computed heat load distribution is transferred as load to a 3D volumetric model for thermal analysis (steady-state or transient). A rapid thermal calculation is carried out incorporating material properties, convective cooling and thermal radiation to the environment. The calculated temperatures were compared with permissible material-dependent specification values. Furthermore, the thermal gradient was checked and, in the case of stronger gradients, a structural mechanical analysis was proposed.

This methodology supports the safe and reliable design/validation of diagnostics, which is particularly relevant for high-power long-pulse operation in Wendelstein 7-X.

Upgrade of KAERI particle flux irradiation facility using in-vessel solenoid coils

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

A divertor particle flux irradiation facility was developed at KAERI using Applied-Field MPD thruster [1, 2]. Our facility can provide deuterium particle flux up to 1.7E23 /m2/s and a total fluence up to 1E27 /m2. Using our facility, we tested SiC first mirror for the ITER VUV spectrometer [3], measured deuterium retention on the ITER grade tungsten, and studied the sputtering yield of 3-D printed tungsten.
In order to further increase the deuterium particle flux, we fabricated an in-vessel solenoid coils that can focus the deuterium ion beam by providing a stronger magnetic field near the target region. The solenoid coils were constructed by winding a rectangular coil with a cross-sectional area of 10 mm2 360 times, capable of producing a magnetic field of 0.1 T when a 60 A current is applied to the coils. The coils will be installed in the KAERI facility in the near future and it is anticipated that increasing the axial magnetic field near the target region to 0.1 T will at least double the deuterium particle flux. We will also introduce automated operation system to increase the operation time and the total deuterium fluence in the future. Furthermore, the sputtering yield, fuel retention, and blister formation of K-doped tungsten will be studied using the KAERI facility.

This work was supported by National R&D Program through the National Research Council of Science & Technology (NST) and National Research Foundation of Korea (NRF) granted by Ministry of Science and ICT (No. GTL25031-410 and RS-2022-00155950).

[1] K.-B. Chai and D.-H. Kwon, Plasma Phys. Control. Fusion 62, 035007 (2020).
[2] K.-B. Chai, D.-H. Kwon, M. Lee, Plasma Phys. Control. Fusion 63, 125020 (2021).
[3] B. Kim et al., Fusion Eng. Des. 200, 114210 (2024).

Procurement Status of ITER Blanket Shield Block in China

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Shield blocks (SBs) are critical components of the ITER Blanket system, providing thermal and neutron protection to the vacuum vessel and external components. A total of 440 shield blocks make up the system, with China responsible for procuring 220 of them across nine rows.

Between June 2015 and March 2018, the China Domestic Agency (China DA) completed process qualification through the manufacture and testing of a full-scale prototype. Series production of SBs began in mid-2018. By December 2023, all 222 pieces of 316L(N) stainless steel forgings had been manufactured and accepted. As of late February 2026, approximately 80% of the procurement has been completed: 169 blocks have been finished and accepted by the ITER Organization, while the remaining blocks are at various stages of manufacturing and testing.

Throughout this procurement effort, China DA established and refined the manufacturing route for SBs and carried out several related R&D activities. These include the development of high-sensitivity hot helium leak testing facilities (capable of detecting leaks as small as 10*10-12 Pa*m3/s at 250 ℃), investigation of key factors influencing leakage rates, and the development of a low-friction ($f < 0.1$) MoS2 coating applied to metal surfaces with roughness below 0.2μm.

The procurement process for ITER shield blocks offers valuable insights and technical references for the design, manufacturing routes, testing criteria, and project management of shielding components in future fusion reactor engineering.

Fusion relevant First Wall manufacturing by Electrical Discharge Machining and Cold Forming: a detailed status review and outlook on further development strategy

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Two different strategies were addressed at KIT for nuclear fusion EUROFER First Wall (FW) manufacturing: a) assembly of 2 FW shells with machined channels by Hot Isostatic Pressing (HIP) and b) machining of channels into a solid plane plate by Electrical Discharge Machining (EDM) + Cold Forming (CF). The strategy EDM + CF provides one key advantage: no welding (except for the connection to surrounding Blanket structures) is needed in the integral manufacturing routine, thus the material properties can be considered comparable to initial conditions.

The results of the relevant KIT research base on nationally funded approaches, a F4E procurement task, EUROFUSION FP8 and FP9. It can be concluded that the process configuration based on EDM and CF has reached a TRL of 4 and therefore provides the presently most advanced solution. The process itself has already been demonstrated and qualified successfully according to RCC-MRx for HCPB TBM Cooling Plate in 2015, however in significantly smaller dimensional scale.

For the EDM process, industrial machining equipment has been realized and tested for cutting channels in BB relevant cross section and length of 2500 mm (a Mock-Up with 8 channels was completed in 2025). Parameter sets are continuously improved and non-destructive examination procedures are developed, qualified and updated. Cost optimization analysis resulted in a configuration of EDM with several channels processed in parallel. Presently the industrial realization of such an advanced processing configuration is prepared.

To demonstrate CF, a bending tool has been realized on basis of TBM relevant specifications. Forming of solid plates (semi-finished shells for HIP based manufacturing) and demonstrators with internal channels (4 channels, 15x15 mm^2, 2x90° and 14 channels, 10x15 mm^2, 1x90°) were conducted successfully. The tools geometry is presently further improved in detail to increase precision in the bending zone. Concerning FW designs in 3D freeform surface configurations feasibility options were demonstrated in experiments in scaled down dimensions and numerical studies. This approach is presently continued aiming on DEMO and alternative (Stellarator) Blanket concepts. Material qualification of samples extracted from the area of maximum deformation is addressed, as well as the completing of a Prototypical Mock-Up for testing in relevant thermo-hydraulic conditions.

This paper describes the present development status and the future strategy to scale up the process steps of EDM and CF to TBM and BB First Wall dimensions and to push the TRL up to a level of 6 by 2030.

A Computational Faceting Method for Optimised Discretisation of ITER First Wall Surfaces

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The geometry of the plasma-facing surface plays a critical role in the thermal and mechanical performance of ITER First Wall (FW) components. In engineering practice, the ideal plasma-facing contour cannot be manufactured as a continuous surface and must be discretised into a finite number of tiles, introducing surface faceting effects. This discretisation can lead to local angular deviations with respect to the plasma surface, potentially affecting heat flux distribution, shadowing behaviour, and component lifetime. In this work, an engineering-oriented faceting optimisation methodology is presented to support the design of ITER First Wall surfaces. A dedicated computational solver has been developed as part of a faceting design workflow to quantify the geometric deviation between a theoretical plasma surface and its faceted First Wall representation. The solver generates optimised faceting configurations by systematically adjusting tile orientation and segmentation while preserving the geometric reference defined by the plasma-facing contour. Key geometric indicators, including local angular deviation and surface conformity metrics at tile centres, are introduced to assess the quality of a given faceting solution. The approach allows consistent comparison between different tiling strategies and supports design decisions by identifying configurations that minimise plasma-facing deviation while remaining compatible with engineering and manufacturing constraints. The proposed methodology provides a practical tool for First Wall engineering, enabling improved alignment between plasma geometry and discretised component surfaces. This approach is intended to support ITER-relevant First Wall design activities and may be extended to other plasma-facing components requiring optimised surface discretisation.

#13157

Boiling Heat Transfer Experiment of a Spontaneous Swirling Flow downstream of a Dual-Elbow Piping for Fusion Divertor Cooling Development

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Fusion divertors are expected to be exposed to extremely high heat fluxes of 10 MW/m2or higher, and therefore require a highly efficient cooling system. For the divertor cooling system, the system using multi-elbow piping has been proposed and its heat transfer characteristics have been investigated over the years. The multi-elbow piping generates a complicated swirling flow downstream and due to the spontaneous swirling flow, heat transfer performance is enhanced compared to a straight pipe turbulence flow. In a previous study[1], boiling heat transfer experiment was conducted using a test section with 20 mm×18 mm in cross-section and 108 mm in length in which a cooling channel of ∅10 mm was fabricated. In the experiment, since a heating area was set on the upstream side of the test section with 10 mm×60 mm, temperature measured in the downstream part of the heated part became low due to heat leak to the downstream side of the test section. In this study, the test section was modified in its size with the same cross-section and length of 61 mm, and boiling heat transfer experiment was conducted using the spontaneous swirling flow downstream of a dual-elbow piping under the condition of Reynolds number of 10000 and 20000, a degree of subcooling of 100 K, and one-sided heating condition with uniform heat flux by an electron gun, ACT2, in NIFS. From the experimental results, the heat flux reaching DNB was predicted based on the measured temperatures despite the critical heat flux could not be obtained in the experiment. The results revealed that the spontaneous swirling flow sustains nucleate boiling up to the higher heat flux than the developed turbulence. Based on these results, the cooling performance of the multi-elbow piping was compared to the swirl tube dedicated for ITER. The pump power [W/m] of this spontaneous swirling flow cooling system required for cooling of ITER steady state operation was lower than that of the swirl tube under the same flow velocity condition. Under ITER pump power condition, the flow velocity would be predicted to be 1.5 times higher. Additionally, it was found that the maximum heat flux of ITER transient operation, 20MW/m2, could be cooled by transition boiling at least 6D length in the flow direction. These findings suggest that the spontaneous swirling flow may be feasible option for fusion divertor cooling.

[1] S. Nakano, et al., Fusion Engineering and Design, 204 (2024) 114408.

Manufacturing and Quality Assessment of a Copper based Plasma-Grid for GLADIS

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The high heat flux (HHF) test facility GLADIS (Garching Large Divertor Sample Test Facility) is used to perform high heat flux (HHF) tests of material samples as well as water-cooled plasma facing components. It is equipped with two ion sources, each of which can generate a 55 keV hydrogen beam at a total power of 1 MW. The most sensitive component of the ion source is the actively water-cooled three-grid extraction system made of oxygen free, high conductivity (OFHC) copper. The existing grids have been manufactured in in the 1980s for the former neutral beam injection system of the tokamak ASDEX. The applied Cu/ Cu and Cu /steel brazing technology was initially designed for a limited number of plasma discharges and a length up to 10 s.
After 20 years of operation in GLADIS with 300.000 pulses and a total operating time of 800 hours, there have been first indications of a loss of beam performance due to wear of the so-called plasma grid. This far exceeds its predicted lifetime and led to the decision to develop a replacement.
The new design is based on state-of-the-art Cu compound manufacturing, in which the most highly loaded grid area was manufactured completely electrochemically. The arrangement of the 187 extraction holes, Ø 7.85 +0.05, each individually water-cooled, requires very fine, vacuum-tight Cu cooling channels with a minimum wall thickness of 0.6 mm at a water pressure of 10 bar. This requires a controlled fine-grained, hard Cu material that is not commercially available in the specified size and quality.
Finally, after all qualification tests and installation, the new grid achieves approximately 15-20% higher performance at comparable operating parameter.
The contribution presents and discusses the design, manufacturing and qualification route in detail. We analyse the wear of the grid, which has been in use for 20 years, due to thermal load and copper erosion. Results of microscopic investigation, 3D laser geometry measurements and a comparison of the beam performance will be presented.

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Keywords: plasma facing component, high heat flux, GLADIS, galvanic process

Additive manufacturing and high heat flux testing of divertor mock-ups with novel cooling structures

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

At Karlsruhe Institute of Technology (KIT) divertor design concepts for future nuclear fusion power plants and, in parallel, additive manufacturing methods of such components are intensively investigated. Complex geometries and fine critical features enabled by the powder bed fusion technologies are highly appealing for fusion divertor component manufacturing. The powder bed fusion technologies electron beam melting (EBM) and selective laser melting (SLM) are promising additive manufacturing methods, as they allow a high degree of freedom regarding complex part structures compared to conventional manufacturing processes. In particular, the manufacturing of fine cooling and grid structures can be ideally achieved by using 3D printing. The ongoing development of the technology in terms of the processing of copper materials offers ideal conditions for the upcoming scientific investigations.
This contribution describes the fabrication route of divertor mock-ups with novel cooling structures, via electron beam melting (tungsten armour) and selective laser melting (copper cooling structure), including computational fluid dynamic (CFD) calculations. Furthermore, the mock-ups were characterized by high heat flux (HHF) tests in the neutral beam high heat flux test facility GLADIS (Garching Large Divertor Sample test facility) at Max Planck Institute for Plasma Physics (IPP) in Garching. The HHF investigations provide information about the thermo-mechanical behaviour to extreme steady state thermal loading. Post mortem analyses are performed quantifying and qualifying the occurring damage with respect to reference tungsten grades by metallographic and microscopical means.

CFD Simulations of Boiling in Stagnant Regions of the DEMO Divertor Cassette Body

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Volumetric neutron heating of the DEMO divertor cassette body (CB) and coolant may locally reduce the saturation margin in the water-cooled channels, potentially leading to boiling. While earlier single-phase CFD analyses indicated such regions, initial two-phase simulations exhibited numerical instabilities that limited confidence in the boiling-risk assessment. In the present work, the two-phase modelling framework has been improved through enhanced mesh quality, refined solid-fluid interface resolution, and simplified wall-boiling assumptions, enabling stable Euler-Euler simulations with RPI heat-flux partitioning and an extended Critical Heat Flux (CHF) treatment. The approach was validated against a reference subcooled boiling experiment.

Applied to a reduced CB domain beneath the inner vertical target, the simulations consistently predict vapour formation in stagnant or low-velocity channel regions, particularly in upper curved sections. High near-wall void fractions coincide with local wall-temperature hot spots and reduced heat transfer due to vapour insulation, indicating a transition toward CHF-like behaviour. These interface hot spots are only weakly reflected at the outer CB surface. A revised correlation-based assessment suggests that CHF cannot be excluded in stagnant regions, motivating design optimisation and further dedicated two-phase investigations.

#13230

INVESTIGATING THE INFLUENCE OF POTASSIUM-DOPING ON THE PLASMA MATERIAL INTERACTION RESPONSE OF TUNGSTEN

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten (W) is the leading candidate for plasma-facing components in fusion divertors, yet the operational envelope of conventional ITER-grade W is constrained by low-temperature brittleness and microstructural degradation under high heat and particle flux. A promising alternative to conventional alloying is potassium (K) doping, which retains a near-pure W matrix while enabling microstructure engineering: K is introduced through the dispersion of very small K bubbles at high density resulting in microstructural refinement. Varying the K-doping changes the density and stability of these features, thereby altering grain-boundary mobility, resistance to recrystallisation and grain growth, and the near-surface defect landscape that governs hydrogen isotope trapping. However, how these doping-controlled microstructural differences translate into plasma material interaction (PMI) behaviour, particularly surface damage evolution and deuterium (D) retention, remains insufficiently quantified.

Here we benchmark ITER-grade W against a set of K-doped W variants spanning different doping/strength conditions under high-flux plasma exposure in the PSI-2 linear plasma device. We isolate the role of K-doping-driven microstructure on surface morphological evolution and damage mechanisms during plasma loading, and we assess the associated D uptake/retention response relative to the undoped baseline. By directly correlating K-doping condition (and resulting strength state) with PMI-induced surface modification and D retention trends, this work maps the viable design space of K-doped tungsten as an advanced, “near-pure W” divertor material concept for ITER-relevant operation.

Heat Treatment Effects on Subcooled Boiling and Thermal Conductivity of ITER-Grade CuCrZr

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The copper alloy CuCrZr is being considered for use in a cooling tube of the divertor of the ITER fusion reactor, which has a high heat flux of 10 MW/m2 . ITER-grade Cu alloy metal materials are processed using solution and aging treatments, and their electric conductivity is increased by precipitation strengthening. The temperature dependence of the properties of materials with such a special structure implies a boiling state in the duct line of a water cooling divertor or in a loss-of-flow accident (LOFA) involving boiling heat transfer. When designing a water cooling divertor, it is necessary to know the boiling characteristics in such a state and the boiling transmission characteristics of the materials in the duct line. The boiling properties of the CuCrZr alloy assumed to be used in ITER materials have been reported and compared with those of anoxic Cu. We investigate the heat properties of matter, such as the thermal conductivities of ITER-grade Cu alloy materials, and spur the development of Cu alloy materials for the new nuclear fusion divertor by elucidating the boiling phenomenon. Specifically, the thermal conductivity of Cu alloy materials greatly depends on their processing method used and the heat-treatment during the manufacturing process and is known to take various distributions. In the present study, we investigated the thermal conductivity of ITER-grade Cu alloy materials to explain their increase in electric conductivity at various temperatures as a result of precipitation strengthening during solution and aging treatments. We conducted boiling experiments to investigate the influence of solution and aging treatments on the peculiar boiling properties of a Cu alloy. As a result, we elucidated the heat transfer properties of ITER-grade Cu alloy metal materials with a few experiment examples, along with the boiling properties of the materials. We expect these results to contribute to the development of Cu alloy materials for the ITER nuclear fusion divertor. Our results indicate that the CuCrZr alloy exhibits adequate heat transfer characteristics even when subjected to quenching because its heat transfer properties closely resemble those of normal Cu.

Effects of Compressive Residual Stress on Helium-Induced Sputtering in Tungsten Surface: a Molecular Dynamics Study

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Tungsten, with its high melting point, excellent thermal stability, and superior erosion resistance, is a leading candidate for plasma-facing materials (PFMs) in fusion reactors. In deuterium–tritium (D–T) fusion, high-energy neutrons and helium ions irradiate tungsten surfaces. When helium ions irradiate tungsten, they break atomic bonds, causing sputtering of tungsten atoms. If these atoms enter the plasma, their high atomic number (high-Z) increases radiative losses, reducing plasma temperature and stability, and potentially triggering plasma disruptions and vertical displacement events (VDEs). Therefore, minimizing atomic sputtering is critical, though direct atomic-scale observation is challenging. One approach to mitigate tungsten atom sputtering is to introduce compressive residual stress on the tungsten surface, and laser shock peening (LSP) is considered as an effective method for imparting compressive residual stress with minimal surface roughening.

Molecular dynamics (MD) simulations are powerful tools for studying such phenomena, as they track atomic motions based on Newton's equations. In this study, MD simulations were employed to investigate tungsten PFMs under fusion-relevant conditions. Press-loading simulations of polycrystalline tungsten were employed to emulate LSP-induced compressive residual stress, successfully reproducing the expected compressive residual stress distribution. Shear bands formed within surface grains, and 1/2 screw dislocations were observed along slip planes, indicating that screw dislocations generate high Peierls stress and influence the maximum compressive residual stress magnitude. Subsequent helium irradiation simulations on surface-modified tungsten showed a reduced sputtering yield compared to pristine tungsten specimens, demonstrating the protective effect of compressive residual stress. The effects of helium incidence angle, irradiation energy, and residual stress magnitude on sputtering were systematically analyzed.

Furthermore, high-energy neutron irradiation poses significant challenges to the structural integrity of tungsten. Incident neutrons generate primary knock-on atoms (PKAs), which initiate collision cascades within the tungsten lattice. These cascades produce substantial atomic displacements and lattice disorder, thereby degrading material reliability and ultimately limiting the operational lifetime of fusion reactors. A comprehensive understanding of collision cascade evolution—particularly thermal-spike-mediated Frenkel pair recombination—is essential for accurate displacement-per-atom (DPA) assessments. Accordingly, future work will simulate neutron-induced damage through PKA-driven collision cascades, with emphasis on directly observing Frenkel pair recombination and systematically quantifying Frenkel defect formation as a function of PKA energy. These atomistic-level insights will advance the fundamental understanding of tungsten behavior under fusion-relevant plasma, thereby providing a scientific basis for the design and optimization of more resilient plasma-facing components.

Synergistic Optimization of Tungsten-Potassium Materials for Enhanced Thermal Stability and High-Heat-Flux Testing

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Nuclear fusion is widely recognized as the ultimate clean energy source for humankind. As a key component exposed to the extreme service environment in tokamak devices, the divertor has become a critical bottleneck restricting the development of fusion reactors, where advanced materials and reliable fabrication technologies are urgently demanded. In this work, the synergistic effects of potassium (K) concentration and rolling passes on the thermal stability of potassium-doped tungsten (KW) alloys were systematically investigated. The optimized condition—100 mg/L K doping coupled with three rolling passes—delivered superior thermal stability. Dispersed potassium bubbles efficiently suppressed grain coarsening, while the tailored strain distribution preserved high dislocation density and alleviated stress concentration, collectively enhancing the high-temperature structural stability of the material. Using the optimized KW alloy as the plasma-facing material (PFM), flat-type divertor target mockups were manufactured by integrating KW plates with oxygen-free copper (Cu), oxide dispersion-strengthened copper (ODS-Cu), and reduced-activation ferritic-martensitic steel (RAFM). The mockups were subjected to high heat flux testing up to 20 MW/m^2 for 1000 cycles. The maximum surface temperature reached 1327.1 °C, the overall structure and interfacial integrity remained well-preserved. No severe KW surface melting or KW/Cu interfacial debonding was detected.

Temperature-Dependent Transmutation-Induced Clustering in Neutron-Irradiated Tungsten and Microstructural Stabilisation via ZrC Dispersion

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten (W) is the baseline plasma-facing material for next-generation fusion devices due to its high melting point, thermal conductivity, and sputtering resistance. However, under fusion-relevant neutron irradiation, transmutation reactions generate rhenium (Re), osmium (Os), and tantalum (Ta), leading to solute clustering that alters microstructural stability and degrades high-temperature mechanical performance. A mechanistic understanding of clustering evolution under relevant temperature and dose conditions is therefore essential for defining the safe operational window of W-based plasma-facing components.

In this work, atom probe tomography (APT) is utilised to quantify the temperature- and dose-dependent evolution of irradiation-induced clustering in neutron-irradiated W. Tungsten specimens were irradiated in the BR2 reactor to 0.2–0.8 dpa over 600–1200 °C. Neutron spectra and corresponding damage and transmutation levels were determined using FISPACT inventory calculations. Across all irradiation conditions, APT mass spectra consistently resolve transmutation products (Re, Os, Ta), enabling quantitative cluster analysis.

Three-dimensional reconstructions reveal diffuse Os-core/Re-shell–type solute clusters without sharp compositional interfaces, indicating that cluster nucleation and further cluster growth precedes second-phase formation. A clear temperature dependence is observed: cluster number density decreases systematically with increasing irradiation temperature, while cluster morphology evolves from compact, near-spherical configurations at 600 °C to strongly elongated, rod-like features at 1200 °C. Radius-of-gyration tensor analysis demonstrates that pronounced anisotropy (elongation ratio >3) dominates at the highest temperature and dose, evidencing a temperature-controlled transition in cluster growth mode. These observations suggest enhanced solute mobility and directional coarsening mechanisms at elevated temperature, with implications for irradiation hardening and creep resistance.

To establish the link between irradiation microstructure and service-relevant behaviour, neutron-irradiated W–ZrC alloys (0.4–0.5 dpa at 400, 800, and 1200 °C) were mechanically tested at 400 °C and subsequently analysed by APT. The results indicate that ZrC act as effective defect and solute sinks, reducing cluster number density, limiting coalescence, and suppressing elongation and progression toward second-phase formation. This dispersion-strengthening approach mitigates transmutation-induced microstructural instability by preventing early cluster-to-new phase transition pathways.

Overall, the results provide quantitative insight into temperature-dependent clustering mechanisms in neutron-irradiated tungsten and demonstrate that ZrC alloying offers a viable microstructural stabilisation strategy. Such controlled defect–solute management is critical for extending the high-temperature performance envelope of tungsten-based plasma-facing materials in fusion neutron environments.

#13282

3D Numerical Modelling of Particle Exhaust in the EU-DEMO LAR Divertor

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The divertor plays a crucial role in the operation and overall performance of fusion reactors by managing plasma-generated heat and particle fluxes while ensuring reactor stability and component longevity. This study investigates the divertor concept for EU-DEMO (DEMOnstration Power Plant), with particular emphasis on the recently developed Low Aspect Ratio (LAR) divertor. The LAR configuration has evolved through successive design stages aimed at improving reactor performance, sustainability, and operational efficiency. A series of three-dimensional numerical simulations was performed to analyse particle exhaust dynamics, geometric effects, and neutral gas behaviour. The simulations were conducted using the DIVGAS framework [1–4], developed by the Vacuum Group at the Karlsruhe Institute of Technology (KIT). The DIVGAS framework offers two modules – deterministic (based on Discrete Velocity Method) and stochastic (based on Direct Simulation Monte Carlo method). Both accurately capture neutral gas behaviour across the full range of neutral gas collisionality. The results highlight the significant influence of divertor geometry, leakage phenomena, and related physical processes on particle transport and exhaust performance. Under conservative boundary conditions with respect to the backgound pressure, the leakage flux accounts for approximately 20–30% of the total deuterium flux, while the entry gaps toward the plasma vessel contribute more than 60% of the incoming flux to the sub-divertor region. However, additional preliminary simulations suggest that, under modified boundary conditions closer to a more realistic scenario, both the pumped flux (maximum 3%) and the flux through the entry gaps increase (maximum 13%). These findings offer valuable insights into optimizing the EU-DEMO divertor configuration, enhancing particle exhaust management and supporting the sustainability of future fusion power plants. They also inform the next LAR divertor design iteration, with implications for more efficient pumping system integration and improved performance in future reactors.

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#13284

The ITER Tungsten First Wall

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The ITER ‘Start of Research Operations (SRO)’ will include the installation of a passively cooled First Wall. This allows a learning phase to early ITER operations that is more forgiving to disruptions, resulting in reduced risk of a water leak in-vessel. This “Temporary First Wall” (TFW) will mimic the actively cooled First Wall (FW) in material, geometry, and interfaces, with a shape and structure that is intentional alike to its successor in DT-1 and DT-2. The TFW design includes a mixture of plasma-facing tile design types, including bulk Tungsten, Tungsten heavy alloy, and potentially Tungsten-coated or Tungsten-clad steel. The latter option is a novel technology for fusion application, so a robust coating qualification plan is therefore underway: first by cyclic transient loading and high heat flux testing (via electron beam), and second by ITER-relevant loading of large areas in tokamak(s). A design challenge for the TFW will be its tolerance to the anticipated high expected transient loads of the SRO phase, especially in the current quench phase during disruptions and runaway electron events. Modelling of these loads will assist in the design development of the TFW to enable successful completion of the ITER research plan for SRO.

The First Wall armor material change from Beryllium to Tungsten eliminates beryllium-specific toxicity, lowers erosion rates, and improves melting temperature. However, the advanced maturity of the Blanket System requires careful efforts to integrate the new armor into the final design. The change of material is also accompanied by a mitigation to runaway electron damage, for which a thicker armor layer is proposed in the impacted areas. The material change and modified thickness requires design adaptation of each panel type to re-align all panels to a new position. As many of the shield blocks (on which the FW panels mount) are manufactured, the re-alignment of panels is done by FW structural thickness modifications. The change to tungsten has design benefits as well, with higher relative melting temperature, and a theoretically more robust bond to the copper alloy heat sink. These together permit simplifications to the design and tolerancing of the panels. While several full-scale prototypes of the beryllium First Wall panel have been completed, work is now underway for re-production of similar tungsten-armored mock-ups to demonstrate the performance of newly developed technology. This qualification work is proceeding in parallel with on-going factory manufacturing of stainless steel and copper alloy parts of the First Wall.

#13285

Closed Island Divertor Concept and Directional Neutral Flow–Driven Pumping Strategy in the GIGA Stellarator Power Plant

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

This contribution establishes the conceptual design of a closed, baffled island divertor for GIGA, the stellarator-based fusion power plant developed by Gauss Fusion. The design directly resolves both heat and particle exhaust challenges, explicitly addressing particle control as a critical reactor-scale constraint for stellarators.

The divertor configuration is guided by the coupled assessment of magnetic connection length and heat flux distribution. The inherently long connection lengths characteristic of optimised stellarators mitigate peak power loads, as confirmed by preliminary heat-flux scans over reactor-relevant parameter ranges. These analyses indicate that established divertor technologies may be sufficient, provided appropriate geometric optimisation is implemented. Thermo-hydraulic evaluations further confirm acceptable component temperatures under realistic loading scenarios.

The GIGA divertor adopts a straight, cassette-based architecture, simplifying integration while preserving tritium breeding blanket coverage and enabling early incorporation of remote maintenance requirements.

Particle exhaust, however, remains the dominant design driver. Enhanced divertor closure is achieved via strategically positioned baffles that maximise neutral confinement without intercepting magnetic islands. A zero-dimensional pumping assessment, informed by Moscheni et al. Nucl. Fusion 66 (2026) 026008, indicates reactor-feasible pump numbers, contingent upon achieving sufficiently high divertor neutral pressures.

To address this critical dependency, a novel pumping philosophy is introduced. The concept exploits experimentally and numerically observed directional neutral flows, converting intrinsic neutral momentum into enhanced static pressure at the pump interface. Preliminary simulations reproduce the underlying mechanisms and indicate improved divertor pressure and pumping performance, supporting the viability of reactor- control in GIGA.

Thermal surrogate modelling for quasi real-time monitoring of actively cooled first wall components in ITER

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

During ITER operation, first-wall components such as the Diagnostic First Walls (DFWs) will be exposed to intense radiative and nuclear heat loads, making active cooling mandatory for sustaining long discharges. During the design phase, high-fidelity numerical models that explicitly resolve the cooling network are used to define the component designs and specifications, ensuring that thermo-mechanical limits are not exceeded. However, such models are computationally expensive and inherently not suitable for thermal state monitoring during operation.\\
In this work, a mesoscale Porous-Media Approach (PMA) is applied to reduce the complexity of the problem while capturing the thermal response of the actively cooled DFW. The DFW is modelled as a single equivalent homogeneous medium, in which the effect of the unresolved cooling network is represented through an additional closure term in the volume-averaged enthalpy transport equation. These closure terms are computed from the intrinsic thermophysical properties of the coolant and the spatial characteristics of the cooling layout. The surrogate model is embedded in an end-to-end framework under development within the ITER Tokamak System Monitor (TSM). This framework automates the meshing of 2D planes of the first wall components using SALOME, sets the prescribed heat flux boundary conditions, and initializes the porous thermal resistance profile from the component's cooling circuit design and coolant properties. The PMA formulation is implemented in the FEM-meshfree MKNIX solver, which computes the slice wise temperature evolution. \\
Applied to the Equatorial Port Plug 12 bottom DFW, the proposed method reproduces plasma-facing temperature as well as internal temperature profiles obtained with high-fidelity models. Efficiency, accuracy and compliancy to the requirements will be presented in this paper. The solver frequency of MKNIX-PMA falls within the $\mathcal{O}$(1–10 Hz) range, making it compatible with quasi real-time temperature monitoring.

Numerical Investigation of Thermal Impact and Plasma-Facing Component Damage Induced by Runaway Electrons

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

In magnetic confinement fusion devices, particularly future reactors, runaway electrons (REs) generated during major disruptions pose a severe threat to plasma-facing components (PFCs). Under unmitigated conditions, RE beams with kinetic energies of tens of MeV and currents up to several tens of MA can deposit MJ/m^2-level energy onto the first wall within milliseconds. Such extreme transient loads can cause melting, evaporation, and coolant leakage. Understanding RE-induced thermal damage is therefore critical for designing resilient plasma-facing structures.

This work presents a numerical investigation of RE–first wall interactions using Geant4–finite element workflow. Geant4 Monte Carlo simulations are employed to calculate spatial energy deposition in a W/Cu monoblock plasma-facing unit (PFU), while the subsequent thermal response and phase transitions are analyzed using finite element methods.

A parametric study under DEMO-relevant conditions examined RE impacting energy densities of 20–1000 MJ/m^2, impact durations of 0.2–20 ms, and incident angles of 2°–90°. Results show that REs can penetrate a 2 mm tungsten armor layer. At high incident angles and high RE energies, the deposition profile exhibits a Bragg peak–like feature, producing localized subsurface heating that may trigger internal melting, vaporization, and droplet ejection. At sufficiently high beam energies, burn-through of the PFU cooling pipe and coolant leakage may occur.

The proposed damage mechanism explains observations from the EAST tokamak. In shot #99323, severe melting was observed at the leading edge of a divertor dome PFU tile. Simulations indicate that tile misalignment combined with high-angle RE incidence produces edge-localized Bragg peak–like energy deposition, leading to subsurface tungsten evaporation and droplet ejection. The predicted damage dimensions agree with experimental observations.

In shot #135886, the experiment was terminated by a disruption, coolant leaked from the antenna limiter subsequently, accompanied by strong hard X-ray and neutron signals which indicates significant RE generation. Analysis suggests that if ~10% of magnetic energy is converted into RE energy and deposited on the PFU, the resulting heat flux can exceed the critical heat flux in the cooling pipe, causing pipe melting and leaking. At incident angles around 30°, the CuCrZr pipe may melt while the tungsten armor remains soild, consistent with experimental evidence.

Future work will incorporate multiphysics coupling with OpenFOAM to capture melt motion, boiling, and thermoelectric J×B effects, enabling more realistic prediction of RE-induced PFC damage.

Structural analysis of DTT divertor PFU assembly under electromagnetic loads

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The primary objective of the divertor tokamak test (DTT) facility is to study the feasibility of innovative approaches to address the power exhaust challenges crucial for future fusion power plants. Several divertor concepts will be tested at DTT under conditions relevant to DEMO. The first divertor under development follows ITER-like target design and technologies with the plasma facing units (PFUs) composed of W monoblocks joined on CuCrZr pipes and stainless-steel XM-19 supports via Cu interlayer and brazing, respectively. The divertor PFUs will accommodate strike points at various positions according to the different equilibria of the promising plasma scenarios. Three PFU target regions are identified as the inner, the central (or dome) and the outer targets, the latter consisting of an outer-vertical target and an outer-horizontal target. The cooling pipes of these regions are joined in series and each region is fixed via the supports to a stainless-steel 316-LM back-plate that acts as the interface with the cassette body. In total, the DTT divertor PFU assembly is composed of 9 individual PFUs and three back-plates.

Previous studies on the structural integrity of PFUs during normal and off-normal operating conditions have shown that electromagnetic (EM) loads are the source of higher (local) stresses in the CuCrZr pipe and supports, and have pointed out the PFU regions where additional supports can improve the preliminary design. As the design of the DTT divertor PFU moves forward, additional analyses of the entire assembly with updated component geometries under the expected EM loads remain necessary to verify its suitability. A finite element (FE) model of the latest design of the divertor PFU assembly has been developed. Realistic interactions, featuring kinematic couplings and contacts with analytical surfaces, mimic the behaviour of the hinge and hinge/slot supports between the PFUs and plates, as well as the joining bolts at the plate/cassette interface. The EM loads during a plasma slow unmitigated downward vertical displacement event (DVDE) with a 40 ms current quench time are considered in the structural analysis with a strategy for using a discrete vector field of forces in the ABAQUS environment. The results of the analysis provide the most vulnerable regions sustaining the highest stresses, and a preliminary verification following the codified rules in the Structural Design Criteria for In-vessel Components (SDC-IC).

Experimental Investigation of Flow Boiling in a Divertor Cooling Channel Mock-up Under Fusion-Relevant Heat Fluxes

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Flow boiling is one of the most efficient heat transfer mechanisms available, which makes it suitable for cooling the divertor targets of future fusion reactors. In these components, extremely high heat loads must be removed in a reliable and controlled way. However, flow boiling at such high heat fluxes involves complex and strongly coupled physical processes that are still not fully understood. High quality experimental data focusing on phenomena near the heated surface are needed to improve and validate the wall-boiling modelling in computational fluid dynamics.

To address this, we developed the Fusion Experiment for Divertor Optimization Research Applications (FEDORA) facility. With the goal to reproduce divertor-relevant boiling conditions in a laboratory environment where visual access for detailed boiling study is possible, fusion operating conditions were scaled down by matching the relevant dimensionless numbers and by using the refrigerant R245fa as a surrogate working fluid. This allowed us to design a transparent cooling channel mock-up with a square cross-section and optical access from multiple sides. High-speed cameras operating in both the visual and infrared spectral ranges were used simultaneously. Heating was applied from one side using a 5 μm thick electrically heated stainless-steel foil attached to the inner wall of the channel, reaching heat fluxes of up to 2 MW/m^2.

In this study, we present the methodology and results for determining bubble size distribution. Simultaneously with optical measurements, we also recorded temperature profiles on the heated foil in contact with the liquid. Particular attention was given to bubble nucleation site density and bubble departure frequency, and their variation with flow velocity and applied heat flux. Our experimental findings are further supported by comparison with measurements performed at other high-heat-flux facilities operating under similar conditions but using water as the working fluid. The data obtained in this campaign provide detailed insight into interfacial dynamics and wall heat transfer under high heat load conditions. These results will facilitate the development and validation of advanced wall-boiling models for simulations of flow boiling in divertor cooling channels.

Additive manufacturing of tailored W-CuCr1Zr interfaces by means of PBF-LB/M multi-material processing for plasma-facing component applications

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

In the divertor region of magnetic confinement fusion devices, plasma-facing components (PFCs) are required to withstand high heat flux (HHF) loads, while corresponding steady-state design loads are considered to reach values of up to 20 MW/m^2. Under such loading conditions, PFC materials have to face microstructural changes, neutron irradiation induced property degradation, as well as significant thermomechanical stresses. These stresses tend to localise at interfaces where materials with differing thermomechanical properties are joined.

While the so-called flat-tile PFC design – based on a bonded interface between plasma-facing tungsten (W) armour tiles joined to a bulky copper (Cu) alloy heat sink – represents a reasonable and scalable design approach, it introduces challenges related to thermomechanical stress concentration along the bonded W-Cu alloy interface, especially due to the mismatch in the coefficient of thermal expansion (CTE) between W and Cu alloy.

Against this background, this contribution will present a novel approach for the realisation of structured W-CuCr1Zr interfaces based on multi-material (MM) additive manufacturing (AM). The goal with this approach is to realise a mechanically robust and potentially optimized joint capable of promoting a more gradual transition in effective material properties while also enhancing mechanical interlocking at the armour-heat sink interface within a single MM fabrication process.

In this regard, the study focuses on the parametric fabrication of tailored W-CuCr1Zr interfaces by means of laser-based powder bed fusion of metals (PBF-LB/M). Corresponding bi-material specimens with differing surface topology were fabricated through a dedicated MM processing in which W and then CuCr1Zr are processed sequentially within the same AM build chamber. Samples were fabricated with varying AM parameters and are investigated by means of microscopy in order to identify suitable processing parameters for corresponding W-CuCr1Zr joint fabrication. Promising parameter combinations are used for the fabrication of specimens with relevant size for divertor PFC application. Eventually, plans regarding the fabrication of small-scale PFC mock-ups realised with the proposed MM AM process and to be tested under relevant HHF loading conditions will be outlined.

Bubble formation on tungsten-based alloys after exposure to fusion-relevant synergistic heat and plasma loads

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Plasma-facing materials (PFMs) are exposed to severe conditions in fusion reactors, in particular in the divertor. PFMs in the divertor are exposed to high heat and particle loads, which can cause cracking, erosion and melting, among other detrimental effects. It has been previously shown, that under certain conditions, such as a base temperature of about 700-800 °C and simultaneous exposure to deuterium and helium plasma together with ELM-like thermal shocks, micrometer-sized bubbles can form under the surface of the material, which could negatively influence heat dissipation, fuel retention and the mechanical properties of the material [1].
At the same time, different tungsten-based alloys have been produced by powder injection molding (PIM), and tested under fusion-relevant conditions. PIM tungsten-based alloys showed better resistance to thermal shock damage from ELM-like loads when compared to ITER-grade tungsten [2].
In the current study, PIM tungsten-based alloys (W-1TiC, W-2Y2O3, W-3Re-1TiC, W-3Re-2Y2O3, W-1HfC and W-1La2O3-1TiC), as well ITER-grade tungsten have been exposed to similar conditions as in [1]. They were exposed to simultaneous D/He plasma and ELM-like thermal shocks at T > 700 °C base temperature, in order to test the influence of the alloying elements on the formation of helium bubbles. After exposure, samples were analyzed with SEM and FIB, where the PIM tungsten alloys showed a significant reduction in the formation of helium bubbles.

[1] J. Nucl. Eng. 2023, 4(1), 204-212; https://doi.org/10.3390/jne4010016
[2] J. Nucl. Eng. 2025, 6(3), 25; https://doi.org/10.3390/jne6030025

#13353

Off-normal transient effects on the structural integrity of the ITER inertial tungsten first wall

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

In the 2024 ITER re-baseline, the first wall armour is exchanged from beryllium (Be) to tungsten (W) and the revised ITER Research Plan (IRP) calls for an inertially cooled, temporary first wall (TFW) for the Start of Research Operation (SRO) phase [1]. During SRO, unplanned disruptions are expected to occur more frequently as experience with machine operation and mitigation is gained. Consequently, it is necessary to assess the impact of the fast transient heat loads which unmitigated disruptions will deposit on the TFW and its supporting structure.

Whilst these off-normal thermal loads are not the design basis for the TFW, they define the operational limits of the system, since the brittle W exhibits surface crack formation under very high thermal gradients induced by fast transients. In the worst case, these seed cracks may rapidly propagate across the tile thickness, compromising structural integrity. This will call for corrective maintenance, seriously impacting the tokamak operational availability.

This paper investigates the thermo-mechanical behaviour of inertially cooled bulk W tiles subjected to disruption-induced heat loads foreseen during plasma ramp-up and under normal operating conditions (e.g. hot Vertical Displacement Events, [2]). The 3D plasma heat flux distribution in [2], calculated accounting for detailed wall geometry and magnetic topology, is mapped onto the TFW geometry. Transient thermal and structural finite element analyses are performed on the most loaded tiles within the ANSYS Workbench environment to resolve the spatio-temporal temperature gradient and related stress field, which is used to evaluate the likelihood of fast fracture. The risk of fast fracture damage mode is assessed by crack propagation studies evaluating the instability of existing material defects under the applied loads, which causes its mechanical failure. The findings will ultimately assist in driving operating instructions to guide operations during SRO for machine protection purposes.

[1] R.A. Pitts et al., NME 42 (2025) 101854
[2] F.J. Artola et al., 30th IAEA Fusion Energy Conference (IAEA FEC 2025)

Steps towards in situ surface regeneration of plasma facing material

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten is currently the most promising candidate for the plasma-facing material (PFM) of a first of a kind fusion power plant (FPP). It features unique properties like the highest melting point of every element, a high thermal conductivity, low tritium inventory and comparatively low erosion rate under plasma loading. However, even a highly resilient PFM such as tungsten can only withstand the harsh conditions experienced by the armor of plasma-facing components (PFCs) over a limited timeframe. Currently, this challenge is being addressed by (1.) developing PFMs and designing PFCs for maximum durability, (2.) operating the reactor in plasma scenarios that minimize wall damage (only possible to limited degree), and (3.) replacing entire PFC units at the end of their lifespan. Such replacements require years due to the enormous remote maintenance effort involved. In order to increase the availability and thus the economic competitiveness of FPPs compared to alternative energy conversion technologies, it is therefore necessary to explore alternative strategies that address this durability problem. Erosion losses of PFCs could be compensated by in situ repairs using additive manufacturing (AM) technology in the form of local direct energy deposition (DED), while surface damage could be healed. The wire-based laser metal deposition (LMD-w) process fulfills several necessary conditions for repair operations inside the fusion reactor vacuum vessel. Since processing tungsten in suitable quality poses a significant challenge due to its unique properties, a particularly customized LMD-w processing unit is developed. The processing unit will be equipped with sensor technology to determine remotely which armor areas require how much material deposition and to subsequently control the LMD-w process results. By utilization of two laser spots, one for melting and one for pre- and post-heating, homogenization of the temperature profile and flattening of the cooling curves during and after the LMD-w process are targeted. Besides experiments under local inert gas atmosphere, the behavior of the process under vacuum is investigated. These measures aim to increase the quality of the LMD-w tungsten coatings. In the context of this work, progress regarding the in situ regeneration concept and development of an LMD w processing unit specifically designed for tungsten will be presented.

Quantifying the effect of impurity-seeded plasmas and high heat fluxes on potassium-doped tungsten

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten (W) is one of the candidate materials for the plasma-facing regions of future fusion reactors, due to its high resistance to erosion, low hydrogen retention and high thermal conductivity. However, it is inherently brittle due to its high ductile-brittle transition temperature (DBTT) and is also expected to undergo recrystallization induced embrittlement over the operation of the fusion reactor. The mechanical properties can be improved with the use of additives, such as potassium (K) as a dopant, which lowers the DBTT and increases its resistance to radiation-enhanced recrystallisation. In this work, the behaviour of K-doped W under fusion relevant conditions is studied, specifically the effects of impurity-seeded plasma exposure and high heat fluxes on surface morphology, erosion rates and thermal fatigue tolerance. This is benchmarked against the behaviour of pure W under similar conditions. The effect of grain size and the influence of impurities present in the alloy (such as Al and Si) are also assessed.

To evaluate the plasma-material interaction effects, samples are exposed to ~50-200 eV mixed plasmas with incidence fluxes between 10^{20}-10^{21} ions·m^{-2}·s^{-1} and fluences as high as ~10^{26} ions·m^{-2} over the temperature range of 600-800 °C, which is relevant for the plasma-facing regions. Impurity-seeded plasmas such as mixtures of deuterium with helium, nitrogen as well as noble gases like argon and xenon are used. The response of the material to different impurity species, varying fluxes and their influence on surface morphology, retention and erosion rates are examined. High heat flux tests are performed to understand the thermal fatigue tolerance of the material under steady state heat loads, with heat fluxes up to 40 MW/m^2. The effect of ELM-like transient heat loads is also studied through exposures in the order of GW/m^2. The post-exposure damage to the samples, such as surface roughening, cracking, melting, evaporation and droplet ejection, is evaluated at different heat loads. This provides useful data on the safe operating conditions for the use of K-doped W in a fusion reactor and optimising the design of plasma-facing components.

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#13386

Analysis, Design and Testing of the JT-60 SA Tungsten Divertor

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

JT-60SA's primary mission is to support ITER operations in preparation for the DEMO design. During the initial research phase, JT-60SA is equipped with an inertially cooled carbon/graphite divertor for short pulses. In the subsequent Integrated Research Phase, when all heating systems are available, the inertially cooled divertor (ICD) will be replaced by an actively cooled divertor (ACD).
The ACD consists of 36 units, each approximately 1.8m wide and 1.3m tall. Each unit includes a Stainless-Steel (SS) Cassette Frame that supports High Heat Flux (HHF) and Normal Heat Flux (NHF) elements. Initially, the plasma-facing components were designed with graphite tiles on top of heat sinks made of TZM for HHF components and CuCrZr/SS plates for NHF components. The heat sinks are cooled by pressurized water at an inlet pressure of 1.4 MPa (NHF) and 2 MPa (HHF).
In 2024, the decision was made to use tungsten instead of graphite, this change allowed for a review of the heat load profile on the divertor Plasma Facing Units. Thermal, electromagnetic, and mechanical analyses were conducted to ensure that the components comply with ITER SDC-IC structural design code rules. Additionally, high heat flux tests up to 15MW/m2 were initiated to test the joining technique and prove the feasibility of the manufacturing procedure
In this paper, the main characteristics of the W divertor are described, together with the main changes with respect to the previous graphite design. The results of the finite element analyses and of the experimental tests are also highlighted.

High heat flux performance of CuCrZr cooling structures produced by laser powder bed fusion

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

For a future fusion reactor, the heat exhaust is one of the most critical components. Stationary heat loads of at least 20 MW/m^2 can be anticipated in the divertor region. Under the tungsten armor, a heat sink made from CuCrZr is foreseen due to the high thermal conductivity and mechanical plasticity.
Besides the choice of material, also the inner shape and surface play an important role in the heat removal capabilities of a heat sink by altering the heat transfer coefficient. In the past, several designs were developed ranging from simple pipes, included swirl tapes, to different variations of hypervapotrons. Usage of additive manufacturing (AM) allows a higher degree of freedom and shapes that are more complex, without increasing the machining time and cost.

This study explores the feasibility of additive manufacturing (AM) via laser powder bed fusion to produce integrated cooling structures from CuCrZr alloy. Four different designs of the internal structure and their influence on the cooling capabilities and flow resistance were investigated. Three different kinds of lamellas (hypervapotron type, fins and double fins) are investigated in addition to one empty channel as reference. Employing laser powder bed fusion, dense CuCrZr cooling structures with wall thicknesses between 0.5 and 2 mm, were manufactured successfully. A common heat treatment protocol of 500°C, 1h was employed for part of the samples to improve the thermal physical properties of AM-CuCrZr.

These structures were then tested in JUDITH 2 using varying water pressures (10 bar – 40 bar) and flow rates (50 l/min – 120 l/min). Thereby, clear differences of the flow resistance, i.e. pressure drop, caused by the different designs could be determined. A minimal wall thickness was identified before significant plastic deformation occurred. Samples which passed the pressure and leakage test, were loaded with high heat fluxes (2 MW/m^2 - 10 MW/m^2) and the obtained surface temperatures were compared with FEM simulations. This comparison also allows disentangling the effect of increased cooling area, different thermal conductivities with and without heat treatment and of changed flow dynamics. The improvement of the heat transfer coefficient and the impact on the hydrodynamic properties of the cooling channel geometry are shown. The results illustrate the potential of AM-CuCrZr for heat sink applications and give useful guidelines for further design development.

#13430

Advancing Liquid Metal Plasma?Facing Component Technologies in the FIRE Collaborative Program

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Flowing liquid metals (LM) are a leading pathway for plasma-facing components (PFCs) capable of meeting the extreme heat-flux, particle-flux, and transient-event requirements of next-step fusion facilities. While solid materials such as tungsten remain central to present designs, their power-handling limits, erosion behavior, and plasma-material interaction challenges motivate the development of LM systems that can provide high heat-flux tolerance, self-healing surfaces, and improved confinement performance. Low-Z LM such as lithium have demonstrated strong potential for enhanced confinement and reduced recycling, while higher-temperature, higher-Z LM such as tin offer improved vapor-pressure characteristics and operational robustness. In addition, a new class of engineered LM alloys is being developed to tailor thermophysical and magnetohydrodynamics (MHD) properties for specific divertor regimes.

The FIRE Collaborative program brings together national laboratories, universities, and private-sector partners to accelerate LM PFC technology maturation by addressing critical scientific and engineering gaps identified by prospective fusion end-users. The program is structured around four major technology development thrusts. The first focuses on integrated LM PFC testing, including new flowing-LM test stands, divertor-relevant prototypes, and coordinated diagnostics for surface evolution, vapor shielding, and transient response. The second thrust advances material property and compatibility data, with updated measurements of corrosion, wetting, evaporation, and plasma-exposure behavior for lithium, tin, and novel LM alloys. The third thrust targets LM flows in magnetic fields, supported by new MHD modeling, electrically insulated flow channel concepts, and experimental validation of j×B-driven and pressure-driven LM flows. The fourth thrust focuses on novel LM alloy development, where engineered compositions are being formulated to optimize vapor pressure, wetting behavior, MHD response, and compatibility with structural materials across both low- and high-recycling divertor regimes.

A major emphasis of current FIRE activities is the development and evaluation of LM divertor technologies that can provide robust power-exhaust solutions for next-step fusion facilities. These include vapor box divertors; divertors with capillary porous system with fast flow called CPSF; lithium/metal infused trenches (LiMIT) which uses thermoelectric MHD drive to flow lithium; divertorlets, compact LM divertor heat transfer enhancing modules; and magnetic centrifuge concepts, which separate liquid lithium hydrides from lithium, using magnetic field from the fusion device to create a hydrocyclone effect. The FIRE program is now positioned to deliver integrated LM PFC demonstrations that address key TRL gaps and support the design of next-step fusion test facilities and pilot plants. The presentation will summarize these developments and outline the path toward deployment-ready LM PFC technologies.

Study on the Microstructure of Rolled W-0.5 wt.% TiC Alloy with High Hardness and Thermal Stability

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The improvement of tungsten materials as PFMs is an important research direction. In this paper, hot rolled W-0.5 wt.% TiC alloy are fabricated via a combination of sintering at 2300℃ and hot rolling with 50% thickness reduction. The as-rolled alloys show high hardness of HV 620 at 25 ℃. The hardness hardly decrease with increased annealing temperature up to 1400 and 1500 °C for 1 h. In addition, the EBSD results show the initial recrystallization temperature of as-rolled alloys is higher than 1500℃。The TEM result show that the dispersed phase in the alloys is not ceramic phases such as TiC, (TiW)C, etc., but a metal solid solution of W,Ti and small amount of C. And a small amount of Ti atom is enriched in the W phase of the alloy. The good thermal stability and unusually high hardness are attributed to the special microstructure structure.

#13450

Summary of hydrogen and helium retention results leading to low recycling operation in HIDRA with liquid lithium PFC's.

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Since 2016, the former WEGA stellarator has been operating as the Hybrid Illinois Device for Research and Applications (HIDRA) at the University of Illinois Urbana-Champaign. Typical toroidal magnetic fields are $B0 \leq 0.5$ T and plasma pulses can be up to $tplasma \leq 10{,}000$s. The focus has been in doing plasma material interactions (PMI), in particular, researching applications of liquid metals (LMs) as plasma facing components (PFCs). The materials used for the divertor and first wall will be critical for successful operation of future fusion devices. Currently solid materials struggle to achieve the desired results, especially when instabilities are present. However, liquid metals, and in particular liquid lithium, offer a possible solution. LMs are resistant to damage from the plasma and neutrons through self-healing effects through liquid metal flows. As part of the UD liquid Metal PFC development program, experiments involving liquid lithium evaporations into helium and hydrogen plasmas showed that lithium will retain these species and provide an extremely low recycling operating regime. Plasma performance increased substantially, showing that the low recycling regime indeed will help plasma performance. Typically, these HIDRA experimental plasmas are anywhere between $tplasma = 600$ -1000s, which allows for a full range of operating parameters, with and without lithium interaction, so high and low recycling regimes can be studied within a single run. This paper will present a summary of many of the retention experiments that have been performed, showing the low recycling operation, potential vapor shielding of lithium on a PFC surface and the mechanism for the retention, especially with inert gases.

Numerical simulation-driven development of the COMPASS Upgrade plasma-facing components

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The COMPASS Upgrade (COMPASS-U) is the next version of the COMPASS
tokamak at the Institute of Plasma Physics in Prague, Czech Republic. It is designed to fill in gaps in physics knowledge, especially in the area of divertor physics. For example, it will be used to study power exhaust or test liquid metal divertors for future devices. The COMPASSU has a high toroidal magnetic field (up to 5 Tesla), a high plasma current (up to 2 MA) and the vacuum vessel and metallic components that come into contact with the plasma are heated up to 500°C. It is a medium size (R = 0.9 m, a = 0.27 m) and is used at high temperatures. It is also expected to be exposed to high heat fluxes and electromagnetic (EM) loads. This contribution presents a summary of the design validation of the COMPASS-U plasma-facing components by finite element analysis. The COMPASS-U has a closed divertor based on a cassette concept and inner guard limiters (IGL), both hosting bulk tungsten tiles. Inconel 718 was chosen as a base material for the supporting structures because of its strength at high temperatures. Tungsten tiles do not have active cooling because the plasma discharges last less than three seconds. This means that the tiles are exposed to high heat fluxes, up to 70 MW/m2 in the divertor and 10-30 MW/m2
on the IGL. A comprehensive numerical modelling study was conducted on divertor and IGL components. Heat loads are determined using a 3D field line tracing code called PFC-Flux and added to a finite element model (FEM). This is done along with peaking factors for safety margin. EM loads are based on 3 most severe disruption scenarios at 5T and 2 MA. The bolts connecting the baffles and the bolts holding the tiles in place proved to be critical parts of the
design. In order to assess cracking in the brittle phase of tungsten tiles, normal stress was calculated during the temporal evolution of temperature in combination with an EM force impulse. This was then compared to the temperature-dependent yield strength of the material. The results showed that the normal stress was always below the limit for yield strength and
was mostly in the ductile phase. The model helped with the design process by highlighting weaknesses, and the final design was eventually validated.

Non-destructive and mechanical characterization of W coatings produced by means of APS for the Outboard First Wall of DTT

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The Divertor Tokamak Test facility (DTT) aims at investigating power exhaust solutions that can be relevant for DEMO and future fusion reactors. Such a relatively compact tokamak is provided with a significant amount of heating power, and it is conceived to be compliant with different plasma scenarios. This poses constraints to the engineering design of the actively cooled plasma-facing components (PFCs) of DTT. For instance, the Outboard First Wall (OFW) must withstand thermal and electromagnetic loads arising during both normal and off-normal operations. Each OFW module consists of water-cooled plates made of AISI 316L. Additive manufacturing is envisaged to realize such plates already integrating cooling circuits with square cross section. Instead, their plasma facing side is W-coated, and the thickness required to cope with the net erosion envisaged in DTT falls in the range 30-300 m. For this reason, Chemical Vapour Deposition (CVD) and Atmospheric Plasma Spray (APS) with local inert gas protection are being investigated as industrially available techniques to deposit thin W coatings on AISI 316L. Following the realization of coated samples and mock-ups, qualification activities are foreseen by means of: non-destructive testing (NDT), mechanical characterization and cyclic thermal fatigue up to 5000 cycles at 0.5MW/m2.

The present work describes the non-destructive and mechanical characterization of W-coatings deposited by means of APS on AISI 316L. A first optimization of the deposition process was carried out in collaboration with the industrial partner, aimed at minimizing the porosity and oxide content of the deposited layers while maximizing the bonding strength with the substrate. In a second step, Ultrasonic Testing (UT) and Eddy Current (EC) were investigated as NDT methods to assess the structural integrity of samples and mock-ups, with particular focus at the interface with the AISI 316L substrate, after the deposition process and after cyclic thermal fatigue. A high-frequency UT system was developed for this application: it is based on the pulse-echo water gap technique, and relies on high frequency transducers to assess the size and depth of discontinuities in the volume of coatings provided with a few hundred microns thickness. For thinner coatings, other techniques were pursued, above all the superficial EC method. Also, metallographic examinations of selected samples provided a final validation of the NDT results. Finally, the adhesive/cohesive properties were assessed on a dedicated tensile test equipment, customized for shear, bending and tensile tests.

Thermostructural verification of the DTT Inboard First Wall Limiter

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The Divertor Tokamak Test facility (DTT) [1] is a superconducting tokamak currently under construction at ENEA C.R. Frascati. During operation, various undesired transient events—such as major disruptions, fast and slow vertical displacement events, and fast discharges—may occur, generating significant electromagnetic (EM) loads in the plasma-facing components (PFCs) due to currents induced by variations in the magnetic field. Among these components, the Inboard First Wall (IFW) limiter of DTT is expected to be particularly affected.

To protect the vessel inboard and in-vessel components during plasma transients, the IFW limiter design consists of CuCrZr coaxial tubes armoured with tungsten monoblocks. However, the low electrical resistivity of the cooling pipe leads to high electromagnetic loads.

This work presents the results of a stress analysis performed on the IFW limiter modules, considering the relevant thermal and electromagnetic loads that may arise under the operating conditions. Finite-element simulations were conducted to determine the stress distribution within the limiter components. The SDC-IC acceptance criteria established for ITER were used to verify the structural integrity of the limiter module layout and to evaluate possible improvements of the design.

[1] Francesco Romanelli et al 2024 Nucl. Fusion 64 112015

Investigation of the oxidation process in boron coatings

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Due to the ITER re-baseline, including the change of the first wall material from Be to W, oxygen and other impurities are no longer sufficiently gettered in the vessel. A thin amorphous boron layer, which is applied via boronization during the regular wall conditioning phase, can reintroduce impurity gettering and a more efficient plasma operation can be obtained.
Oxygen impurities in fusion reactors restrict the transition to high-performance plasma discharges in divertor configurations. Boron coatings address this by reducing the oxygen content. Even though the boronization process is well established and used for several decades, the detailed understanding of the oxidation of boron coatings is missing. Since graphite tiles were used primarily in past fusion devices and are still used in some cases and C and B are forming stable molecules which probably change the functionality of B coatings, the prediction of the effect of boronization in full W fusion devices as ITER is challenging.
In order to describe the oxidation process of boron in detail, B layers were fabricated by magnetron sputter deposition on W and Si substrates.
After characterization, B layers on Si substrate are exposed to well-defined artificial atmosphere with specific composition, humidity (70 % r.H.) and temperature (400 °C) in a thermobalance. To investigate the mechanism of oxygen gettering by B layers, SEM images, FTIR and XPS spectra are taken before and after exposition to dry and humid synthetic air. Changes in surface morphology and chemical bondings will be reviewed for different temperature and humidity levels. Furthermore, the influence of D and W in B layers on the oxidation process will be studied by fabricating mixed B layers.

Utilisation and Topology Optimisation of Tungsten Fibre-Reinforced Materials and Interfaces in Plasma-Facing Components

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The divertor of a magnetic confinement fusion reactor faces some of the most challenging conditions in the entire device and must be designed to withstand thermal cycling, plasma erosion and neutron bombardment. A current leading component design combines tungsten (W) armour tiles and a copper (Cu)-based heatsink due to the excellent erosion resistance, low hydrogen retention and high thermal stability of W, and the high thermal conductivity of Cu. This concept also has a lower associated cost than previously used ‘monoblock’ constructions by limiting W use to only where it is most needed. The main challenge associated with this design is the difference in coefficient of thermal expansion (CTE) of the two materials, which can lead to delamination and component failure under thermal cycling. Additionally, in their monolithic forms, these two materials have no overlap in their ideal operating temperatures.

This work explores the use of W fibres as reinforcement in the form of a three-dimensional weave. This design can provide both mechanical reinforcement of the joint from fibre bridging, and reduce the CTE mismatch at the material interface. W fibres are particularly attractive for this application due to their high strength and evidence of ductility in both the as-manufactured state and after preliminary irradiation trials. These two properties make it possible to weave the fibres into textile preforms that provide reinforcement across a wide temperature range. The flexibility offered by the textile industry means that W loading can be carefully designed to optimise a balance of thermomechanical properties, strength and cost. In order to ensure that the preforms are woven in the ideal layout for the application, topology optimisation has been used, the results of which are discussed in this contribution. For this work, a representative volume element (RVE) approach was used to simplify the fibre-matrix interactions and focus on optimal fibre reinforcement layout. Finite element analysis (FEA) was then used to simulate thermomechanical loading in the component, followed by optimisation for stress minimisation with a focus on the material interfaces. These results will be used to inform the final design of fibre-reinforced component mock-ups, manufactured using a combination of melt and chemical vapour infiltration.

Crack investigation at the divertor tungsten tile using FEM

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Since 2010, tungsten has been widely recognized as the leading candidate for plasma-facing materials in next-generation plasma fusion devices. In general, solid tungsten is used for shielding the plasma chamber interior against the high heat generated from the extremely hot plasma. The most loaded structural parts inside the plasma chamber are the so-called divertor plates composed of individual components.

The Axial Symmetric Divertor Experiment Upgrade (AUG), a plasma fusion experimental device has been gradually equipped with solid tungsten divertor tiles beginning with 2012. The full-scale prototypes, dimension of 250x80x15 mm3, have been intensively tested in the high heat flux test facility Garching LArge DIvertor Sample test facility (GLADIS) [1]. The applied GLADIS heat loading profiles are Gaussian with central heat flux of 10 - 30 MW/m2, resulting in an integrated absorbed power of the W tile between 100 and 280 kW. Thus, simulating the expected highest power and energy loads during adiabatic loading in AUG. The corresponding measured surface temperatures reached values between 1500 °C and 3300 °C. In addition, cyclic loading tests have been performed with 200 cycles at 10.5 MW/m^2, 3.5 s duration. The total number of applied heat load cycles corresponds to the expected thermal loading of about 4 years of AUG operation with approximately 50 high power discharges per campaign.

Despite the successful performance of tungsten tiles in GLADIS tests simulating AUG operating conditions, after one campaign of AUG operation (about 1200 plasma shots) almost all tungsten divertor tiles exhibited cracks.

A comprehensive investigation of the damages has been performed to find out the origin of the crack initiations [2].

This paper is presenting a bundle of numerical simulations of the AUG solid tungsten divertor tiles on the basis of the theoretical hypothesis for failure of brittle materials. Accordingly, thermomechanical analyses with cyclic loading, simulating both the GLADIS and AUG load profile have been performed. Additionally an assessment of the crack initiation induced by material fatigue under thermal cyclic load has been studied. Finally, the design optimisation considerations of divertor tiles are discussed.

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Multi-objective optimization of the next-generation W7-X divertor

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The quasi-isodynamic, neoclassically optimized stellarator Wendelstein 7-X (W7-X) employs an island divertor, whereby a magnetic island chain is used to divert plasma onto target plates. Since 2022, these target plates have been actively water-cooled and use CFC as the plasma-facing material. The next-generation W7-X divertor is being developed to incorporate reactor-relevant, actively water-cooled, metallic target technology [1]. In parallel to this technological development, the plasma-facing geometry of the divertor (targets and baffles) is being optimized within the complex design space inherent to the island divertor concept. One key motivation is particle exhaust: the current W7-X divertor geometry exhibits low net neutral pumping, with the pumped fraction remaining below one percent, as analyzed by EMC3-Eirene and DIVGAS simulations [2,3]. This underscores an outstanding challenge in the progression towards a stellarator power plant.

In this contribution, we formulate a multi-objective design problem based on three coupled performance metrics: (i) the maximum heat flux deposited on the divertor targets, (ii) the diversion efficiency, defined as the fraction of the cross-field particle flux from the plasma edge diverted to the targets, and (iii) the collection efficiency, defined as the fraction of neutralized particles entering the sub-divertor region. The maximum heat flux and diversion efficiency are quantified using the anisotropic diffusion model EMC3-Lite, which has been demonstrated to align well with W7-X experimental values in attached conditions [4]. The collection efficiency is evaluated using Mercury, a newly developed, low-fidelity neutral gas dynamics model designed for rapid geometry scans.

Although these low-fidelity models lack physical processes such as particle drifts and plasma-neutral interaction, they allow us to introduce a divertor design cycle that automatically iterates the parametric design of targets and baffles, the EMC3-Lite evaluation of heat load and diversion metrics, and the analysis of neutral collection with Mercury. Using this workflow, Pareto-optimal divertor concepts are identified and compared with each other and with the current W7-X divertor. The discussion focuses on the trade-offs in performance when transitioning to more closed divertor concepts, which aim to increase collection efficiency and consequently neutral exhaust.

References

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Fabrication and performance evaluation of CuCrZr/SS316L bimetallic transition tubes by explosive welding for divertor

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

CuCrZr/SS316L bimetallic transition joints are critical connection in the divertor from monoblock Plasma Facing Unit(PFU) to cooling pipes of fusion reactors. However, the inherent metallurgical incompatibility between these dissimilar alloys poses significant challenges for conventional fusion welding techniques. This study investigates a novel tube-to-tube explosive welding (EW) configuration designed to produce high-integrity CuCrZr/SS316L transitions. The influence of lap length on the joint's mechanical integrity was systematically evaluated through tensile testing. Multiscale interfacial characterization, utilizing optical microscopy (OM), scanning electron microscopy (SEM) with energy-dispersive X-ray spectroscopy (EDS), and electron backscatter diffraction (EBSD), revealed a characteristic periodic wavy interface featuring significant grain refinement and the absence of deleterious intermetallic compound layers. Experimental results indicate that tensile strength exhibits a positive correlation with lap length. Furthermore, helium leak detection confirmed the hermetic integrity of the joints, validating the technical viability of the proposed explosive welding approach for the robust manufacturing of divertor transition components.

Study on Neutron Streaming Effects through Gaps between Monoblocks and Inter-module Gaps in the Divertor

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The divertor is one of the most critical components of a fusion reactor, as it must withstand extreme thermal loads and particle fluxes. To meet these requirements, most divertors, including those for ITER, employ a water-cooled tungsten monoblock (MB) concept. For post-ITER devices designed for D-T burning, the divertor will also play a vital role in shielding against the 14.1 MeV neutrons generated by fusion reactions. However, in such ITER-like MB-type divertors, preventing neutron streaming through the poloidal gaps between MBs and the toroidal gaps between Plasma Facing Unit (PFU) modules remains a significant challenge. In these gap regions, the vacuum vessel and divertor components are directly exposed to 14.1 MeV neutrons without the protector of the tungsten (W) armor, potentially leading to significant neutron damage and activation.

In this study, neutron streaming effects originating from these gaps were investigated using a simplified divertor model of Compact Pilot Device considering D-T burning, which is currently under conceptual design in Korea. The model consists of W armor, a copper (Cu) interlayer, a CuCrZr heat sink, water coolant, and an Advanced Reduced Activation Alloy (ARAA) cassette body (CB). Unlike ITER, ARAA, a reduced-activation ferritic/martensitic steel, was used for the CB material to mitigate neutron activation. Neutron transport calculations were performed using MCNP6 with the FENDL-3.1 library, and activation analyses were conducted using FISPACT-II with the TENDL-2017 library.

The results indicate that the peak neutron flux in the gap regions of the W, Cu, CuCrZr, and coolant is approximately 6–8% higher than the minimum flux observed directly beneath the MBs. In the ARAA CB top region (at a depth of 1 cm from the surface), the difference increases to approximately 11%. Most of the flux variation is attributed to 14.1 MeV neutrons. In activation assessments over a 100-year cooling period, the difference in activity increases to 11–37%, with the ARAA CB top region exhibiting differences of up to 65%. These results demonstrate that even small variations in high-energy neutron flux can significantly increase activation levels due to threshold reactions.

Impact of the EUROfusion DEMO LAR Baseline on the WCLL Breeding Blanket Thermal-Hydraulic Performance

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The EUROfusion DEMO design has recently undergone a significant revision, resulting in the release of an updated baseline defining the size and main parameters of the future European tokamak reactor. The review process preceding this update identified several critical issues, including excessive divertor heat fluxes during re-attachment events, strong sensitivity of plasma performance to confinement assumptions, fatigue concerns in the central solenoid, and significant challenges related to the Breeding Blanket (BB) remote maintenance.

Among the key design parameters, the aspect ratio (r/R0, i.e., ratio of the major radius to the minor radius of the toroidal plasma) was identified as a strategic lever to optimize overall machine size while reducing technological constraints on the toroidal field coils and divertor protection requirements. From a magnetic standpoint, the updated Low Aspect Ratio (LAR) configuration provides clear benefits, particularly in terms of magnetic confinement performance.

Within this framework, the Water-Cooled Lead Lithium (WCLL) BB remains one of the leading candidates for the EU-DEMO driver blanket. Its performance critically depends on the effective cooling of both the structural components and the LiPb alloy, which acts as neutron multiplier and tritium breeder.

Following the introduction of the new LAR baseline, minor updates were implemented in the BB layout, accompanied by moderate modifications in the estimated thermal loads acting on the system. Updated CFD analyses have been therefore carried out, consistently adopting the same methodology used for the previous baseline to enable a direct comparison between configurations. In addition, targeted sensitivity studies were performed to assess the impact of key modeling assumptions on the predicted thermo-hydraulic behavior. The results provide quantitative insight into the implications of the revised DEMO configuration for WCLL BB cooling performance.

Overview of the Korean Strategic Research of Innovative Fusion Divertor for the Next-Step Fusion Device

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The divertor is one of the most challenging components in a magnetic confinement fusion reactor, as it must withstand extremely high heat loads and high particle fluxes from the scrape-off layer (SOL) plasma. While ITER has adopted the water-cooled tungsten monoblock concept to manage a steady-state heat flux of 10 MW/m^2, demonstration (DEMO) or next-generation innovative fusion devices, such as the compact pilot device (CPD) currently under design in Korea, are expected to face even more severe environments. With a fusion power of 500 MW and a major radius of 3.5 m, the CPD's size is approximately 44% smaller than ITER for the same fusion power level, leading to a significant increase in divertor heat flux. Furthermore, the D-T burning operation necessitates accounting for intense nuclear heating, damage, and radioactive activation from 14.1 MeV neutrons, as well as stringent shielding requirements for the vacuum vessel and superconducting magnets.

To overcome these engineering bottlenecks, Korea launched the "innovative fusion divertor (iFD) R&D Program" in 2025. This comprehensive initiative integrates the entire lifecycle of divertor development, encompassing integrated design, advanced materials, innovative manufacturing, non-destructive inspection, and performance evaluation. The iFD program, scheduled to continue through 2030, operates as a "One-Team" consortium involving 19 institutions from industry, academia, and government research institutes, alongside active international collaboration.

The primary objective of the iFD program is to develop an innovative divertor capable of handling peak heat loads 50% higher than ITER baseline, ensuring structural integrity and preventing tungsten recrystallization under steady-state heat fluxes of at least 15 MW/m^2. To achieve this, the program is exploring high-efficiency cooling geometries, such as square-channel hypervapotrons and multi-swirl structures, which are being validated through high-fidelity CFD analysis and manufacturability assessments. Simultaneously, we are developing potassium (K)-doped tungsten to raise the recrystallization temperature from the conventional 1,250°C to over 1,500°C. Furthermore, to evaluate thermal performance under these extreme conditions, we are constructing a world-class High Heat Flux Test (HHFT) facility. This facility will feature three 800 kW electron beams for large-area testing and dual cooling loops capable of providing low-temperature/low-pressure water for copper alloys and high-temperature/high-pressure Pressurized Water Reactor (PWR) condition for Reduced Activation Ferritic-Martensitic (RAFM) steel structures.

Comprehensive study on the neutron irradiation effects of the latest ITER-grade CuCrZr for Japanese DEMO divertor application

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

CuCrZr has been identified as the optimal material for the ITER divertor heat sink, owing to its favorable balance of thermal conductivity and mechanical properties, and it is also a candidate for the future DEMO divertor. Since CuCrZr is an age precipitation hardened alloy, its properties can vary depending on processing history, so ITER requires strict ITER grade specifications in order to meet the demands of the fusion divertor environment. For DEMO, neutron irradiation becomes a major concern, but earlier irradiation studies from the 1990s during the ITER material screening campaign used materials with inconsistent quality and the properties different from the ITER-grade specification. Furthermore, the strengthening and deformation mechanisms of this alloy are not yet fully understood. Therefore, a comprehensive study has been initiated to characterize the latest ITER-grade CuCrZr, incorporating the effects of neutron irradiation.

The material prepared for the study underwent additional heat treatment relevant to Japanese DEMO divertor fabrication. Initial characterization on non-irradiated material included tensile testing, thermophysical analysis, and microstructural evaluation. The microstructural evaluation revealed that the microstructure is dominated by both coarse μm-sized Cr rich and Zr rich precipitates and densely formed fine nm-sized Cr rich precipitates. Iso thermal annealing and hardness tests demonstrated that the fine precipitates are primarily responsible for strengthening, since they lose stability above 500°C, which correlates with hardness reduction, while coarse precipitates remain stable up to 600°C.

Irradiation effects were evaluated using HFIR neutron irradiated samples. The material irradiated up to 5dpa exhibits irradiation hardening of about 60% at 100°C, decreasing to about 5% at 300°C. Unlike former study that showing irradiation softening above 250 °C, the result showing irradiation hardening even at 300°C, demonstrating a more gradual softening trend. Additionally, ductility loss at 100°C was smaller than previously reported, which had been one of a concern, providing a new insight for using CuCrZr at lower temperatures.

Detailed investigation on the microstructure changes by means of TEM, SEM, and in-situ EBSD tensile testing are also on-going conjunction with DIC analysis of the tensile tested specimens to assess the effect of irradiation on deformation mechanism. From these results, the effects of neutron irradiation on ITER-grade CuCrZr will be comprehensively discussed.

Irradiation-induced hardening and defect characterization in ARAA ferritic/martensitic steel

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The advanced reduced-activation alloy (ARAA) was first developed in South Korea as a candidate structural material for fusion reactors due to its superior resistance to thermal expansion, void swelling, and He-induced swelling. Its elevated-temperature mechanical properties are comparable to those of commercial reduced activation ferritic/martensitic (RAFM) steels, such as Eurofer 97 and F82H. However, the irradiation behavior of ARAA has not yet been investigated.

In this study, ion irradiation was employed to emulate neutron irradiation damage because of its advantages, including low radioactivity, short experimental time, and reduced cost. A hot-rolled, normalized, and tempered ARAA alloy was irradiated with Fe^{+} ions up to 20 dpa at 300 °C. The hardness of the irradiated samples was measured by nanoindentation as a function of depth. The depth distributions of implanted Fe ions and dpa were calculated using the Monte Carlo SRIM (Stopping and Range of Ions in Matter) 2013 code. The correlation between hardness and irradiation damage along the depth was analyzed. Transmission electron microscopy was used to characterize irradiation-induced defects and to identify the boundary between irradiated and unirradiated regions, confirming the SRIM-predicted damage profile.

Modelling D/H pumping speed and outgassing after plasma discharges in WEST

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

During the WEST campaigns in deuterium and hydrogen the outgassing from the
vacuum vessel has been analyzed by means of mass spectrometers and pressure gauges located at the midplane and at lower divertor pumping systems. The analysis has been performed over a period of more then 1000 s after plasma discharges.
After WEST discharges the data show a typical temporal behaviour characterized
by an initial rapid gas pressure rise at the disappearance of the plasma followed by a (exponential) short term decay due to volume pumping and a final long term decrease due to outgassing from the walls.
Experimental and theoretical studies [1] suggest a power law decay in the outgassing rate with respect to time, and according to these findings we have derived an outgassing equation and used it to predict the characteristic parameters of the WEST pump-down system: volume pumping speed, initial outgassing rate and outgassing power law decay index.
Preliminary calculations confirm a pumping speed in agreement with the values estimated from calibration gas injections (without plasma) and the index decay (less then unity) from previous works [2].
The procedure can be used to analyze a large number of discharges in order to determine the evolution of the parameters with respect to the evolving wall conditions throughout an experimental campaign.
The results obtained are scaled per unit area for comparison with similar data from other tokamaks in operation or that have operated, and can then be extrapolated to larger tokamaks under construction.
[1] V.Phillips and J.Ehrenberg, J.Vac.Sci.Technol.A 11 (1993) 437.
[2] T.Loarer et. al., Nucl. Fusion 60 (2020) 126046.

3D Design Optimization of KSTAR Tungsten Divertor Using HEAT

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The management of heat loads on plasma-facing components (PFCs) remains one of the fundamental challenges for the successful operation of fusion reactors. As future fusion reactors will operate under significantly higher power densities than current experimental devices, ensuring the thermal integrity of the divertor is essential to prevent irreversible material degradation and premature component failure. Conventional 2D analyses insufficiently account for complex 3D geometries, assembly misalignments, and magnetic shadowing, which can lead to severe localized failures such as leading-edge melting. Therefore, 3D analysis using the Heat flux Engineering Analysis Toolkit (HEAT) is required to accurately evaluate the effects of 3D PFC shaping—such as macroscopic tilting and microscopic chamfering—intended to avoid leading edges and to mitigate locally concentrated heat fluxes.

The objective of this study is to perform 3D shape optimization of the KSTAR tungsten divertor using HEAT and to evaluate its thermal stability through CFD analysis. Surface heat fluxes are calculated using HEAT, and temperature distributions and thermal stability are estimated by using ANSYS CFX. The simulations utilize magnetic equilibrium reconstructions from KSTAR discharge #38798 as primary inputs. While the current KSTAR divertor monoblocks were designed with symmetrically rounded corners to allow for reversed toroidal field operations, this study focuses on optimizing the 3D PFC design specifically for uni-directional operation. By parametrically varying geometric features—specifically the chamfering depths of the monoblocks and the tilting angles of the targets—variations in 3D surface heat flux distributions and their resulting temperature distributions of the KSTAR divertor are investigated. The analysis focuses on the impact of these geometric configurations on the overall heat flux distribution, the maximum surface temperature, and the potential for leading-edge exposure under the given magnetic equilibrium.

The 3D design optimization provides the design window for optimal tilting and chamfering configurations among the parametrically varied models derived from the baseline geometry. These configurations effectively mitigate leading-edge exposure while maintaining the maximum heat flux and surface temperature within the limits of thermal integrity. This work shows the feasibility of using HEAT coupled with ANSYS CFX to optimize 3D PFC shape, providing a practical reference for the design of divertors in future fusion reactors.

#13747

LBM-based Simulation of Hydrogen Isotope Permeation and Retention in Fusion Reactor Structural Materials

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

During fusion reactor operation, hydrogen isotopes (HIs) escaping from the plasma implant into and accumulate within plasma-facing components (PFCs). This rapid tritium retention not only leads to material embrittlement but also poses severe radiological hazards upon accidental release.

This study establishes an HI transport model based on the lattice Boltzmann method (LBM) and develops a corresponding simulation code to investigate tritium transport processes within multi-material structures. Compared with conventional continuum-based schemes, LBM is well suited to this problem because of its simple mesoscopic formulation, high parallel efficiency, and natural treatment of coupled transport across heterogeneous interfaces. The proposed LBM framework integrates comprehensive physical models, including heat conduction, particle implantation, defect trapping, and surface release. Employing D1Q3 and D2Q5 lattice arrangements, the model effectively resolves the temperature and concentration continuity issues at heterogeneous material interfaces by implementing conjugate heat transfer and chemical potential boundary conditions.

To evaluate the reliability of the developed LBM code, one-dimensional simulations were performed using a reference object analogous to an ITER first wall monoblock(composed of tungsten, copper, and a copper alloy) and compared against finite element method (FEM) results. The comparison demonstrates the high accuracy of the LBM approach; the maximum relative error for the temperature field is 1.2%, and the concentration profiles of both mobile and trapped tritium exhibit excellent agreement. Subsequently, the transport behavior was analyzed under simulated fusion reactor operating conditions, comprising 100 plasma cycles with varying HI implantation fluxes. The results indicate that tritium can migrate against the concentration gradient at the tungsten-copper interface due to disparities in the saturation concentrations of the respective materials. These preliminary findings demonstrate the code's predictive capability under typical fusion scenarios, providing a robust analytical tool for optimizing the anti-permeation performance of PFCs and supporting comprehensive reactor safety assessments.

Electromagnetic Model for the lower divertor of ASDEX Upgrade: how to improve the operational performance

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

ASDEX Upgrade, a medium-sized tokamak operated by the Max Planck Institute for Plasma Physics, plays a key role in the development of plasma scenarios and technologies relevant to next-generation fusion reactors, including ITER, which has also adopted tungsten as a plasma-facing material. ASDEX Upgrade has been a pioneering device in the adoption of tungsten, being among the first tokamaks to implement a full tungsten first-wall and divertor configuration. This transition, motivated by the need for materials with high melting points and low tritium retention, has provided valuable operational experience for future fusion devices.
The tungsten divertor of ASDEX Upgrade has been in operation since 2014 and has consistently experienced mechanical damage, particularly in regions where bulk tungsten tiles are installed. Over the years, several mitigation measures have been implemented to address divertor damage. However, during the most recent experimental campaign, severe damage was observed on a specific type of tile located at a well-defined sector position. The tiles were damaged as a result of a rotation around their longitudinal axis, leading to the formation of a pronounced leading edge, causing a significant increase in local heat loads and severely limited the operational performance of the machine. The observed displacement patterns indicate that the rotation was induced by a positive axial bending moment. In contrast, a similar failure observed in early 2014 was associated with a negative axial moment. These differing failure mechanisms are difficult to explain without detailed numerical analysis.
To address these issues, this work focuses on the development of a comprehensive finite element model of the entire lower divertor, including accurate electromagnetic loading conditions. The model is implemented using Ansys APDL and is extensively benchmarked against data from the numerous magnetic probes installed in the device. In addition, to obtain indirect measurements of the forces acting on the tiles, displacement sensors will be installed at multiple positions to provide insight into the magnitude and distribution of currents induced in the components.
This approach aims to improve the understanding of the underlying failure mechanisms and to support the optimization of divertor design for future fusion reactors.

High Heat Flux testing and Investigation of the Recrystallization Characteristics of Potassium-doped Tungsten fabricated by Spark Plasma Sintering

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

While tungsten (W) is the primary plasma-facing material for the divertor of Fusion devices such as ITER, advanced tungsten-based materials with enhanced performance as armor materials are required for the divertors of next-generation Fusion reactors. Among the key material properties of tungsten, stability against high heat flux and resistance to recrystallization at high temperatures are crucial for the reliable operation of divertors under higher fusion power conditions. A major limitation of the currently available industrial W plates produced through severe thermo-mechanical processes is the pronounced degradation of their mechanical properties upon recrystallization at high temperatures (> 1250 ℃).

In this study, we investigated the thermal stability of potassium (K)-doped tungsten fabricated using the spark plasma sintering (SPS) method and compared the characteristics with those of commercial W plate produced by hot rolling. Fine-grained (~3.2 micro m) microstructure and high strength (Vickers hardness ~460) were achieved along with the improved recrystallization resistance by small amount of K content (0.0033 wt%) and scaled-up K-doped W materials (45 mm in diameter) were successfully fabricated while maintaining homogeneous material properties. Analysis of the heat-treated specimens (up to 1450℃) using electron backscatter diffraction (EBSD) and hardness measurements showed almost no change in K-doped W, whereas significant hardness reduction accompanied by grain growth was observed in the commercial W plate. ITER-type monoblock PFC mockups were manufactured employing the fabricated K-doped tungsten, and the mockups underwent high heat flux (HHF) testing using electron beam facility. The heat flux was increased stepwise, imposing 500 cycles at each heat flux level, and the tungsten surface temperature reached up to 1900℃. Post-mortem analysis of the tested mockups showed distinctive differences in the microstructure, hardness and surface state between the K-doped W and the commercial W plate. The results of the high heat flux testing for the different mockups are presented, and the potential of K-doped tungsten for enhanced HHF components will be discussed.

#13760

Manufacturing & Qualification activities of DTT First Wall small-scale mock-ups

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The current design phase of the First Wall (FW) of the Divertor Tokamak Test (DTT) facility [1] is mainly featured by the qualification activities of small-scale mock-ups, representative of the plasma-facing components (PFCs) involved in the current design of both the Outboard FW (OFW) modules and the Limiter Inboard FW modules (LIFW) [2].
This work presents the roadmap followed for both the manufacturing and the qualification tests of these mock-ups.
In order to qualify the design of the OFW-PFCs with respect to the thermal fatigue, small-scale mock-ups were manufactured by means of additive manufacturing technology. Their design is based on steel-plates provided with internal cooling channels and a tungsten (W) coating on their plasma facing surface (~ 80x100 mm2). Due to two different values of the W-thickness deposited, two different coating technologies were investigated: atmospheric plasma-spray up to 300 ?m and the chemical vapour deposition up to 30 ?m.
Finally, thermal fatigue tests were performed at the HELOKA facility along 5000 cycles at 0.5 MW/m^2, with inlet water conditions relevant for DTT: 60 °C, 4 MPa and mass flow up to 3 kg/s, and the results are presented in this work.
At the inboard side of DTT, each LIFW – covering 10° toroidally – aims to intercept the conducted heat load during both ramp-up and rump-down phases and off-normal operations (i.e. disruption events).
The LIFW module consists of 7 Plasma Facing Units (PFUs): each of them has a design based on a CuCrZr coaxial pipes' system as heat sink and on W-monoblocks as armour material featuring a minimum W-thickness surrounding the outer pipe of 3 mm. With the aim of increasing the heat exchange efficiency of the cooling water, the coaxial annular channel is provided with a helical turbulator, joint to the inner pipe of the LIFW
In order to qualify this design, 8 small-scale mock-ups (~ 400 mm) of the LIFW were manufactured and, with the aim of performing at a later stage high heat flux (HHF) and critical heat flux (CHF) tests taking into account also the kinematic constraints arising from the actual fixing system, at the middle of each mock-up a W-monoblock is provided with a XM19 leg, joined to it by a brazed layer of soft copper.
Finally, both non-destructive and leakage tests were performed for the acceptance of the manufacturing. The results are presented in this work.

Impact of twisted tape spacing on the thermal-hydraulic performance of ITER-like Plasma Facing Units

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The first divertor of the Divertor Tokamak Test (DTT) facility [1] is designed to operate in a wide range of magnetic configurations and plasma scenarios. These characteristics impose stringent requirements on the engineering design of actively cooled plasma-facing components (PFCs). The DTT divertor consists of four targets, two Vertical Targets (VTs) and two Horizontal Targets (HTs), hydraulically connected in series and fully armored by tungsten monoblocks. Within the qualification program, mock-ups representative of the VTs and HTs were manufactured by ENEA and tested in the high heat flux facilities. Thermal fatigue behavior up to 2000 cycles at a heat flux of 20 MW/m2 and the margin to the critical heat flux were assessed. To achieve high operational flexibility and steady-state heat removal capability, the Plasma Facing Unit (PFU) monoblock design was optimized and demanding hydraulic conditions were adopted. Cooling water flows at approximately 12 m/s and 5 MPa inside a CuCrZr tube with an inner diameter of 12 mm and a wall thickness of 1.5 mm. Under such conditions, design aspects usually considered as secondary in less demanding applications, become critical.

In this work, the role of the interspace between the twisted tape and the cooling channel of Plasma Facing units is investigated. During the experimental campaigns in the GLADIS facility, anomalous thermal-hydraulic behavior was observed, characterized by a significant degradation of the heat removal performance. Post-test analysis identified an excessive clearance between the twisted tape and the inner tube surface as the most probable cause of this phenomenon. This interpretation was supported by dimensional analysis and computational fluid-dynamics simulations.

A subsequent high heat flux test, performed after replacing the previous twisted tape with a new one with a larger diameter, confirmed the recovery of the expected performance. Finally, a preliminary evaluation of the allowable manufacturing tolerances for both tubes and twisted tapes is proposed, highlighting the need for dedicated experimental campaigns to validate the evaluated limits.

[1] S.Roccella et al., IEEE Transactions on Plasma Science, 52 (9), 4167-4173.

#13780

Glow Discharge cleaning efficiency analyses in view of RFX-mod2 operation

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

RFX-mod2 is the upgraded version of the RFX-mod experiment, a toroidal plasma confinement device capable of operating in both Reversed Field Pinch and Tokamak configurations. Installed at the Consorzio RFX facility (Padova, Italy), RFX-mod2 has a major radius of 2 m and a minor radius of 0.5 m. The upgrade aims at increasing the plasma volume and optimizing its position with respect to the active and passive components of the Magnetohydrodynamic (MHD) control system, enabling the exploration of plasma regimes with higher temperature and improved stability.
As in all magnetic confinement devices, the initial operational phase requires effective first wall conditioning, typically performed by Glow Discharge Cleaning (GDC). The GDC system of RFX-mod2 has been significantly upgraded and now includes eight new fixed electrodes, which will act as Plasma Facing Components (PFCs), together with two movable electrodes designed for the boronization procedure. The new configuration is intended to improve plasma uniformity and enhance the overall efficiency of the wall conditioning process.
This work presents the first experimental results obtained during hydrogen and helium discharges using the new fixed electrodes. Experimental campaigns have been carried out in a dedicated facility where representative first wall materials for RFX-mod2 — including alumina-coated copper components, graphite tiles, and Torlon 5030 support rings — are installed. Plasma temperature and density measurements have been performed as a function of probe position and electrode feeding power. In addition, Residual Gas Analysis (RGA) and optical spectroscopy diagnostics have been used to assess the effectiveness of the GDC process. The results provide key information on the operational reliability and conditioning performance of the upgraded GDC system, contributing to the optimization of future RFX-mod2 plasma-wall conditioning campaigns.

Design and Prototyping of a Liquid-Metal CPS Divertor Tile for COMPASS-U

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Liquid-metal (LM) plasma-facing components are a promising route to enable self-healing surfaces that can withstand extreme divertor heat/particle fluxes. We report the design and development of a capillary porous system (CPS) divertor tile prototype for COMPASS-U, located at IPP Prague, that is compatible with lithium or tin operation. Two CPS embodiments integrated into tungsten (W) are pursued in parallel to compare performance between technologies: (i) an additively manufactured (AM) tile produced by laser powder bed fusion with a surface-layer CPS created via selective laser melting, and (ii) a conventionally machined W body hosting a flexible CPS mesh currently being developed by ENEA.

The AM variant targets a CPS microstructure with ~37% porosity and ~23 μm effective pore size – as previously achieved at DIFFER [1] – to provide sufficient capillary pressure for LM retention while minimizing thermal resistance to the W heat sink. Monolithic, near-surface porosity reduces joining interfaces and allows geometric tailoring of capillary pathways adjacent to cooling-proximal structures. The mesh-based variant emphasizes modularity, rapid iteration, and maintainability; CPS mesh parameters (weave/knit, thickness, pore size distribution) will be selected to optimise wicking performance and physical robustness while mitigating loss of LM under magnetic loading. Both concepts incorporate a reservoir to allow resupply of LM to the plasma-facing surface under LM erosion.

Pre-qualification will include inspection of microstructure of printed components to ensure structural integrity of bulk-W regions, and of successful infiltration of the CPS layer and feeding structures with LM. This will be followed by investigation of wetting and capillary action at high temperatures under vacuum, and high-heat-flux (HHF) exposure in devices such as HADES to examine thermal performance and LM replenishment. Following HHF validation, the prototype will be integrated for testing in COMPASS-U, with a comparative assessment of AM-CPS versus mesh-CPS performance against common figures of merit (temperature at flux, LM loss rate, impurity influx, and post-exposure integrity).

This dual-path effort is expected to clarify the trade-space between additive and mesh CPS implementations, de-risk LM-CPS divertor concepts on COMPASS-U, and inform scaling toward higher-duty operation in next-step devices.

[1] J. G. A. Scholte et al., “Design and Gladis testing of a liquid tin divertor module prior to exposure in ASDEX upgrade,” Nuclear Materials and Energy, vol. 37, p. 101528, Dec. 2023. doi:10.1016/j.nme.2023.101528

#13800

Electromagnetic loads mapping through the SCILLA force density interpolative profiles

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

In the design of fusion devices several concerns are posed on the plasma facing components due to possible transient and heavy electromagnetic loads in case of disruption events. These critical loading conditions, which are caused by the interaction of the eddy and halo currents with the total magnetic field, justify the need for coupled electromechanical analyses. However, due to the diverse nature and properties of the Finite Element Method (FEM) formulations employed by the electromagnetic and the mechanical analyses, very different meshes are used in the two analyses making the coupling workflow of relevant importance.

The electromagnetic loads mapping aims to preserve the resultant forces and momenta computed in the electromagnetic domain within the transferring process from the source points – usually the electromagnetic elements' barycentres, to the target points - usually mechanical elements' nodes. Most of the algorithms found in the literature are usually quite sensitive to the positioning of the mechanical nodes with respect to the electromagnetic element barycentre.

In this context, the Structural Completely Independent Linear Loads Assumption (SCILLA) method [1] preserves the resultant forces and momenta for each electromagnetic element through full linear interpolation functions, which are also suitable for mapping force densities. This approach better reflects the electromagnetic computation and further increases the possible variability of the forces across the elements due to the distribution of the volumes assigned to the mechanical nodes. The present work further investigates this feature. The algorithm proposed is based on setting the method of undetermined coefficients for calculating exact integrals within the electromagnetic element sampling the integrand at the target nodes. The resulting possibly non-linear distribution of the forces across the electromagnetic element could result in different values and localization of the mechanical stresses, depending on the positioning of the constraints.

Keywords: electromagnetic loads mapping, force density, multiphysics analyses

[1] G. Camera, N. Isernia, G. Di Gironimo, F. Villone, "Electromagnetic loads mapping: the SCILLA method", submitted to "Finite Elements in Analysis and Design"

#13801

Adjoint-Based Optimisation Framework for Plasma Edge Simulations with AI-Assisted Sensitivity Modelling

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The design of plasma-facing components in magnetic confinement fusion devices requires accurate modelling of complex plasma–material interactions in the boundary region of tokamaks. In particular, the divertor system must manage extremely high heat and particle fluxes while ensuring acceptable material lifetime and reliable reactor operation. Numerical tools such as the SOLPS-ITER code suite are widely used for plasma edge simulations and play a key role in the interpretation of experiments and the design of next-generation fusion devices, including ITER and DEMO. However, the strong coupling between multiple physical models, including fluid plasma transport and kinetic neutral particle simulations, results in computationally expensive multi-physics workflows that significantly limit systematic optimisation and design exploration.

This work presents the development of a computational framework that enables efficient optimisation of complex plasma edge simulations. The framework integrates automated grid handling, adjoint sensitivity analysis, and optimisation control within the SOLPS-ITER workflow, transforming the simulation chain into a design-oriented computational environment. Adjoint methods allow efficient computation of sensitivities of objective functions with respect to a large number of design parameters at the cost of only a single additional simulation, making them particularly attractive for high-dimensional optimisation problems in fusion modelling.

A central element of the framework is the development of automated grid generation and mesh morphing strategies that support iterative modifications of divertor geometry and magnetic configuration during optimisation. These tools ensure consistent grid quality and enable efficient propagation of sensitivities across the full simulation chain. Algorithmic differentiation techniques are integrated to automatically generate forward and adjoint sensitivity information within the multi-physics plasma edge model.

In addition, machine learning approaches based on deep neural networks are investigated to accelerate adjoint computations by learning relationships between local plasma variables and adjoint sensitivities. Such surrogate models have the potential to reduce computational cost in large-scale optimisation studies while preserving the physical fidelity of the underlying simulations.

The presented framework represents a step toward transforming plasma edge modelling from a purely diagnostic tool into an integrated optimisation platform capable of supporting design studies for future fusion reactors.

#13807

Construction and operation of CRAFT high heat flux test facility

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The plasma-facing components (PFCs) in fusion reactors, specifically the divertor and the first wall of the blanket, are subjected to extreme thermal loads from high-temperature plasmas, imposing stringent demands on both armor materials and joining quality. To ensure the reliable operation of these components within the fusion reactor system, rigorous performance evaluations under service conditions must be conducted. Among these, high heat flux testing using electron beams is considered an essential qualification step in the development and mass production of divertor and first wall components. In order to verify the design and manufacturing process of the CFEDR divertor and blanket components, the CRAFT project proposed to build an advanced high heat flux test facility. In the design, it consists of two electron-beam guns: a 60kW e-beam gun and an 800kW e-beam gun. To conduct the tests of full-scale divertor and blanket components, a vacuum chamber is designed with a diameter of 3 meters and a length of 4 meters. To simulate its working conditions, the test facility is equipped with a high-temperature and high-pressure cooling water loops, which can achieve a maximum pressure of 15.5MPa and a maximum inlet temperature of 285℃.

Now, the CRAFT high heat flux test facility (referred to as C-HEAT) has completed construction and passed the acceptance tests, and has officially entered the operational phase. During the construction period, we solved a series of technical problems. For example, in order to achieve precise temperature measurement of the test samples, we successively installed contact-type thermocouple systems, non-contact surface infrared camera, and multi-band pyrometers. Based on the complete temperature measurement system, we carried out tests and calibrations for the uniformity of the heat flux loading of electron beam. Additionally, to monitor the damage behavior, a high-speed camera and DIC strain measurement system have been designed and built, which can monitor the stress and cracking behavior on the surface and interface of the sample in real time.

Based on the construction and operation of C-HEAT, we have successively completed the high heat flux testing tasks for components such as the upper divertor on EAST, the BEST divertor, and the shielding blanket first wall. This has supported the development of design and manufacturing technologies for PFCs.

Effects of the Streamwise Extent of Periodic Computational Domains for LES Simulations on Turbulent Flow and Heat Transfer of the AV2 Geometry in the HETREX-PT Experiment

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The HCPB blanket concept for EU-DEMO fusion reactors employs high-pressure (8 MPa) helium gas as coolant for the plasma facing first wall. Up-to-date estimations for the total maximum of heat flux go up to 0.73 MW/m^2. Higher short-term transient loads are possible. Long term objectives of the qualification work for numerical simulations for the helium cooled First Wall channels of DEMO are to provide a consistent, fully validated and practically feasible (in terms of computing resources) numerical approach to provide simulations of complete rib enhanced FW channels/ full-size blanket components. The HETREX-PT (HEat TRansfer Enhancement eXperiments - Pressure, Temperature) experiment provides experimental data with the primary objective to directly compare a number of FW channel geometry options in terms of heat transfer and pressure drop.The rib geometry “AV2” is used in the HETREX-PT experiment and in a modified form (AV2s) in the FW-PMU experiment. This geometry is the main candidate for use in FW cooling channels (with the current manufacturing approach). LES data for the AV2 rib geometry is needed to benchmark RANS approaches for the HETREX-PT experiments (heat transfer, pressure loss). LES simulations have been performed for a Reynolds number of 50k with precisely specific boundary conditions from a HETREX experiment (air 4 bar) for calculations areas of sections with different numbers of ribs. Due to the demanding grid requirements of LES, periodic boundary conditions in the streamwise direction were applied to approximate fully developed flow conditions. These simulations showed significant differences in turbulent flow and heat transfer indicating the need of bigger streamwise extent of the simulation domain. Sections of different length are analysed and compared. In addition to Nusselt numbers and friction factors, the evaluation of time-averaged fields/profiles of velocity and Reynolds stresses (uiuj and uiT'), as well as secondary flow, vortex configuration, and length of the separation area are analysed. The integral length scales are compared. The size of the necessary calculation area is evaluated with these findings and discussed.

Progress in development and application of CFC based components for EXL-50U

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

EXL-50U is a spherical tokamak hydrogen-boron fusion experimental device independently designed and constructed by China's ENN Group. The performance of the Plasma-Facing Components (PFC) on the EXL-50U device directly impacts its operational parameters, steady-state operation capability, and plasma confinement level. To address the challenges posed by high heat fluxes, this work focuses on the independent research and engineering application of high-performance Carbon Fiber reinforced Carbon matrix (CFC) based plasma-facing components.

This report systematically presents the CFC-based components developed for the EXL-50U device, primarily including the divertor target plates, limiters, and key components of the first wall. It emphasizes the development of CFC materials (characterized by high strength, high purity, high density, high thermal conductivity, and plasma irradiation resistance), integrated 3D weaving and forming technology for CFC materials, and CFC material joining techniques. Through process optimization, the following material properties have been achieved: density of 2.0 g/cm^3, tensile strength of 190 MPa, ash content below 10 ppm, thermal conductivity of 600 W/(m·K), and porosity below 1%.

Furthermore, the report will present the results of comprehensive performance tests and evaluations conducted on these CFC-based components. This includes thermal fatigue testing under high heat flux (e.g., 5 MW/m^2 under electron beams and ion beams), thermomechanical property analysis, and studies on material surface behavior under combined high heat flux and particle irradiation. These assessments provide critical data to support the safe application of these components on the EXL-50U device.

Finally, the preliminary application status and performance of the first batch of CFC-based components during experimental campaigns on the EXL-50U device will be reported. This includes an evaluation of their heat load capacity under actual plasma discharge conditions, erosion behavior, and their impact on plasma discharge performance (such as impurity control and boundary parameters). The related work has accumulated a significant technical foundation and engineering experience for the future design and manufacturing of high-performance, long-lifetime plasma-facing components for fusion devices.

#13839

Numerical Simulation and Experimental Testing of a Reduced-Scale Additive Manufactured TPMS Divertor Tile for Wendelstein 7-X

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

One of the key engineering challenges in nuclear fusion is the efficient removal of high heat flux insisting on plasma-facing components such as the divertor and first wall. In the W-7X stellarator, plans are underway to replace the current carbon-fiber composite (CFC) divertor with a tungsten-based flat-tile design to demonstrate the plasma performance of a stellarator with a low tritium-retention plasma facing material [1]. In a previous study, we explored an alternative cooling strategy for divertor tiles which deploys a Triply Periodic Minimal Surface (TPMS) [2] structure manufactured by Additive Manufacturing (AM). A gyroid-based lattice with unit cell dimensions of 10 mm × 10 mm × 5 mm demonstrated, via numerical simulation, the ability to withstand heat fluxes up to 10 MW/m^2 [3]. Based on that, a test specimen was manufactured in CuCrZr using laser powder bed fusion (LPBF) at Fraunhofer IGCV. It measures 140 mm in length, 10 mm in height, and 50 mm in width, with additional 20 mm inlet and outlet pipes incorporated for integration via VCR connectors through copper
galvanization. Due to the inherent surface roughness of AM metallic components, experimental pressure drops are expected to be approximately twofold higher than those predicted by smooth-wall simulations. In order to quantify the discrepancy between numerical and real pressure drops, we both numerically and experimentally investigated a reduced-scale TPMS sample that replicates the layered structure of the full tile design. The thermal performance of the sample was also investigated using integrated electrical resistances.

Numerical simulations are performed using the Computational Fluid Dynamics (CFD) software STAR-CCM+, with a Realizable k-ε two-layers turbulence model in addition to a dedicated roughness model. The results are benchmarked against experimental data obtained from a dedicated experimental setup
at Politecnico di Milano. This methodology ensures an extended database for model validation and enables high-fidelity performance predictions for AM-integrated cooling systems.

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Towards Functionally Graded W-CuCrZr Cellular Heat Sinks for Plasma Facing Components: Residual Stress Analysis and Homogenization-Based Design

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Plasma Facing Components (PFCs) for next-generation tokamak reactors must withstand extreme thermal loads while maintaining structural integrity over long operational cycles. Tungsten is the reference material for the plasma-facing armour, while CuCrZr alloys are typically employed as heat sink materials due to their high thermal conductivity. However, the interface between these materials represents a critical design challenge because of the large mismatch in their coefficients of thermal expansion, which can lead to severe thermally induced stresses and reduced reliability of the joint.

A proposed design solution involves the use of a tungsten cellular architecture acting as an intermediate region between the armour and the heat sink. In particular, Kelvin-type lattice structures manufactured in tungsten and infiltrated with CuCrZr are being investigated to enhance heat dissipation while mitigating thermo-mechanical incompatibilities. In this concept, the tungsten lattice can be produced monolithically with the armour, while the CuCrZr infiltration provides the thermal pathway towards the cooling circuit. The complex multiphase and cellular nature of the structure makes direct parametric finite element (FE) modelling computationally expensive, especially for design optimization studies. To address this challenge, a homogenization framework was developed to derive effective thermo-structural properties of the infiltrated cellular material from a representative volume element. The framework condenses the temperature-dependent anisotropic behaviour of the composite architecture into effective tensorial properties, enabling smeared modelling of these regions in large-scale FE analyses of PFCs. This approach allows efficient parametric studies, design exploration and optimization. However, homogenized models inevitably neglect microscale phenomena. For this reason, detailed FE analyses of the representative cellular geometry were performed to investigate key mechanisms associated with the infiltration process, including the development of residual stresses during cooling and the potential disruption or separation of CuCrZr ligaments within the lattice. Such defects can significantly affect thermal transport pathways and therefore the effective conductivity of the composite material.

Although the numerical model idealizes the geometry and does not fully capture process-induced irregularities such as porosity, cracks or incomplete infiltration, the analysis provides valuable insight into the thermo-mechanical behaviour of the infiltrated structure and supports the interpretation of experimental observations. The developed framework is not limited to Kelvin cells and can be generalized to arbitrary periodic cellular topologies, enabling future optimization of PFC architectures based on spatially varying lattice networks tailored to locally optimized thermo-structural performance.

Modeling of material damage by runaway electrons in tokamaks

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Runaway electrons (RE) deposit energy into matter volumetrically which, combined with the fast timescales and high energy densities, often results in explosive response of plasma-facing components (PFC). The consequences of RE-induced PFC damage for tokamak operation range from deep volumetric melting and in-vessel loss-of-coolant accidents to debris expulsion and subsequent nonlocal damage [1-3]. Explosive events upon RE impacts have been observed in multiple machines, accidentally as well in controlled experiments [4].

The modelling chain for material damage starts with RE simulations for prediction of the deposition patterns on the wall as well as of the energy and the incidence angles of the incoming electrons. These serve as input to Monte Carlo simulations of the energy deposition into matter providing a volumetric heat source for the thermo-mechanical modelling of the PFC response. Until recently, the development of the thermo-mechanical models has been impeded by the absence of controlled experiments [4].

The first dedicated RE-induced damage experiments with graphite samples were carried out in DIII-D in 2022 [5]. The controlled variation of the energy deposited, as well as camera observations of the debris, allowed for validation of the failure and material expulsion models for brittle materials with no liquid phase [6-9]. The failure model was also shown to give reasonable results for cumulative accidental damage of boron-nitride tiles in WEST [10].

The problem of material expulsion in the case of tungsten (W) PFCs is far more complex due to the presence of the liquid phase and the high temperature ductile behavior [4,10]. Challenges facing the development of such models for W will be discussed, along with initial results on the first controlled experiment on RE-induced damage of the W instrumented performed in WEST in 2025 with fast IR and visible camera diagnostics.

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[8] A. Kavroulakis et al, in preparation
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Field-line-guided design of self-shadowed W7-X divertor targets for enhanced particle exhaust

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Improving particle exhaust while maintaining reliable power handling is a key objective for future divertor upgrades of Wendelstein 7-X (W7-X). The island divertor topology motivates concepts aiming at increased neutral compression and pumping effectiveness, but it also imposes strict geometric constraints: target surfaces must be aligned with the local field-line incidence, avoid optical leading edges, and remain compatible with realistic engineering tolerances and integration boundaries. As part of a broader effort to develop practical design tools for a next-generation W7-X divertor concept with significantly enhanced particle exhaust, we present an engineering workflow that generates and evaluates “island-submerged” target geometries (i.e. targets that remain inside the island over their full toroidal extent, and do not “dip” in or out).

The workflow starts from a user-defined geometry in a given toroidal cross-section, which is then advanced toroidally in small increments. At each increment, the field-line mapping defines how points move on their respective flux surfaces between neighboring toroidal planes, and an additional in-surface shift is applied immediately so that the newly created surface segment matches a prescribed local incidence angle. As the field-line pitch varies along the island, the required shift magnitude is adjusted at every step based on the requested incidence and the local field-line pitch. Repeating this procedure produces a smooth island-submerged target surface whose incidence is controlled by design while remaining consistent with the underlying flux-surface topology. The resulting surfaces can be exported in CAD/mesh-compatible formats to enable fast iteration between physics-driven shaping and engineering integration studies.

In addition, we outline and illustrate a few design principles for island-submerged targets in W7-X that are directly relevant to particle-exhaust-oriented upgrades, such as: (1) the poloidal angular placement relative to the island separatrix and LCFS; (2) incidence-direction control, enabling more relaxed manufacturing and assembly tolerances; (3) intentional self-shadowing, so that potentially exposed downstream edges are shadowed by the main target surface for expected incidence directions; and (4) robustness across a representative set of W7-X magnetic configurations and operational scenarios.

These design principles can be embedded into an automated optimization loop for W7-X island-submerged targets, providing a practical pathway to iterate divertor concepts aimed at improved particle exhaust without compromising power-handling capability.

#13883

Coupled Thermal–Structural Simulation of W and W/Mo Coatings on Graphite under High Heat Flux

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Thermal stability of tungsten (W) and tungsten/molybdenum (W/Mo) coatings on graphite under high heat flux conditions remains a critical issue for plasma-facing components in fusion applications. Understanding the response of these coatings to extreme thermal loads is essential for predicting surface damage and ensuring component reliability.

In this work, coupled thermal–structural simulations are performed using ANSYS Mechanical to investigate the behavior of W and W/Mo coatings under high heat flux. Both standalone thermal simulations and fully coupled analyses are considered to capture temperature distribution, thermal stresses, and potential structural damage. The modeling approach is designed to complement experimental investigations on graphite substrates, providing insight into surface emissivity measurements and supporting validation of experimental data.
The study aims to quantify the extent to which high heat flux can induce structural degradation in the coatings and to assess the role of material combinations in mitigating damage. The results offer guidance for optimizing coating design and for interpreting experimental findings on graphite, contributing to the development of resilient plasma-facing surfaces in fusion devices.

Design advancement of the DTT First Wall through the integration of laser powder bed fusion technology

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The design of the DTT First Wall (FW) is under finalization, paving the way to the upcoming procurement stage. Recent qualification activities have identified the laser powder bed fusion
(L-PBF) technology as a highly promising additive manufacturing method for the steel-based plasma-facing components that form the FW at the outboard and top sides. This approach offers enhanced design flexibility and mechanical properties. In the ambit of the engineering design structural verification, the most relevant loading conditions were considered. These include the deposition of thermal power during plasma operation and the generation of electro-magnetic loads during disruption events.
This contribution presents a refined configuration of the FW modules, developed after extensive design optimization and simulation effort. The design was carefully adapted to meet the specific requirements of L-PBF manufacturing technology, ensuring compatibility with large-scale production, remote maintainability requirements and interfaces with in-vessel systems behind the FW modules. Subsequently, a detailed assessment of structural stability against the defined loading conditions was performed. These results provide the groundwork to proceed and conclude the engineering design activities.

Qualification of Additively Manufactured CuCr1Zr Heat Sinks for the Wendelstein 7-X Divertor

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Additive manufacturing (AM) is gaining increasing relevance in the nuclear fusion industry due to its capacity for application-specific component design. Layer-based fabrication offers the design freedom to integrate advanced component functionalities, thereby enabling effective management of extreme thermo-mechanical loads during plasma operation.

This potential is being considered for the redesign of the carbon-based high-heat-flux (HHF) divertor for Wendelstein 7-X (Max Planck Institute for Plasma Physics, Greifswald). As the world's largest stellarator with modular superconducting coils, Wendelstein 7-X currently operates with carbon-fiber-reinforced carbon (CFC) plasma-facing surfaces. To demonstrate stellarator performance with tritium-compatible, reactor-relevant materials, a project was initiated to develop a tungsten (W)-based HHF divertor concept. The proposed design consists of a single additively manufactured CuCr1Zr (CW106C) heat sink armored with W or a W-heavy alloy (W95NiFe). It has been designed to sustain steady-state heat fluxes of up to 10 MW/m^2 under water-cooling conditions. In contrast to the previous CFC divertor, which required 82 manufacturing and 44 inspection steps per target module, the W-based concept aims to significantly reduce production and qualification efforts by minimizing weld seams and consolidating functions, inspections, and assembly steps into a single additively manufactured heat sink with integrated manifolds.

In this contribution, Laser-based Powder Bed Fusion (PBF-LB/M) is utilized to fabricate fully integrated CuCr1Zr heat sinks featuring customized, high-efficiency cooling channels. More specifically, this work focuses on the material qualification process of additively manufactured CuCr1Zr, aiming for its application in industrial-scale nuclear fusion components. The evaluation encompasses the optimization of powder characteristics, process parameter development to achieve densities exceeding 99.8%, and the verification of helium tightness (mbar·l/s at 150 °C and 28 bar) for thin-walled structures. Furthermore, various heat treatment strategies, including Direct Age Hardening (DAH), were investigated to enhance mechanical strength and thermal conductivity. Additional analyses addressed the manufacturability of delicate geometries, their thermo-hydraulic performance with regard to heat transfer and pressure drop, and the production of small-scale mock-ups. Finally, HHF validation of first demonstrators confirms the technological readiness of additively manufactured CuCr1Zr heat sinks for the upgraded Wendelstein 7-X divertor.

A Physics-Informed Foundation for a Digital Twin of Hydrogen-Isotope Retention in Tungsten Plasma-Facing Components

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Hydrogen-isotope retention in tungsten plasma-facing components remains a major fusion technology challenge because the behaviour of real tungsten is not fully captured by idealized atomistic models or isolated experiments alone. Perfect-crystal calculations provide fundamental adsorption and migration energetics, but practical tungsten surfaces are rough, defected, irradiation-damaged, and dynamically evolving under plasma exposure and neutron loading. Conversely, experiments probe realistic behaviour but often convolve multiple coupled mechanisms, making it difficult to isolate atomistic origins or transfer results predictively across conditions. This creates a gap between mechanistic understanding and predictive retention assessment. Bridging this gap requires physically grounded atomistic inputs that capture realistic surface features while remaining computationally tractable. To address this, we are developing a physics-informed multi-scale workflow as a foundation for a digital twin of hydrogen-isotope retention in tungsten.

The broader framework is designed to couple atomistic calculations, OpenMC-based neutronics inputs, reduced transport descriptions, and a physics-informed learning layer that closes the loop between first-principles physics and rapid retention prediction under reactor-relevant conditions. Within this workflow, the present study establishes the atomistic foundation by resolving the earliest stage of hydrogen uptake at tungsten plasma-facing surfaces.

Density functional theory calculations were performed on layered tungsten models, using W (110) as an idealized low-index terrace and stepped W (211) as a simplified representation of atomic-scale roughness and under-coordinated environments expected in realistic tungsten plasma-facing materials. On clean W (110), H adsorption at the surface is strongly favorable, with an adsorption energy of −1.09 eV, whereas insertion into a subsurface site is endothermic at +0.89 eV, corresponding to a 1.98 eV thermodynamic penalty for inward transfer. In contrast, stepped W (211) exhibits three distinct adsorption minima spanning 0.44 eV, with adsorption energies of −0.87 eV for a step-edge bridge site, −0.64 eV for a cross-layer bridge site, and −0.43 eV for atop adsorption, indicating a distribution of binding strengths across roughened surface environments.

These results show that hydrogen uptake in tungsten is strongly controlled by local surface topology. W (110) stabilizes surface residence and disfavors spontaneous subsurface ingress, while stepped W (211) presents a broader and weaker adsorption landscape. The present energetics therefore provide physically grounded first-principles descriptors for hydrogen uptake and form part of the atomistic dataset feeding the downstream neutronics- and learning-integrated digital-twin workflow for predictive hydrogen-isotope retention modelling.

#13922

Development of an Ultrasonic Inspection for qualification and acceptance Criteria for Advanced Tungsten Plasma-Facing Units of the Wendelstein 7-X Divertor

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Within the framework of the divertor design activities for Wendelstein 7-X, increasing attention is being devoted to the qualification of Plasma Facing Units (PFUs) manufactured from high-performance refractory materials. In this context, the present work addresses the role of the non-destructive analysis by ultrasonic testing (UT) as a pivotal tool both for the qualification of the design and the manufacturing process and for establishing technically sound acceptance criteria for component series production.

Although ultrasonic inspection techniques are well established in industrial practice, the adoption of advanced materials and novel material combinations introduces non-trivial measurement challenges. For instance, differences in the ultrasonic response have been observed between pure tungsten (W100) and the tungsten alloy W95NiFe. In addition to a difference in the sound propagation velocity, and consequently in the acoustic impedance, a distinct reflection pattern has been also identified.

This work investigates whether and to what extent such discrepancies can be attributed to microstructural features, including grain size distribution, phase composition, and binder morphology. It further examines whether these microstructure-dependent variations influence attenuation characteristics and defect detectability, especially in the presence of microcracks and sub-millimetric discontinuities.

The systematic analysis presented here seeks to quantify the impact of material-specific ultrasonic signatures on flaw detection capability under ultrasonic immersion testing conditions.

In addition, by correlating ultrasonic parameters with metallographic characterization and controlled artificial reflectors, the study evaluates detection thresholds and sizing accuracy for representative defect classes relevant to PFU service conditions.

Beyond the immediate goal of supporting divertor component qualification for W7-X, this work aims to contribute to a broader reassessment of the capabilities and intrinsic limitations of immersion ultrasonic testing when applied to advanced plasma-facing materials. Rather than providing definitive answers, the outcomes are intended to offer a technically grounded reference framework that may assist in the formulation of acceptance criteria and inspection protocols tailored to tungsten-based PFUs. By clarifying the interplay between material innovation and ultrasonic response, the study seeks to represent a meaningful step toward enhancing structural reliability, manufacturing reproducibility, and long-term operational safety in high-performance stellarator divertors.

Thermal-Hydraulic Analysis and Design Validation of the DTT Outer Target Test Frame

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Within the framework of the EUROfusion Work Package Divertor (WP-DIV), focused on the design, prototyping and construction of the Divertor Tokamak Test (DTT) divertor module, a dedicated test frame for the Outer Target (OT) has been developed. The test frame is intended to support the realization of a prototype aimed at assessing the thermal fatigue performance of the OT component under realistic mechanical constraint conditions through experimental testing in High Heat Flux (HHF) facilities. In addition, the tests will enable the qualification of manufacturing processes of the Plasma Facing Units (PFUs), including the welding of monoblocks to CuCrZr pipes and the welding of pipes to steel manifold, fully representative of the configuration foreseen for the divertor module.
To ensure compatibility with HHF facilities, the test frame design integrates three PFUs mounted on a dedicated support structure and cooled in parallel. The experimental campaign foresees thermal cycling up to 20 MW/m2 for 5000 cycles and up to 24 MW/m2 for 500 cycles, with water coolant at 60 °C and 50 bar.
Thermo-hydraulic and mechanical performances of the test frame design were assessed through numerical analyses. Computational Fluid Dynamics (CFD) simulations, performed with ANSYS CFX software, have been carried out to investigate flow distribution, pressure drops and localized losses induced by twisted tapes inserted into the tubes to promote swirl flow. Mechanical analyses were conducted to assess the structural integrity of the system and to support design validation.

#13939

Divertor PFUs with advanced-W plasma facing materials

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten is considered the most promising plasma facing material (PFM) for tokamak in-vessel components. Above all, W is the preferred PFM of the most critical components such as the ITER water-cooled divertor target. In the case of the ITER divertor plasma facing units (PFUs), pure W monoblocks are used as armour material with Oxygen Free Copper as coupling interlayers between W monoblocks and a CuCrZr ITER Grade (CuCrZr-IG) cooling pipe, which works as heat sink material. In this divertor monoblock configuration, pure W has performed well at high heat flux (HHF) tests, under both cyclic thermal fatigue up to 20 MW/m2 and overload (25 MW/m2) tests.

Despite its positive thermophysical properties, pure tungsten has intrinsic drawbacks that might be a limiting factor for applications in future fusion power plants (FPPs). For instance, W is subjected to recrystallisation at high temperatures which results in a loss of strength. In addition, the intrinsic brittleness of pure W below the Ductile-to-Brittle-Transition-Temperature (DBTT) can generate cracks enhancing the risk of loss-of-coolant accidents (LOCAs). In case of LOCA combined with air ingress, tungsten armour will be eroded at high temperature with the formation of volatile and radioactive species of tungsten oxides.

In the context of the Work Package Materials (WPMAT) of the EUROfusion consortium, advanced-W materials have been developed to overcome the intrinsic drawbacks of pure W. Several small-scale ITER-like divertor PFU mock-ups have been produced using advanced armour materials. Advanced-W monoblocks have been manufactured respectively using sintering techniques, powder metallurgy and diffusion bonding. The Cu interlayer was fabricated by casting and the Cu/advanced-W blocks were joined to the CuCrZr-IG pipe with a process based on solid state diffusion bonding, named Hot Radial Pressing.

This work gives an updated overview on using both composite W materials and new W-alloys as divertor PFMs. Tungsten fiber reinforced tungsten composite (Wf/W) armours were capable to reduce the crack propagation by exhibiting a pseudo-ductile behaviour. The addition of chromium and yttrium generated self-passivating W alloys, which protect the underlying W material in case of LOCAs with air ingress. Potassium-doped tungsten divertor mock-up has demonstrated to withstand up to 2000 HHF cycles at 20 MW/m2 without showing embrittlement. The results of this study help highlight which types of advanced-W material could substitute pure W as divertor PFM of future FPPs.

#13960

Density and pressure estimation of He trapped in nano-bubbles inside tungsten matrix using S/TEM-EELS measurements

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

In ITER, the magnetic confinement nuclear fusion reactor, the tungsten (W) divertor is the main plasma-facing component and will be exposed during plasma operations to extreme fluxes of helium (He) and hydrogen isotope ions as well as high thermal loads. Despite tungsten's relevant thermo-mechanical properties (e.g. high melting point and erosion resistance), the low solubility of He in W leads to the formation of near-surface He nanobubbles. These bubbles severely damage the microstructure, particularly when they burst, potentially creating what it is referred to as "fuzz" and consequently degrading the properties and lifetime of the material. This work seeks to expand our understanding of bubble and fuzz formation in the inner walls of fusion reactor tokamaks, resulting from the He-W interaction.

Here, we study the effects of the irradiation parameters on the He–W interaction, by mimicking some of the divertor's exposure conditions in controlled laboratory plasmas. Polycrystalline W samples were thus exposed to He plasmas with various ion fluxes, surface temperatures, or W defect density. The size, shape and He density of bubbles, is then analyzed as a function of these parameters, providing us with a full He inventory. Bright-Field Transmission Electron Microscopy imaging was performed on FIB lamellae to identify individual bubbles and measure their density, size and shape. STEM-EELS measurements were then carried out to acquire the He K-edge signal for each individual bubble.

An increase in the average size of bubbles was observed as a function of the temperature, while a decrease in defect density appears to result in smaller bubbles on average, at similar surface temperatures. On the other hand, EELS measurements provided us with the He K-edge energy and intensity, which were used to estimate the He density using the framework developed by Walsh et al.. These densities were thus analyzed for different irradiation conditions, as a function of the bubble size, as well as the energy shift due to confinement in the bubble. This energy shift appears to increase as a function of the He density, in accordance with theoretical predictions for W, as well as previous experimental results in other materials. However, it reaches unprecedented values of over 3 eV in some bubbles. Finally, these results were then compared to theoretical models and were further used to estimate the pressure inside the bubbles.

#13974

Assessment of the DTT Plasma Facing Units under EM loads

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

The Divertor Tokamak Test (DTT) facility is the Italian experimental tokamak that plays a central role in the European Fusion Roadmap for studying heat and power exhaust issues. It is designed to develop alternative divertor solutions in edge-plasma scenarios relevant to ITER and future Fusion Power Plants (FPPs), thereby bridging the gap between these machines and current devices in this field.
The first DTT divertor consists of 54 modules with three plasma facing components: an inner target (IT), central target (CT) and an outer target (OT). IT and CT are composed by seven Plasma Facing Units (PFUs), while the OT by nine. The PFUs are based on the ITER-like design that is, copper alloy tubes covered with tungsten monoblocks. These PFUs are mounted on interface plates screwed to the cassette body. Each PFU is anchored to the interface plate at specific monoblocks with a perforated C-shaped leg that, via a pin-slot interface, connects the monoblock to the interface plate. This coupling limits the movement of the PFUs toward the plasma without preventing elongation due to temperature increases. This work discusses the verification and qualification analysis of this attachment system, which, although conceptually similar to that of the ITER divertor, differs in terms of resistant cross-section and loads.
As is known, electromagnetic (EM) loads resulting from disruptions are the most demanding in terms of mechanical stresses for plasma-facing components. Once the most demanding disruption event for this component was determined, the first step of the analysis consisted in using a beam model of the PFUs that allowed for the simple and rapid determination of the loads on the attachment systems of the PFUs to the cassette. The beam elements with multiple cross-sections simulate both tubes and monoblocks, thus accounting for a good approximation of the actual stiffness of the components. The EM forces, calculated as loads per unit length, were distributed along the axis of the PFU. The appropriate degrees of freedom were constrained at the position of the attachment systems to the cassette. This model was used to calculate the constraint reactions used as input to size the attachment system and evaluate the forces at the interface between the monoblock and the legs. The verification was conducted according to the codified rules of SDC-IC v3. The results obtained were compared with the tensile tests on the monoblock conducted at the ENEA laboratories in Frascati.

#13992

Thermal Management and Performance Evaluation for Capillary Porous System Based Liquid Metal Divertor

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

To support high parameter and long pulse fusion operation, divertor must withstand severe steady-state heat loads and extreme transient thermal shocks. Conventional solid divertor therefore face persistent challenges including melting, cracking, re-deposition, which induced limited lifetime. Capillary porous system based liquid metal divertor (CPS-LMD), which combine self-healing surfaces, passive replenishment, phase change heat removal, and vapor shielding capability, have emerged as a promising route for next generation fusion power exhaust. However, their thermal behavior is governed by strongly coupled multi-scale processes, including capillary transport, interfacial evaporation, vapor diffusion, porous conduction, and external cooling, and a unified framework suitable for both physical interpretation and engineering design has remained lacking.

This work establishes a theory to application framework for CPS-LMD thermal management by organizing performance around two complementary dimensions: heat transfer efficiency and heat transfer limit. For efficiency, a cross-scale equivalent thermal resistance network is developed to decompose the overall heat removal process into porous conduction, nonequilibrium interfacial evaporation, vapor side mass transport, radiative loss, and heat sink removal, thereby providing a system level description of phase change heat transfer in high temperature rarefied environments. For the heat transfer limit, dryout is taken as the key failure mechanism, and unified criteria are formulated for both steady and transient operating conditions to quantify liquid metal replenishment capability, failure threshold, and safety margin under extreme heat fluxes. The analysis further shows that the thermal management envelope of CPS-LMDs is jointly determined by geometry, wettability, pore architecture, and thermophysical properties of the working liquid. In this context, micro-channel based architectures serve as a representative and highly designable implementation route, because they enhance capillary transport and provide an effective structural handle for improving both steady and transient resistance to dry-out.

At the application level, the proposed framework and derived design principles are further extended to the conceptual design of a liquid lithium CPS component for EAST, targeting the mid-plane limiter/test module scenario. The design integrates CPS architecture, lithium circulation, active cooling, thermal insulation, and start-up heat management, and demonstrates the engineering feasibility of translating CPS-LMD theory into a device-oriented solution. Overall, this study provides a unified physical picture of CPS-LMD heat transfer under both steady and transient extreme thermal loads, and builds a coherent bridge from mechanism understanding to component design and future reactor scale application.

Post -characterization of COMPASS-U tokamak plasma-facing components prototypes after cyclic HHF tests in Judith 2

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

COMPASS Upgrade is a new medium-sized tokamak currently under construction in the Czech Republic. Cyclic high heat flux (HHF) tests on large tungsten samples were performed using the electron beam facility JUDITH2 to simulate steady-state plasma loading on the outer divertor target plates during ramp-up and ramp-down conditions, in order to estimate their lifetime and validate the design. Three different conditions were chosen: HHF1=70 MW/m^2 with Eich profile & 0.5 s duration, reaching 2000°C; HHF2=35 MW/m^2 with Eich profile & 3 s duration, reaching 1500°C, and HHF3= 20 MW/m^2 with square profile & 3s duration, reaching 2000°C, all with 10 pulses and time in between to cool down. Three suppliers were selected to characterize tungsten materials for COMPASS-U: ALMT, AT&M, and Plansee. Comprehensive materials characterization was performed to support their downselection.

All three tungsten materials sustained HHF tests. No macroscopic damage was observed on any of the samples examined. To assess microscopic damage, the surfaces of all three tested samples were examined using scanning electron microscopy and optical profilometry. Microscopic examination of the zones subjected to heat fluxes and areas not affected by heat fluxes was carried out. Photographic documentation was prepared, documenting/detailing the characteristic elements of the surface morphology with descriptions of their features. In all tested samples, the strongest changes occurred in the HHF1 zones. For the HHF1 zone of the ALMT sample, the damage was the greatest of all the areas tested. It was characterized by numerous cracks, predominantly linear (not branched), with a direction mainly parallel to the tile's longer edge, an average length of 135 μm, and a width of 20-50 μm. The surface density of cracks locally reached 10 cracks/mm2. Roughness measurements were carried out on each of the three tungsten tiles in the four areas: HHF1, HHF2, HHF3, and the untested surface, allowing a quantitative assessment of roughness changes in the tested zones and determination of their character (3D maps). As part of the study, the height maps and profiles of several cracks present in the HHF1 zone of the ALMT tile were created to determine their depth and width. Additionally, the height maps of the HHF1 zones were generated for the ALMT, Plansee, and AT&M tiles. They clearly show that the HHF1 zones are elevated relative to the rest of the samples' surface.

#14002

Capillary Imbibition and Ultrafast Transport of Liquid Metal in Open Microchannels toward Liquid Metal Divertors

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

To improve the thermal endurance of capillary porous structure (CPS) liquid metal divertors for fusion reactors, this work investigates the spontaneous capillary flow and ultrafast transport of GaInSn in open microchannels fabricated on stainless-steel substrates. Under high heat flux operation, the local liquid metal layer can be continuously consumed, redistributed, or even driven toward partial dryout. In this context, capillary imbibition is directly related to the ability of the surface to sustain liquid coverage and therefore to the accessible heat transfer limit. Focusing on semicircular channels as well as channels with varied aspect ratios and interior angles, the study examines the coupled effects of channel geometry, interfacial wetting, and hydrodynamic resistance, with particular attention to their influence on imbibition length, transient transport velocity, and characteristic imbibition time. Experimentally, arrays of microchannels with well-controlled geometrical parameters are prepared on stainless-steel surfaces, and high speed visualization is employed to track the evolution of the advancing liquid metal front. The objective is to provide experimental evidence and physical guidance for microstructure enabled liquid supply enhancement in CPS based liquid metal divertors.

The results indicate that the dominant transport stage of GaInSn remains broadly capillary driven, with the imbibition length showing an approximately linear dependence on the square root of time. However, the transport capability cannot be described as a monotonic function of hydraulic diameter alone. Instead, it is jointly governed by the channel aspect ratio, interior angle, and effective contact line configuration. A smaller interior angle tends to strengthen corner induced capillary suction and promote precursor wetting, while a moderate increase in the transverse scale helps reduce viscous resistance. When the channel becomes excessively wide or the local meniscus curvature is weakened, the capillary pressure decreases and the net transport enhancement is limited. These competing effects suggest the existence of a favorable geometric window for open liquid metal microchannels, within which capillary driving force and viscous dissipation are better balanced. A scaling interpretation further suggests that the characteristic transport velocity should increase with effective capillary pressure and decrease with viscosity and transport distance, with the geometric influence embodied in a dimensionless function of aspect ratio and interior angle. Overall, these findings highlight the critical role of microchannel geometry in governing rapid liquid metal replenishment and provide useful guidance for structural optimization of CPS-based liquid metal divertors.

Fabrication of divertor target mock-ups with W-WC monoblocks

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Tungsten is considered to be the primary candidate plasma facing material (PFM) for fusion power plants. At the high temperatures reached by the divertor targets, tungsten is subject to recrystallisation which results in a loss of strength. The recrystallisation of tungsten can be suppressed by the incorporation of ceramic particles in the form of oxides or carbides. W2C particle reinforced W (W-WC) materials can prevent detrimental effects of recrystallization while keeping a good heat removal capability. In the context of the Work Package Materials (WPMAT) of the EUROfusion consortium, new W-WC PFM have been developed with a composition of ~ 98 wt % W and 2 wt% W2C (W4WC) in which W2C is a ceramic phase located at W grain boundaries.

In this work, ITER-like small-scale target divertor mock-ups with W-WC monoblocks were fabricated. The fabrication of the final mock-ups required Cu casting to produce a copper interlayer for each monoblock and afterwards an hot radial pressing (HRP) process to join the W-WC-Cu blocks to a CuCrZr pipe. Non-destructive examinations by ultrasonic technique (UT) were carried out to verify the fabrication quality of the W-WC armour and the other plasma facing elements. The UT was performed also after the HRP process to assess if W-WC can substitute pure W of divertor targets.

Characterization of Self-Passivating Tungsten Alloy Produced by Industrial Upscaling and its Comparison to Laboratory Samples

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Tungsten is currently considered to be the most promising material for plasma-facing surfaces in fusion reactors. Self-passivating tungsten alloys have been developed to be oxidation resistant at high temperatures. In a loss-of-coolant accident scenario, radioactive tungsten oxides would be released to the gas phase, if hot tungsten surfaces come in contact with oxygen. Bulk samples of such alloys have been produced on lab scale by mechanical alloying and subsequent hot isostatic pressing (HIP) by CEIT (San Sebastian). As a result of various laboratory investigations, W-10Cr-0.5Y was identified to allow the most effective suppression of the sublimation of W oxide.

In order to investigate the possibilities for large scale alloy manufacturing, EUROfusion supports two lines of industrial upscaling pursued by the Research Centre Jülich (FZJ) and the MPI for Plasma Physics (IPP). IPP cooperates with the industrial partner MBN nanomaterialia in Vascon di Carbonera (Italy). Several batches of powder have been produced by High Energy Ball Milling at MBN and sintered with the Field Assisted Sintering Technology (FAST) process at FZJ. Oxidation experiments were performed on these samples at 1000 °C in a 20% oxygen / 80% argon mixture for 60 hours. Initial attempts failed; however, a stable passivating layer was achieved when the alloy powder was produced by replacing steel milling tools with titanium and optimizing process energy, provided that the chromium content remained above 10%. Although these results do not yet match the extreme oxidation resistance of lab-scale samples, they demonstrate a successful proof-of-concept for industrial manufacturing. As a byproduct of the production technique, a titanium content of 2 – 4% was also found in the alloy. This Ti impurity could be avoided by using different milling tools, such as tungsten-based materials, for the mechanical alloying.

The unintentional Ti content might have a positive influence on the oxidation behaviour however could negatively impact the hydrogen retention. Earlier experiments with pristine hipped laboratory alloy samples showed a higher retention than bulk W, but irradiated (self-damaged) samples were even lower in retention than their bulk W counterpart. This was interpreted as a possibly increased vacancy annihilation in the alloy. Similarly, the industrially produced FAST sintered alloy (containing the additional Ti) showed also a higher D retention determined by nuclear reaction analysis than bulk W under similar loading conditions. Further investigations are ongoing in order to determine the influence of the consolidation behaviour (hipping versus sintering) and the retention of self-damaged samples.

Surface Engineering of Tungsten Capillary-Porous Systems for Liquid Metal Divertor Application

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Liquid metal plasma-facing components are being actively investigated as a potential solution to cope with the extreme heat and particle loads expected in future fusion reactors. Among the proposed concepts, tungsten-based Capillary-Porous Systems (CPS) represent a promising approach for liquid metal divertors, as their intrinsic capillary network can retain and redistribute the liquid metal while maintaining a stable interface with the plasma. However, the surface morphology of tungsten knitted meshes, commonly used as CPS substrates, plays a crucial role in determining the overall performance of the system. In particular, the relatively high surface roughness of these meshes may influence the uniformity of power deposition and the thermal behaviour of the plasma-facing surface.

In this work, different compression techniques are investigated with the aim of reducing the surface roughness of tungsten knitted meshes while preserving their capillary properties. The studied approaches include differentiated thermal expansion, hot pressing, and screw pressing. These methods were applied to folded tungsten mesh samples in order to promote structural compaction and improve surface uniformity.

Following compression, the samples were subjected to liquid tin wetting tests to evaluate their wettability and capillary behaviour, taking advantage of the intrinsic capillarity of the tungsten yarn structure. Microstructural observations and surface analyses were performed using optical microscopy and focus-stacking techniques to reconstruct the surface topography and quantify roughness parameters.

The results indicate that all investigated compression techniques lead to a reduction in surface roughness compared to the as-produced meshes, while maintaining good wetting behaviour at the fibre level. The liquid-metal-impregnated CPS samples produced in this work were subsequently employed by a collaborating group in high heat flux experiments carried out at the HADES electron beam facility. While the detailed results of these tests are beyond the scope of the present contribution, this activity represents an additional step toward the experimental validation of CPS-based liquid metal plasma-facing components

Structured W–Cu composite as an alternative to tungsten for plasma-facing components

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

In fusion devices, plasma-facing components are subjected to extreme conditions, including intense heat and particle fluxes from the hot plasma and, in fusion reactors, neutron irradiation. Tungsten has emerged as a prospective plasma-facing material candidate, mainly thanks to its high melting point, strength at high temperatures, good thermal conductivity, and high sputtering threshold. However, tungsten has certain drawbacks, such as brittleness and poor machinability. Therefore, alternatives like tungsten alloys or tungsten-based composites are being considered.

This work presents the development and performance testing of W–Cu composites in various configurations. The underlying concept is based on the presence of a lower-melting-point phase that undergoes a phase transition when subjected to transient heat loading. The associated latent heat of fusion uses part of the incident energy, which decelerates the temperature increase compared to pure tungsten. In this study, several geometrical concepts and their combinations were investigated, including homogeneous composites, those with tungsten coatings, those with a Cu-depleted surface layer, as well as composites with periodic structures. The durability of selected composites and the temperature evolution in comparison with pure tungsten were demonstrated in high heat flux tests.

Comparative Overview of Divertor Cassette Design and Integration in EU-DEMO, VNS and DTT

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

The divertor design represents one of the most critical challenges in developing fusion reactors, as it must withstand multiple intense loads and relies heavily on physics-based modelling. Its primary function is to exhaust a substantial fraction of the plasma heating power - from fusion alpha particles and auxiliary plasma heating - that is conducted through the plasma scrape-off layer during normal, transient, and off-normal operating conditions. While its plasma-facing components must be compatible with plasma–material interaction phenomena, the divertor cassette must protect the vacuum vessel from neutrons.

The very high heat fluxes together with the high neutron irradiation require advanced engineering solutions, extensive technology development and qualification, and the careful integration of multiple subcomponents under stringent thermal, structural, and operational constraints.

This contribution provides a comparative overview of divertor design strategies adopted for three major European tokamak systems currently under development within the EUROfusion programme: EU-DEMO, the Volumetric Neutron Source (VNS), and the Divertor Tokamak Test facility (DTT). While the EU-DEMO divertor design is dominated by reactor-scale dimensions, the VNS and DTT divertors are comparable in size but differ significantly in magnetic configuration flexibility and neutron irradiation environment.

Starting from the key design drivers and boundary conditions of each machine, the analysis examines both shared features and distinctive challenges across the three concepts. Particular emphasis is placed on geometric and system integration aspects, heat flux handling strategies, coolant operating regimes, structural and electromagnetic (EM) loads, and compatibility with remote handling (RH) and maintainability requirements.

The comparative approach highlights design synergies among the three devices and explains how specific engineering solutions are shaped by each machine's mission, operational targets, and physics context. The discussion ultimately contributes to identifying cross-cutting technologies and integrated design principles that can support the European strategy toward reactor-relevant power exhaust solutions.

Enhanced Monte Carlo simulation of the W activation in the WEST tokamak due to runaway electrons impact of Plasma Facing Components

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Disruption events in tokamaks are magnetohydrodynamic instabilities that can occur with the consequent generation of runaway electrons. These highly energetic electrons can strike tungsten-made components of the machine causing thermomechanical damages and radiological concerns. Recently, spectroscopy analyses on the divertor have led to the assessment of radioactivity levels in tungsten components around 42 Bq/g for 181W, above the regulatory limits fixed by French Law for the release of material in the public domain.

To better understand the measurements performed at WEST on plasma facing components (PFC), calculations based on Monte Carlo method simulations have been conducted with TRIPOLI-4® and GEANT-4. A first of a kind calculation has been performed, based on a two-step scheme. The first step consists of simulating the electromagnetic shower (EM) in the PFC and the transport of high energy X-rays produced by bremsstrahlung from electrons in tungsten. The photonuclear reaction rates 182W(γ, n)181W are used to evaluate the subsequent gamma and X-rays decay photon sources. The second calculation is dedicated to the transport of decay photons in order to assess the HPGe detection response used for gamma spectroscopy.

A precise model of the PFC without (for EM simulations) and with the HPGE detector (for spectroscopy simulations) was setup to accurately simulate the impact of the runaway electron beam on tungsten and the subsequent measurements. Monte Carlo simulations without modelling the magnetic field provided information on the physical behavior of the electron beam and X-rays, depending on the incident energy and incident angle of the electrons. In addition, the simulated response of the HPGe detector has been calculated for each of the incident electron beam parameter variation. The spectroscopic measurements came from both the whole surfaces of the PFCs and from specific slices assessing the distribution of activation in depth. Trying to describe the physics of the phenomenon and fit the measurements, a matrix problem was implemented. The direct solve of the system provided inconsistent solutions. Therefore, a minimization procedure was preferred and the latter helped to fit the experimental results. The minimization process led to almost acceptable physical configurations of the angular and energetic distribution of the electron source, which could be improved while considering a magnetic field in the EM simulation. The calculation scheme will be used to evaluate the activation and the subsequent dose rate of the First Wall and Divertor at ITER.

Finally, GEANT-4 simulations enabled cross-validation with TRIPOLI-4®, showing consistent results.

Thermal-Hydraulic Assessment of HHF Cooling Channel Geometries for the W7-X Divertor Target Module

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

W7-X prepares the transition to full metal plasma facing components in order to explore the behaviour of stellarator plasmas under conditions as close as possible to a future HELIAS reactor. Development of the highly loaded divertor target elements armoured with pure tungsten or tungsten heavy alloy requires the design and qualification of technologies for a future industrial production of the next generation of W7-X target elements. Part of the conducted activities related to the development of the suitable high heat flux (HHF) target modules, has also relied on the use of Computational Fluid Dynamics (CFD). It has been shown that the two-phase flow (gaseous and liquid) in the cooling channel cannot yet be reliably simulated as the existing models are based on pool boiling with buoyancy driven bubble transport. However, it was also shown that the mass flow rate in W7-X is sufficient to ensure that the flow in cooling channel of the target element remains in the single-phase regime. Under these conditions, CFD provides a reliable basis for design optimisation and comparative analysis of the proposed designs.

This study presents a thermal-hydraulic assessment of different HHF cooling channel geometries. The analysis is based on steady-state CFD simulations using a three-dimensional model of a single cooling channel. In total, ten channel geometries were investigated. A comparison between simulation results obtained with ANSYS Fluent and STAR-CCM+ is also provided. The designs are evaluated in terms of peak top plate temperature (Tmax), average heat transfer coefficient (HTC), and hydraulic pressure drop (dp/dz). The results demonstrate that cooling performance correlates strongly with the wetted area (fluid–solid interface area) and with the distance between the top far-edge and the wetted surface. The investigated geometries can be grouped according to their hydraulic performance. Channels with variable cross-sections yield the highest pressure drops but also provide the most effective cooling. These findings highlight the trade-off between thermal performance and hydraulic losses, both of which are often limiting factors in design.

Source-Dependent Kinetic Separation of H in Tungsten via Tunable Doppler Spectroscopy

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Retention of Hydrogen (H) isotopes in plasma-facing materials is a critical issue for nuclear safety and self-sufficiency in fusion devices operating with Deuterium-Tritium (DT) fuel mixture and tungsten (W) plasma-facing components. Its reliable detection, either in-situ or ex-situ, is of great importance for nuclear licensing and material characterisation after plasma exposure. However, we observed that in vacuum conditions, the Balmer-alpha line emission of Hydrogen (Hα) measured by Laser-Induced Breakdown Spectroscopy (LIBS) exhibit complex Doppler shifts in the laser-induced plasma (LIP), leading to significant spectral distortions. In this work, we report the first systematic study of Doppler shifts in Balmer-alpha line emission of hydrogen obtained by picosecond LIBS on W samples. Our results reveal not only Doppler shifts associated with single-velocity H atoms, but also, between delay time from 0 to 300 ns, distinct dual-wavelength shifts of the emission lines arising from the simultaneous presence of fast and slow H atoms in the LIP. The fast component exhibits a laser fluence-dependent shift, whereas the slow component shows the opposite behaviour. Fast and slow H atoms can be temporally separated during detection due to their different velocities. Moreover, we identified, that in the near surface layer, fast hydrogen corresponds to surface-adsorbed H, while slow H corresponds to H present in the W bulk material. Building on these findings, we demonstrate a new approach for tuning the Balmer-alpha wavelength as function of laser fluence, making it possible to temporally isolate fast and slow H atoms and thereby suppress spectral interference. The Doppler characteristics additionally offer a potential spectroscopic tool using LIBS for distinguishing between H present at the surface and subsurface of the W bulk material. Utilizing this method, we successfully observed the dynamic adsorption kinetics of H on the W surface at a pressure level as low as 10-7 mbar. This work not only resolves a fundamental spectral distortion mechanism, but also provides guidelines for improving the accuracy of depth-resolved H isotopes detection in fusion-relevant plasma-facing materials (PFMs) utilising LIBS measurements.

Experimental Assessment of Sweeping Effects on Divertor Targets with Tungsten Monoblock Design

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

Managing extreme heat loads on divertor targets remains one of the key technological challenges for plasma-facing components in future fusion devices. The strike-point sweeping is considered a potential mitigation technique to reduce peak heat flux by distributing the deposited power over a larger surface area and over time. Understanding how sweeping parameters influence thermal response and surface degradation is therefore essential for the qualification of divertor components.

Within the framework of the qualification activities for the Divertor Tokamak Test (DTT) facility divertor, an experimental campaign was carried out at the HADES high heat flux testing facility to investigate the effect of beam sweeping frequency on the thermo-mechanical response and surface damage of tungsten monoblock divertor mock-ups. Two mock-ups, consisting of CuCrZr tubes (15 mm outer diameter, 12 mm inner diameter) equipped with 18 tungsten monoblocks each, were exposed to identical heat flux conditions while applying different sweeping frequencies in order to assess the impact of the temporal distribution of the thermal load. The electron beam was swept over a length of approximately 150 mm across the surface of the mock-ups. The first mock-up was tested with a sweeping frequency of 5 Hz, while the second was tested at 2.5 Hz. Each specimen was subjected to 1000 thermal cycles consisting of 20 s beam-on followed by 20 s beam-off, with a heat flux of 80 MW/m^2 applied over a footprint of 16 mm (two monoblocks in the DTT divertor target design). Prior to testing, the surface condition of the central monoblock and the two adjacent lateral monoblocks was characterized by optical microscopy to establish a reference for post-exposure comparison. During the experiments, three thermocouples positioned 3 mm below the loaded surface provided temperature measurements that will be used to calibrate thermal simulations aimed at reconstructing the surface temperature evolution. Infrared thermography was also employed to monitor the spatial and temporal thermal response of the exposed surfaces.

The study aims to assess the influence of sweeping frequency on the development of surface damage mechanisms, including cracking, plastic deformation, and local surface modification. Post-test microscopy analyses were performed to compare the degradation patterns between central and lateral monoblocks and to correlate the observed damage with the reconstructed thermal histories as a function of the applied sweeping frequency. These results provide experimental insight into the role of sweeping parameters in mitigating localized heat loads and their implications for the lifetime of tungsten divertor components.

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Plasma-wall interactions activities in support to ITER operation with full-W first wall

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Topics

G. Plasma-Facing Components

Communication type

Oral

Abstract

A new ITER baseline research plan has recently been developed, covering essentially a twostage approach: (i) inertial cooled tungsten (W) components in the main chamber during the Start of Research Operation phase in predominantly hydrogen plasmas, and (ii) deuteriumtritium operation aiming at early Q=10 demonstration with W components actively cooled. Beryllium (Be) as armor material has been replaced by W from day one, bringing ITER from the point of view of plasma-wall interactions and exhaust (PWIE) as well as integrated coreedge plasma operation closer to a fusion reactor. The usage of W shall enhance the powerhandling capabilities during critical transients like vertical displacement events and reduce the steady-state wall erosion and long-term fuel retention during long-pulse operation in DT at high Q. Lessons learned from JET operation with Be/W-mixture, from e.g. ASDEX Upgrade operation with W walls, and enhanced modelling capabilities for PWI and impurity transport during steady-state operation and modelling of transient events paved the way towards the fullW ITER.

However, boronization, a glow discharge with diborane inducing a thin boron (B) layer, will be applied to getter oxygen, reduce the W first wall source and influx, and ensure a wide operational window as well as permitting fast progress in scenario development. As consequence, fuel retention by co-deposition in B layers occurs and needs to be measured, controlled, and removed by wall-conditioning techniques to minimize the fuel inventory and stay within licensing limits. The retention in B layers is higher than in Be, but the amount of B is limited to the number of boronizations applied.

In close collaboration with the ITER Organization and ITPEA, the EUROfusion program was refocused to address critical issues for operation and safety in the areas of PWIE with W armor, i.e. W sources and influxes, and B, i.e. boronization lifetime and fuel retention in B-layers. An overview of the supporting experimental and numerical studies within the program on PWIE physics and beyond will be presented. This covers boron-related physics including MD simulations of chemical erosion, validation erosion and retention experiments in linear plasmas, experiments in tokamaks with material samples, ex-situ and in-operando measurement of retention in B layers, fuel-removal experiments, predictive modelling of B migration, etc. This is complemented by W-related studies with a focus on W source and influx estimates, W re-deposition, W layer stabilities, and the support in W armor qualification for the temporary first wall under steady-state and transient conditions.

Structural analysis and verification of DEMO Divertor Dome and Reflector Plates according to the 2025 Low Aspect Ratio long leg option design

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

In 2025, a new design of the DEMO Divertor was developed: the so-called “Low Aspect Ratio long leg” baseline has been adopted to study the whole set of fusion parameters. This work deals with the structural analysis executed for the DEMO Divertor Shielding Components that are the Dome, the Inboard Reflector Plate and the Outboard Reflector Plate. The applied loads have been obtained with dedicated analyses: the pressure loads, the electromagnetic transient loads (including disruptions and Halo currents), ferromagnetic loas, and the thermal loads (due to neutron flux) have been considered. An elastic analysis was performed and, to apply the RCC-MRx rules, a general overview analysis was carried out beforehand to choose the proper Stress Classification Line (SCL) positions. Moreover, a parametric element size study was conducted to select a suitable mesh. Both Type P and Type S damage assessments were performed.

Design and qualification of water-cooled W-based divertor for W7-X

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Topics

G. Plasma-Facing Components

Communication type

Poster

Abstract

Wendelstein 7-X, the world's largest superconducting stellarator (IPP Greifswald), began long-pulse plasma experiments in 2022 using a water-cooled, CFC-based divertor. To ensure the low tritium retention required for future operation, plasma performance must be demonstrated with carbon-free materials. Consequently, a project was launched to develop a Tungsten (W) based divertor. This involves two parallel tasks: optimizing the plasma-facing geometry to prevent overloads and improve exhaust/impurity retention, and developing the structural design and manufacturing technology within the EUROfusion WPDIV framework.

This paper presents the current status of the structural design and technology qualification. The design utilizes the existing cooling infrastructure and matches the current CFC modules in shape and weight, while prioritizing simplified manufacturing and maximized heat load capacity. The primary high-heat-flux technology under qualification is a laser powder bed fusion (LPBF) CuCrZr heat sink with a W-based coating at the plasma exposed side. Backup solutions involve an armor of W or WNiFe tiles with soft Cu interlayer bonded to the heat sink via Au/Cu brazing or nano-Cu sintering. Results from simulations, hydraulic and small-scale high-heat-flux tests are presented, and the remaining challenges to achieving a full-scale qualified solution are discussed.

Keywords: Wendelstein 7-X, stellarator, divertor, additive manufacturing, cold gas spray, W, WNiFe

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