First International Workshop on Materials and Mechanics for Fusion Energy

29–30 August 2024

Institute of Physics, London, UK

iop.org

IOP Institute of Physics

First International Workshop on Materials and Mechanics for Fusion Energy

Programme

Thursday 29 August

09:00	Registration and Refreshments
09:15	Welcome
	Session I Chair: Guangnan Luo
09:20	PLENARY: A Phased Roadmap for the Development of Fusion Reactor Materials in China **Yugang Wang,** Peking University
10:00	Micro-Breeder Blanket: New experimental capability to investigate performance of breeder blanket system, including tritium breeding, of a fusion reactor using engineered fusion-relevant neutron spectrum **Joven Lim,** UKAEA/CCFE
10:30	Recrystallization and grain orientation dependence of surface morphology in helium-implanted tungsten after high-temperature annealing **Long Cheng,** School of Physics, Beihang University
11:00	Coffee Break
	Session I Chair: Chris Hardie
11:20	Continuum Dislocation Mechanics: bridging the scales **Anish Roy,** University of Loughborough
11:50	Multiscale Modeling for Irradiation Hardening of Tungsten **Yao Shen,** Shanghai Jiao Tong University
12:20	Micro-nano scale damage on tungsten/tungsten alloy surface under low-energy high flux hydrogen plasma exposure **Wanqi Chen,** China Nuclear Power Engineering Co., Ltd
12:40	Lunch
	Session III Chair: B Long
13:40	Study of radiation hardening and embrittlement of structural alloys by utilizing high-energy ions **Chonghong Zhang,** Institute Of Modern Physics, CAS

14:10	Impact of structural materials' in-service degradation scenario on fusion first-wall/blanket design **Arunodaya Bhattacharya,** University of Birmingham
14:40	R&D of advanced FM structural steels for fusion energy at IMR **Wei Yan,** Institute of Metal Research
15:00	A novel ODS-RAFM steels with excellent high-temperature mechanical properties by SLM **Kailun Li,** Institute of Engineering Thermophysics, Chinese Academy of Sciences
15:20	Coffee Break
	Session IV Chair: Anish Roy
15:40	KEYNOTE: Reviews on design strategy and scalable production routs of tungsten-based materials for nuclear fusion application **Yucheng Wu,** Hefei University of Technology
16:10	Project LiFTOFF - Lithium Facility by Oxford Sigma for Fusion **Dr Thomas Davis,** Oxford Sigma
16:40	Amorphous and anisotropic fatigue damage initiation on tungsten surfaces under repeated high-flux hydrogen plasma loads **Yu Li,** Institute of Plasma Physics, Chinese Academy of Sciences
17:00	One-step preparation and performance evaluation of SPTAs/CLAM joint via powder metallurgy route **Xiaoyue Tan,** Hefei University of Technology
17:20	Reception
18:30	End of Day 1

Friday 30 August

	Session V Chair: Yugang Wang
09:00	KEYNOTE: Recent Progress of PFMC R&D and Testing Facilities toward Fusion Reactor in ASIPP **Guang-Nan Luo,** Institute of Plasma Physics, Chinese Academy of Sciences
09:30	Nuclear materials research at the University of Birmingham **Yu-Lung Chiu,** University of Birmingham
10:00	Progress on Standardization of Mechanical Testing of Small-scale Samples for Fusion Reactor Materials **Bin Long**
10:30	Kyoto Fusioneering: Overview of Fusion Materials R&D Acitivities **Mario Oliver,** Kyoto Fusioneering
10:50	Coffee Break
	Session VI Chair: Thomas Davis
11:10	KEYNOTE: Prediction of material response in unchartered fusion environments **Chris Hardie**, Uk Atomic Energy Authority
11:40	High performance W-ZrC alloy and W nanoparticle reinforced Cu for plasma-facing components **Rui Liu,** Institute of Solid State Physics, HFIPS, Chinese Academy of Sciences
12:00	Modelling high-energy gamma-ray spectrum from deuteron-triton collisions **Natalia Timofeyuk**, University of Surrey
12:20	Equivalent Methods to Predict Cavity Swelling in Structural Materials Induced by Neutron Irradiation **Chenxu Wang**, Peking University
12:40	Lunch
13:40	Roundtable Industry Discussion
14:40	Tea Break
15:00	End and Depart

Session I

PLENARY: A Phased Roadmap for the Development of Fusion Reactor Materials in China

Yugang Wang[1]
[1]Peking University, China

This talk will present the roadmap for the development of fusion reactor materials at different stages of fusion energy development in China. Firstly, the service environments and requirements of fusion reactor materials will be introduced and then we will demonstrate our consideration on the roadmap for fusion reactor materials. Finally, some recent R & D progress of fusion reactor materials in China will be introduced.

The general strategy of FRMs roadmap in China is that the development of FRMs in China should match with the requirements of Chinese Fusion Roadmap at different stages and in different service environments, and the roadmap of FRMs needs to have a time schedule.

There are three major missions in the roadmap:

1. Developing of new fabrication techniques for ODS-steel, Copper Alloy and etc.; improving of material properties of Chinese RAFM-steel, pure tungsten and etc.
2. Using CRAFT/BEST to test PFMs and some functional materials; using fission reactors, spallation source to obtain irradiation data of FRMs and triple-beam ion-irradiation to benchmark multiscale simulation; standardizing of small specimen test technology.
3. Constructing of facilities to test irradiation effects of fusion neutrons.

Micro-Breeder Blanket: New experimental capability to investigate performance of breeder blanket system, including tritium breeding, of a fusion reactor using engineered fusion-relevant neutron spectrum.

Joven Lim[1], Louis Butt[2], David Foster[1], Lee Packer[1], Tony Turner[1], Chris Jones[1], Jonathan Pearl[1], Monica Jong[1], Sam Cullen[1], Duc Nguyen[1], Ben Phoenix[2], Yu-lung Chiu[2], Martin Freer[2], and Mark Gilbert[1]
[1]UK Atomic Energy Authority, UK, [2]University of Birmingham, UK

The micro-breeder blanket (μBB) serves as a distinctive experimental tool, specifically designed to assess the performance of breeder-facing materials (BFMs) within various breeder blanket systems' environments. It achieves this by utilizing an engineered fusion-relevant neutron spectrum. Neutrons in the μBB are generated through interactions of energetic protons with materials such as lithium and beryllium. Leveraging neutronics inputs, the μBB can generate tailored neutron spectra relevant to next-generation nuclear reactors, particularly fusion reactors, facilitating testing of reactor components.

The significance of the engineered neutron spectrum in fusion materials testing lies in its ability to match the spectrum of Primary Knock-on Atoms (PKA) and transmutation rates, particularly for helium and other gases, in BFMs. These elements impact the stability and mobility of Frenkel pairs and point defects induced by energetic radiation in BFMs. Additionally, employing accelerator technology to generate fast neutrons enables scaling up the integrated neutron flux, aligning it with the expected flux experienced by components within the bio-shield of a fusion reactor.

This presentation highlights the validated neutron spectra achievable through the μBB and their relevance in comparison to the neutron spectra of DEMO and fission reactors for materials testing purposes. Furthermore, it presents into early findings from the Vanadium-Lithium breeder blanket system, particularly focussing on the rate of Fusion fuel production.

Recrystallization and grain orientation dependence of surface morphology in helium-implanted tungsten after high-temperature annealing

Jiaguan Peng[1], Tiangang Zhang[1], **Long Cheng**[1], Yue Yuan[1], and Guang-Hong Lu[1]
[1]School of Physics, Beihang University, China

To investigate recrystallization and the effect of grain orientation on surface morphology changes in tungsten after helium ion implantation and thermal treatment, rolled tungsten samples were implanted by 40 keV helium ion at ~673 K, reaching doses ranging from 5E21 to 1E22 m-2 at a high flux of 1E20 m-2s-1. The samples were subjected to annealing cycles ranging from 1573 K to 1873 K, lasting 1 hour per cycle. Quasi in-situ analysis was performed between each cycle at identical locations on the samples to study the morphology and microstructure evolution due to thermal annealing.

Retarded recrystallization due to helium was confirmed in this work, and the minimum Vickers hardness was observed after annealing at 1573 K with partial recrystallization. However, an unexpected hardness recovery was observed after annealing at temperatures higher than 1673 K. After annealing at 1673 K, ridge-like microstructure measuring about 100 μm in height were observed on {111} grains with increased hardness and geometrically necessary dislocation density compared to measurement at 1573 K, while numerous blisters and holes appeared on {110} grains. Microstructure analysis of the sample annealing at 1673 K highlighted an orientation-dependence of morphology evolution linked to helium bubble distribution. In {100} grains, helium bubble distribution was in line with SRIM simulation. Abnormal large helium bubbles with sizes up to 200 nm were observed in {110} grains, associated with blisters and holes on the surface. In {111} grains, a deeper bubble depth distribution extending beyond 400 nm reaching the adjacent grain boundary. Ion channeling effect and orientation dependent helium bubble growth mechanism were employed to elucidate these findings.

According to this work, a novel way to reinforce tungsten is proposed by using high-flux and short-time helium ion implantation, which results in an elevated recrystallization temperature and hardness recovery ability at high temperatures.

Session II

Continuum Dislocation Mechanics: bridging the scales.

Anish Roy[1]
[1]Loughborough University, UK

Progress on our species relies on us being able to design and engineer material physical systems to tackle a host of practical and applied problems in science and engineering. Predicting how these systems respond to external stimuli is crucial for reliability. Interestingly, most materials are known to exhibit different behaviours at different scales, both in terms of length and time. We have come to use terms like “fine-scale” and “coarse-scale” to refer to the physics of materials at two distinct ends of this “scale”. For many practical problems, it is impossible to resolve the full details of the fine-scale problem due to the overwhelming computational costs. Therefore, we must seek alternative approaches that are computationally efficient and affordable. Herein, lies the challenge of bridging the gap across both time and length scales. This bridging scale is known as the “meso-scale”. Designing, developing and implementing computationally efficient yet physically accurate representations of the underlying fine scale at the meso-scale is a practical way to achieve a multiscale modelling approach. These are particularly relevant in engineering applications where component sizes can neither be classified as “coarse” nor “fine”. In this talk I will explore why a continuum approach is crucial to bridge the scales while accounting for crystalline defects that drive plasticity (and deformation) in metals and alloys.

Multiscale Modeling for Irradiation Hardening of Tungsten

Yao Shen[1], Guisen Liu[1], Ping Yu[1], KaiTao Wu[1], and Jiaqing Shi[1]
[1]Shanghai Jiao Tong University, China

Tungsten has been chosen as the plasma-facing material for fusion devices such as ITER and DEMO. To assess the safety of tungsten under the harsh service conditions, this study develops a multiscale modeling approach to predict the irradiation hardening of tungsten. The hardening is caused by the impediment to dislocation gliding by irradiation-induced defects such as voids, dislocation loops, and precipitates. First, molecular dynamics (MD) simulations are performed to reveal the mechanisms for the interactions between dislocations and irradiation defects at the atomic scale, analyzing the morphological and energy changes that occur as dislocations glide through defects driven by external loading. The information obtained from the MD simulations is then used to calibrate the continuum scale dislocation-defect interaction models, dislocation dynamics (DD), which consists of a long-range elastic interaction model and a short-range contact scheme. The elastic interaction force is evaluated efficiently using the concept of Eshelby's equivalent inclusion. The contacting details such as dislocation gliding along void surface and the critical stress for the dislocations to unpin from the defects are realized by introducing an additional surface drag force that is calibrated against MD results. These treatments enable efficient and accurate DD simulations of dislocation interactions with high-density randomly distributed voids and loops. Based on the DD results, the study proposes laws for voids hardening, dislocation loops hardening, and their synergistic hardening. These hardening laws are then incorporated into a crystal plasticity finite element model (CPFEM) to simulate the deformation behavior of irradiated tungsten at the meso/macroscale. The results of the CPFEM are validated against fission irradiation experiments from the literature, and it is found that the CPFEM accurately predicted the hardness increments at different neutron irradiation doses. These multiscale developments are believed to provide a solid foundation for predicting the irradiation hardening of tungsten in fusion service.

Micro-nano scale damage on tungsten/tungsten alloy surface under low-energy high flux hydrogen plasma exposure

Wanqi Chen[1,2], Wei Liu[2], Kailun Li[3], Mingshen Li[2], and Menghan Ma[2]
[1]China Nuclear Power Engineering Co., Ltd, China, [2]Tsinghua University, China,
[3]Institute of Engineering Thermophysics, Chinese Academy of Sciences, China

Irradiation-induced damage will occur on the tungsten after plasma exposure, which causes performance degradation and negative effects on device safety. In this study, nucleation and growth mechanism of surface blisters and corresponding hardening mechanism of H plasma exposed tungsten are investigated. Blistering and retention behavior of 3D printed tungsten alloys after H/D plasma exposure are also investigated. This study will contribute to fundamentally suppress the blistering behavior and could

provide a new solution for the tungsten alloy design and performance optimization, which is important to improve security and service lifetime of nuclear fusion device.

After low-energy high flux H plasma exposure, the intra-granular blisters in tungsten are found to form on the {100} planes, with <100> edge dislocation existing nearby. Besides, a few chains of dislocation loops in the vicinity of H blisters are observed, including prismatic loops and shear loops aligning along <111> directions with Burgers vectors b=±1/2<111>. Therefore, a {100} plane nucleation mechanism and a multi-stage growth mechanism are creatively proposed in this study. Moreover, the hardening behaviour induced by low-energy high flux H plasma exposure is found to be related with blistering behaviour. The quantitative relationship between damage defects and hardness is established and the intrinsic mechanism of hardening behaviour is revealed.

In terms of 3D printing of tungsten, approaches on optimizing alloy composition and scanning strategy were carried out to solve the cracking problem. It is also found that the blistering is inhibited in the 3D printed tungsten and W-ZrC alloy after H plasma irradiation. And the retention of 3D printed tungsten alloys is lower than that of rolled tungsten. The results highlight the capability of tailoring the microstructures of W alloys by 3D printing to achieve improved plasma-compatible performance.

Session III

Study of radiation hardening and embrittlement of structural alloys by utilizing high-energy ions

Chonghong Zhang[1], Zhaonan Ding[1], Xianlong Zhang[1], Yuguang Chen[1], Xuxiao Han[1], Jianyang Li[1], Yitao Yang[1], and Yin Song[1]
[1]Institute of Modern Physics, Chinese Academy of Sciences, China

Hardening and embrittlement of in-pile key structure components under high-fluence fast neutron irradiation is an important issue for the service and safety of nuclear reactors. In recent years, we engaged in developing a high-energy ion-accelerator-based method to efficiently evaluate the radiation resistance of materials candidate to nuclear reactors. Heavy ions like Feq+ with a kinetic energy of several MeV per nucleon were used to produce atomic displacements in specimens of materials under well-controlled conditions (ion flux, ion fluence, longitudinal damage homogeneity, substrate temperature) at the irradiation terminal of HIRFL (Heavy-ion Research Facility in Lanzhou), to simulate the radiation damage of materials in nuclear reactors. Several methods (nanoindentation, Vickers hardness test, specialized small-punch test) were used to investigate the post-irradiation mechanical properties. Simulations based on kinetic Monte Carlo as well as analysis by finite element method (FEM) were carried out to study the mechanisms underlying the radiation responses. By these methods, we studied the radiation hardening and embrittlement of several structural materials including the reactor pressure vessel (RPV) steel A508-3 and its model alloys, the reduced-activation ferritic/martensitic steel CLF-1 candidate to the fusion reactor blankets, the FeCrAl-base oxide-dispersion-strengthened (ODS) steel and vanadium alloys. This paper gives an introduction of our recent study on some interesting topics of radiation damage of these materials.

Impact of structural materials' in-service degradation scenario on fusion first-wall/blanket design

Arunodaya Bhattacharya[1]
[1]University of Birmingham, UK

Future fusion energy system's in-vessel operating environment presents one of the harshest conditions structural materials will face. This includes unprecedented neutron bombardment with simultaneous presence of elevated operating temperatures, corrosive coolants/breeders, high thermo-mechanical loading, and harmful solid/gaseous transmutation product generation. Moreover, plasma-facing armour and divertor exhaust will experience the added complexity of high heat flux exposure with simultaneous impingement of low-energy plasma-particle flux. Therefore, success of future fusion concepts inherently depends upon the availability of radiation-tolerant materials and a thorough understanding of their in-service degradation phenomenon. This talk will summarize how the safe operating conditions of fusion first-wall/blanket materials are determined, discusses the current state-of-the-art with first-wall/blanket materials and presents the latest progress in our overarching understanding of neutron irradiation effects in structural materials with latest results from HFIR reactor irradiated RAFM/ODS steels.

R&D of advanced FM structural steels for fusion energy at IMR

Wei Yan[1], Yan Fen Li[1], and Yi Yin Shan[1]
[1]Institute of Metal Research, China

Advanced Ferritic/Martensitic (FM) heat-resistant steels are promising candidate structural materials for fusion energy system. The R&D of 9–12%Cr reduced activation FM heat-resistant steels as well as the corresponding oxide dispersion-strengthened (ODS) steels such as 9Cr-ODS steel, 12Cr-ODS steel and the novel MX-ODS steel that are developed at IMR, has been reviewed from the following aspects: (1) alloying design strategy; (2) manufacturing; (3) microstructure and properties; and (4) the compatibility with the high-temperature high-pressure water vapor.

A novel ODS-RAFM steels with excellent high-temperature mechanical properties by SLM

Kailun Li[1], Wei Liu[2], Shubo Zhang[2], Wenjing Zhang[2], and Hao Chen[2]
[1]Institute of Engineering Thermophysics, Chinese Academy of Sciences, China,
[2]Tsinghua University, China

Improving the high temperature mechanical properties of reduced activation ferrite/martensite (RAFM) steels is a topical issue in cladding materials for nuclear fusion reactors. Oxide dispersion strengthened reduced activation ferrite/martensite (ODS- RAFM) steels prepared by powder metallurgy are complicated to process and difficult to apply to large parts with complex structures due to poor reprocessing properties. The creep and irradiation resistance of CNAs with nano MX (M=metal; X=C,N) reinforcement is lower than that of ODS-RAFM steels. The size and distribution of oxides are difficult to control effectively for ODS-RAFM steels prepared by casting, whose homogeneity of micro structure and mechanical properties are not desirable. We propose a new idea for the preparation of ODS-RAFM steels based on additive manufacturing technology. In view of the high cooling rate and limited melting pool of additive manufacturing technology, the oxygen atmosphere during the additive manufacturing process is exploited to cause in-situ oxidation of the oxygen-active elements (Ti, Y etc.) in the matrix by controlling the thermodynamics and kinetics of oxide formation, resulting in the formation of dispersed in-situ oxides (5-20 nm), thus greatly improving the high-temperature mechanical properties of the additively manufactured RAFM steels, with the tensile strength at 600°C increasing from 499 MPa to 886 MPa and time to rupture at 650°C 120MPa from 20h to 800h. This work provides a new approach for the integrated preparation of large and complex components of ODS-RAFM steels, which has excellent high-temperature mechanical properties similar to those by powder metallurgy.

Session IV

KEYNOTE: Reviews on design strategy and scalable production routs of tungsten-based materials for nuclear fusion application

Yucheng Wu[1,2], and Xiaoyue Tan[1,2]
[1]School of Materials Science and Engineering, Hefei University of Technology, Hefei, China, [2]National-Local Joint Engineering Research Centre of Nonferrous Metals and Processing Technology, China

Development of high-performance material and its engineering process route is critical for the operational of the future fusion reactors. Tungsten (W) is the main candidate of plasma facing materials as it is resistant to erosion, has high melting point and exhibits peaceable transmutation under neutron irradiation. However, W is intrinsically brittle and faces further embrittlement in service condition. To overcome these issues,

internal and external toughening strategies including alloying, second-phase doping and plastic processing are being combined to develop W-based material. Because it is refractory metal material, powder metallurgy and subsequent deformation processing become the efficient production route to regulate the high-performance W-based material.

In this contribution, the design strategy and research findings of W-based material will be overviewed. The alloying elements of Rhenium and Vanadium aim to improve the mobility of dislocation and inhibit grain growth, and the second-phase particle of yttria targets additionally to achieve dispersion strengthening. Deformation processing of rolling or swaging aims to future increase the density of the W-based powder materials and increase the amount of the movable dislocation. Following these strategies, the W-based materials achieve high strength and good elongation. Notably, the ductile-brittle transition temperature (DBTT) approaches room temperature.

In summary, the process maturity is now high enough for industrial upscaling. Here describes scalable production line of wet-chemical routes including sol-gel method, spay drying and continuous hydrogen reduction to produce to W-based powders, and the powder forming technology of cold isostatic pressing combines normal sintering and deformation processing techniques to consolidate and modify the W-based bulks. The pilot production line has been built, reaching the 100 kg level. Based on the advanced W-based bulk, flat-type mono-block has been produced using a joint technique of vacuum casting and brazing. Here will show the latest test results about thermal loading and service performance at EAST.

Project LiFTOFF - Lithium Facility by Oxford Sigma for Fusion

Thomas Davis[1]
[1]Oxford Sigma, UK, [2]Bangor University, UK

Project Lithium Testing Facility by Oxford Sigma for Fusion (LiFTOFF) seeks to evaluate the effect of liquid lithium exposure on both corrosion resistance and mechanical performance of different materials that will be used in nuclear fusion energy fuel cycle system. Nuclear fusion requires two hydrogen isotopes as fuel: deuterium and tritium. Deuterium can be easily extracted from seawater (about 33g per tonne) while tritium is rare and difficult to source due to its short half-life (12.3 years). Therefore, tritium is proposed to be bred inside the fusion reactor in a component called breeder blanket. To produce tritium, liquid lithium must be exposed to neutrons coming from the fusion reactions and must be contained in the breeder blanket for further separation from the remaining lithium. How the liquid lithium interacts and degrades the mechanical performance of the breeder blanket components will largely determine the success of commercial fusion powerplants.

This presentation will provide an overview of the company, project, scientific mission, technical mission, construction, and operation. In particular, the presentation will focus on how the scientific data collected to date has uncovered better insight to the liquid metal corrosion mechanisms on advanced nano-oxide strengthened materials.

Amorphous and anisotropic fatigue damage initiation on tungsten surfaces under repeated high-flux hydrogen plasma loads

Yu Li[1]
[1]Institute of Plasma Physics, Chinese Academy of Sciences, China

Facing extreme plasma loads, the structural integrity of the tungsten divertor is crucial for ITER, an engineering marvel in nuclear fusion reactors, to achieve its fusion performance targets. Induced by repeated transient heating from plasmas, the thermal fatigue damage of tungsten–typically accompanied by the formation of surface relief–has been identified as a critical issue but an in-depth understanding is still lacking. Here, we report the formation of amorphous and anisotropic surface relief on ITER-grade tungsten surfaces under ITER-relevant hydrogen plasma loads. Measured by both electron backscatter diffraction over large fields of view and transmission Kikuchi diffraction of site-specific lamellae, such surface relief preferentially forms on grains with {110} planes parallel to the surface. This cannot be explained by the orientation-dependent resolved shear stress according to the Schmid law, threshold displacement energy anisotropy, or oxidation anisotropy. Furthermore, surface relief amorphization is revealed by high-resolution transmission electron microscopy imaging and selected area electron diffraction analysis, and is explained by a novel vacancy-induced amorphization mechanism. The results provide new insights into the thermal fatigue behavior of tungsten for fusion applications.

One-step preparation and performance evaluation of SPTAs/CLAM joint via powder metallurgy route

Xiaoyue Tan, Shuyuan Liu[1], Chenjun Xu, Qingbo Tu[1], Laima Luo[1], and Yucheng Wu[1]
[1]Hefei University of Technology, China

Self-passivation tungsten alloys (SPTAs) and China low activation martensitic (CLAM) steels are promising to be the first wall and structural materials respectively in future fusion reactor from the consideration of safety operation. However, preparation SPTAs/CLAM component faces several challenges including microstructure controlling, inevitable residual stress and the formation of brittle intermetallic phase, due to the existence of metallurgical reaction and significant difference of melting point and coefficient of thermal expansion between SPTAs and CLAM steel. Here, the spark plasma sintering (SPS) technique is employed to directly consolidate the powers to prepare SPTAs/CLAM joint, and the vanadium interlayer is introduced to relief the thermal stress.

The critical of one-step preparation SPTAs/CLAM joint using raw powders is the design of gradient temperature field. Here will show COMSOL Multiphysics system was used to simulate the temperature field during the SPS consolidation of SPTAs/CLAM joint based on the construction of electro-thermal contact model and to optimize the geometric shapes and dimensions of graphite tools. A sharp temperature gradient of 251 oC can be achieved in the step-typed graphite tool. Basis of the simulation results, the preparation of the SPTAs/CLAM and SPTAs/V/CLAM joints using SPS technique will be also reported. The SPTAs/CLAM joint featured wave interface can tolerate 20 times thermal fatigue test (300-700 oC) and possesses a shear strength of 246±11 MPa, and the SPTAs/V/CLAM joint can only tolerate 10 times thermal fatigue test but has a shear strength of 295±6 MPa. These are attributed to the wave interface could relief effectively thermal stress and the V interlayer could avoid the formation of brittle intermetallic (Fe-W) from microstructure characterization. In addition, how the current direction effects on the formation of brittle phase at interfaces will be reviewed. These are important explorations to develop a reliable technology to achieve SPTAs/CLAM joint featured high performance and promised to apply in future fusion reactor.

Session V

KEYNOTE: Recent Progress of PFMC R&D and Testing Facilities toward Fusion Reactor in ASIPP

Guang-Nan Luo[1]
[1]Institute of Plasma Physics, Chinese Academy of Sciences, China

In the past years, ASIPP has focused its efforts on advancing the technology of plasma-facing materials and components (PFMC) toward fusion reactors in China, especially, the fabrication of divertor and blanket, as well as the development of relevant testing facilities.

ITER-like mono-block and flat-type W-CuCrZr PFCs have been developed for ESAT divertor application in experimental campaigns since 2015. And the plasma-wall interactions such as W impurity generation and transport, and mitigation methods, have also been investigated under the unique environment of the full W wall in EAST.

As to the blanket R&D, Chinese modified RAFM steels are being developed together with an advanced water-cooled blanket design. A systematic investigation of the thermo-mechanical properties as well as the in-service performance of these RAFM steels is ongoing. A half-scale blanket module with W armor on RAFM steel was fabricated.

In addition, PFMC testing infrastructure has been actively developed in ASIPP, including a new superconducting linear plasma machine and a high heat flux test device in the

Comprehensive Research Facility for Fusion Technology (CRAFT), one of the new Chinese National Major Scientific and Technological Facilities. The linear machine has achieved a continuous ~1024 m-2s-1 plasma flux for more than 24 hours. The high heat flux testing device has a total power of 860 kW and is capable of testing full-scale modules of divertor and blanket.

Nuclear materials research at the University of Birmingham

Yu-Lung Chiu[1]
[1]University of Birmingham, UK

Nuclear energy is regarded an important element of future energy mixture spectrum in the UK. The last decades have seen the investment in R&D for fission and fusion energy research continuously increased. This talk will summarise the recent development at the University of Birmingham focusing on fusion materials R&D, including the corrosion and degradation of RAFM steels.

Progress on Standardization of Mechanical Testing of Small-scale Samples for Fusion Reactor Materials

Bin Long[1], Xinfu HE, Ke JIN, Lifeng He, Kaishu Guan, and Bintao Yu
Beijing, China

To meet the design and construction requirements of the China Fusion Engineering Test Reactor (CFETR), it is crucial to develop high-temperature and radiation-resistant candidate structural materials and establish rapid evaluation methods for these materials' radiation performance. Evaluating the radiation resistance of structural materials using small/micro-sample technology has advantages such as occupying less irradiation space, lower radioactive dose, and simpler sample preparation. However, the current testing methods are not unified, and the data cannot directly assess the standard performance of the materials, making standardization essential for engineering applications.

The purpose of this study is to obtain systematic test data on the tensile, impact, fracture toughness, fatigue, and creep properties of low-activation ferritic-martensitic steel (CLF steel) and ODS steel as they vary with sample size and their correspondence to standard samples, to improve existing theoretical models. It also aims to establish hot cell testing technology and to conduct mechanical property tests on neutron irradiated samples with 3 dpa to verify the validity of small sample testing techniques. Additionally, it seeks to establish standardized testing techniques and methods for micro-sample after ion irradiation, with the goal of creating a series of standards for the mechanical properties testing of irradiated small samples that can be used for engineering applications.

The project is led by the China Institute of Atomic Energy, with participation from the Institute of Materials of the China Academy of Engineering Physics, Beijing Institute of Technology, East China University of Science and Technology, the Institute of Modern Physics, and the Nuclear Industry Standardization Institute. The project team possesses capabilities in neutron irradiation, ion irradiation, small sample hot cell testing, micro-sample analysis, size effect analysis, and standards establishment, which effectively supporting the progress of this project.

Kyoto Fusioneering: Overview of Fusion Materials R&D Activities

Mario Oliver[1], Naoko Ashikawa[2], Colin Baus[2], Paul Barron[2], Reuben Holmes[2], Richard Pearson[1], Taishi Sugiyama[2], and Satoshi Konishi[2]
[1]Kyoto Fusioneering UK Ltd, UK, [2]Kyoto Fusioneering Ltd, Japan

Kyoto Fusioneering (KF) is advancing the field of fusion materials through internal research, significant demonstration projects, and strategic collaborations. With the establishment of unique integrated testing facilities, UNITY-1 and UNITY-2, KF is developing fusion power thermal and fuel cycle systems. These facilities will allow for the comprehensive simulation and evaluation of material performance under conditions more relevant to a fusion power plant in terms of temperature, magnetic field and tritium presence.

For the thermal cycle, and specifically the blanket system, materials must withstand harsh plasma facing environment while demonstrating stability at high temperature and neutron fluxes, as well as compatibility with all possible contact media. KF is exploring the use of fusion-grade SiCf/SiC composites, a promising material for breeding blankets due to its capability of withstanding extreme conditions, including blanket operating temperatures around 1000 °C and high neutron loading. KF’s latest accomplishments include the successful manufacturing of a small blanket mock-up, relevant for improving understanding in design of KF's SCYLLA© breeder blanket concept. SiCf/SiC composites and Mo alloys are also being considered for LiPb compatible hear exchangers. Among other phenomena, corrosion products inside LiPb must also be understood.

In the fuel cycle, a number of alloys, including Zr-Co, are of potential use for gettering tritium. Materials for componentry such as proton conductor pumps and palladium alloy foils, important in the management of tritium-containing isotope mixtures and monitoring of tritium inventory in a fusion plant, present materials challenges in their manufacture, tritium compatibility and tritium retention.

Finally, KF’s gyrotrons pose unique challenges in terms of heat management and electrical insulation under high electric field which may benefit from novel materials. Research on structural materials for magnets and current testing capabilities at Japan-site are also explored.

Session VI

KEYNOTE: Prediction of material response in unchartered fusion environments

Chris Hardie[1], Ben Poole[1], Luke Hewitt[1], Vaasu Anandatheertha[1], Vikram Phalke[1], Dave Lunt[1], Eralp Demir[2], and Daniel Long[2]
[1]UK Atomic Energy Authority, UK, [2]University of Oxford, UK

The environment anticipated in a commercial fusion reactor is extraterrestrial. Simultaneous mechanical, thermal and extreme radiation loading of materials is anticipated to radically change mechanical properties within the first few hours of reactor start-up. The acceleration of known failure modes and uncovering of unknown ones is likely, which is concerning given that it is impossible to physically expose materials to fusion environments for testing, before the first reactors are in operation.

UKAEA have initiated a project which aims to develop microstructural (meso-scale) models to support the qualification of materials for these unchartered environments. This approach is centred on finite element-based crystal plasticity and related models, where the representation of microstructure and its evolution provide means to predict property degradation in service. Unlike conventional approaches reliant on large, fixed material property databases, this approach supports engineering analysis beyond the design phases and is enhanced (and not contingent upon) in service surveillance testing by continuous development. The models provide versatility in their ability to span complex simultaneous loading that is more representative to applications, which is well beyond that catered for by databases and nuclear design codes. Therefore, simulations can reduce the ambiguity and associated over conservatism in conventional approaches, thus provide much needed breadth to design envelopes limited by the demanding fusion environment. The project is underpinned by targeted use of surrogate irradiation environments and development of bespoke rigs, state-of-the-art micromechanical testing and physically based model development. This talk will describe early efforts within this project, which include conducting targeted irradiation experiments, the development of rigs and the application of novel experiments to feed the continuous development of physically representative models. An early case study demonstrating application for simultaneous mechanical, thermal and radiation loading will be given.

High performance W-ZrC alloy and W nanoparticle reinforced Cu for plasma-facing components

Rui Liu[1], Zhuoming Xie[1], Qianfeng Fang[1], Xuebang Wu[1], and Changsong Liu[1]
[1]Institute of Solid-State Physics, HFIPS, Chinese Academy of Sciences, China

Tungsten is one of the most promising candidate materials for plasma-facing components (PFCs), however, pure tungsten exhibits drawbacks such as low-temperature brittleness, recrystallization brittleness and irradiation-induced embrittlement. In order to improve the comprehensive properties of tungsten materials, we explored nano-carbide dispersion and grain boundary/phase interface control methods and developed high performance nanostructured W-ZrC alloy plates suitable for industrial production. Compared with ITER-grade W, the W-0.5ZrC alloy exhibits higher strength, lower ductile-to-brittle transition temperature (DBTT~100 °C), higher recrystallization temperature (~1600 °C), and enhanced neutron irradiation resistance. As a candidate material for China Fusion Engineering Test Reactor (CFETR), the W-0.5ZrC alloy was subjected to neutron irradiation (2~4 dpa) in High Flux Engineering Test Reactor. W-ZrC/CuCrZr mock-ups were also fabricated and subjected to 1000 cycles 10-20 MW/m2 high heat load tests. The high performance of W-ZrC alloy makes it a very promising candidate plasma-facing material (PFM) for future fusion reactors. Additionally, neutron embrittlement in tungsten occurs up to temperatures of at least 600-800 °C, strongly limiting the low temperature operational range. Higher operational temperatures above 900 °C may alleviate neutron embrittlement. However, ITER-grade CuCrZr cannot operate at such high temperatures. Therefore, heat sink materials with high thermal stability are required. Recently, W nanoparticles (~7.6 nm) reinforced Cu (NS Cu-W) with high strength and high thermal stability (stable up to 800 °C for 1 h) was prepared by sol-gel synthesis and spark plasma sintering. The high thermal stability of Cu-based material is important to close the operational temperature gap to the PFM.

Modelling high-energy gamma-ray spectrum from deuteron-triton collisions

Natalia Timofeyuk[1], Greg Bailey[2], and Mark Gilbert[2]
[1]University of Surrey, UK, [2]UKAEA, UK

Measuring temperatures between 100 and 150 million degrees, expected to be achieved in future fusion reactors, is a great challenge. Existing methods deliver ion temperature results with a delay from 5-10 minutes to several days after taking the measurements. A promising novel approach to real-time monitoring of deuterium-tritium plasma temperature could be based on detecting γ-rays coming from the d+t --> α+n+γ reaction which occurs approximately once in a few hundred thousand d+t --> α+n fusion events. The new approach requires detailed knowledge of the γ-emission spectrum, especially in the region of 17 MeV where a relatively narrow peak from the low-energy α-n resonance should be most prominent. We report the first model calculations of the dtγ-spectrum and compare them to existing inclusive on γ-energy measurements. We found out that due to pecilarity of the γ-emitting reaction mechanism the inclusive measurements should be sensitive to the lower cut-off on γ-energy. Comparing our predictions to existing results from accelerators, employing cutoffs of 13 and 14 MeV, and inertial confinement fusion facilities, with the low-limit cutoff of 0.4 to 10 MeV, reveals previously unnoticed contradiction between these two types of experiments. Our predictions favour accelerator measurements and corroborate cutoff dependence observed in inertial confinement experiments. Finally, the calculated reactivity of the d+t --> α+n+γ reaction was found to be strongly dependent on reactor temperature, which makes it suitable for deuterium-tritium plasma temperature diagnostics.

Equivalent Methods to Predict Cavity Swelling in Structural Materials Induced by Neutron Irradiation

Chenxu Wang[1], Wei Ge[1], Yiheng Chen[1], Fengping Luo[1], Jin Wang[1], Bowen Zhang[1], Denghuang Chen[1], Jianming Xue[1], Yugang Wang[1], Shijun Zhao[2], and Steven Zinkle[3]
[1]State Key Laboratory of Nuclear Physics and Technology, Center for Applied Physics and Technology, Peking University, China, [2]Department of Mechanical Engineering, City University of Hong Kong, China, [3]Department of Nuclear Engineering, University of Tennessee, USA

Due to the lack of fusion neutron sources, multi-ion beam irradiation has been widely used to study the radiation effects in structural materials. However, predicting the volume swelling of structural materials induced by neutron irradiation using ion irradiation data has been a long-standing challenge. In this study, based on the mean-field rate theory, we developed a relation between the volume swelling of alloys and irradiation dose for the whole process of void growth. With the relation between swelling with dose, we further established a quantitative method that can predict swelling induced by high-dose neutron irradiation using low-dose data or predict the equivalent dose required for heavy-ion irradiation in the same materials. The swelling induced by neutron irradiation can be well predicted with our experimental results and existing data from neutron and heavy-ion irradiations, verifying the reliability and validity of the equivalent method. Furthermore, we established the PKU multi-ion beam irradiation platform and investigated the effect of the irradiation parameters (including damage rate) under triple-ion beam irradiations. As the damage rate increased, the cavity size, density, and swelling decreased, due to the constraint of cavity nucleation and growth processes. The effect of damage rate on cavity evolution under triple ion irradiation strongly depends on two competing factors: the enhancement of aggregation and binding of H/He/vacancies, and the enhancement of vacancies–interstitials recombination with increasing damage rate. However, due to the lack of systematic understanding of the original interaction progress under triple ion beam irradiation, the synergistic effects on cavity size and swelling obtained from different research groups and studies are controversial. Therefore, a uniform standard for performing irradiation experiments and closer international cooperation are strongly required.

First International Workshop on Materials and Mechanics for Fusion Energy
29–30 August 2024
Institute of Physics, London, UK