Abstracts

A Perspective on the Role of In-Situ TEM for Improved Understanding of Irradiation Effects in Materials

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In-situ transmission electron microscopy (TEM) observation during irradiation is widely regarded as a valuable tool for fundamental radiation effects studies in materials. Important temporal and spatial information can be directly observed such as in-cascade defect cluster production vs. nucleation and growth of clusters from point defects, radiation dissolution or growth of precipitates vs. temperature, nanoscale dislocation-obstacle interactions during tensile straining, etc. The geometric thin foil requirements for in-situ TEM investigations necessarily produce some important differences compared to neutron or "bulk" (typically > several MeV) ion irradiations. These thin foil effects include near-surface defect-free zones and enhanced loss of mobile defects due to surface image force effects. By embracing these thin foil phenomena, important fundamental materials parameters for novel materials (e.g., high entropy alloys or ceramics) can be experimentally derived such as the rate-controlling (slowest) interstitial and vacancy migration energies by observing grain boundary- or surface-denuded zone widths for interstitial dislocation loop and cavity behavior, respectively. Similarly, comparing in-situ thin foil irradiated microstructures with conventional ion beam samples (involving severe lateral constraint by the unirradiated substrate) can provide important insight on the effects of elastic stress on defect cluster nucleation and growth and defect migration. Whereas most TEM samples produced by focused ion beam (FIB) techniques have a nearly planar geometry, there can also be value in producing wedge-shaped samples in order to investigate a wide range of TEM foil thicknesses in a single sample. Collectively, this highlights the importance of in-situ TEM irradiation studies as a valuable scientific probe to understand single- and combined-effect mechanisms.

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Steve Zinkle is the Governor's Chair Professor for Nuclear Materials at the University of Tennessee, Knoxville, with a joint appointment at ORNL. His research encompasses basic and applied materials science and engineering investigations under extreme operating conditions (e.g., high temperatures, applied stresses, and radiation environments), with a particular emphasis on using TEM and other characterization techniques to elucidate the linkage between microstructure and properties/performance in materials. His research interests include using TEM to probe deformation and fracture mechanisms in structural materials, advanced manufacturing, and radiation effects in ceramics, fuel systems, and metallic alloys.

Enabling Rapid Qualification of Advanced Reactor Core Materials with Ion Irradiation

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Radiation-induced degradation of materials properties is the showstopper for advanced reactor concepts, making licensing of designs, without data on the performance of core structural materials, the biggest challenge to deployment. Industry is facing the prospect of not being able to acquire the data they need to support license applications because of the lack of test reactor availability and the amount of time it takes to obtain the required data. The solution to this obstacle to rapid deployment is ion irradiation in combination with advanced characterization and predictive modeling. This talk will describe the hurdles to materials qualification faced by start-up companies with innovative reactor design concepts that require data on the irradiation response of candidate core materials to support their license application. The extreme compression of both time and cost make ion irradiation an attractive alternative to test reactor irradiation for companies that lack the financial certitude to obtain high dpa data from test reactors. The solution is in the form of a predictive tool that incorporates ion irradiation and computational materials modeling to determine the microstructure and mechanical properties of core structural materials, benchmarked against reactor data, and codified in ASTM standards, thus providing licensees with a justification of the efficacy of their core material performance in their safety case to the regulator.

Gary Was is the Walter J. Weber, Jr. Professor Emeritus of Sustainable Energy, Environmental and Earth Systems Engineering, and holds appointments in Nuclear Engineering and Radiological Sciences, and Materials Science and Engineering at the University of Michigan. He has held positions as Director of the Michigan Memorial Phoenix Energy Institute, Associate Dean of the College of Engineering, and Chair of the Nuclear Engineering and Radiological Sciences Department. Professor Was' research is focused on radiation materials science and materials for advanced nuclear energy systems. He leads the development of ion irradiation as a technique for predicting neutron irradiation effects in reactor structural materials. Professor Was has published over 300 technical articles in referred, archival journals, presented over 500 conference papers and talks, delivered over 260 invited talks and seminars, and published a graduate level textbook on Radiation Materials Science in 2007 and a second edition in 2017. He is a fellow of five technical societies and serves as Editor-in-Chief of the Journal of Nuclear Materials.

The Study of Irradiation-Induced Nanostructure Patterning with *in situ* TEM

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Ion beam irradiation is not only an effective simulation tool for the study radiation effects in nuclear engineering materials, but also has been widely used for as a tool for the modification of material structure and properties. This talk summarizes the results of the authors' research group over the past ~35 years on ion beam-induced nanostructure formation and patterning from 1-D to 3-D in several very different materials, especially the 3-D patterning of nano-scale voids in CaF2 and a unique nanofiber/cavity assembly in and Ge, GaSb and InSb, similar to the "fuzz" structure observed on the surface of tungsten first wall component for fusion reactors after exposure to a high flux of helium plasma. Understanding the formation mechanism of the latter structure may lead to a solution for mitigating the fuzz problem thus increasing the durability of the plasma facing materials. Furthermore, the ion beam modified nanostructures may have great potentials for other technological applications. The most recent *in situ* TEM study revealed a direct evidence of 1-D interstitial cluster migration associated with the 3-D cavity superlattice formation in CaF_2, and the progressive formation process of the nano-fuzz like structure under 1.2 MeV Kr ion irradiation in SbGa has also been captured.

Dr. Lumin Wang got his undergraduate education from Beijing University of Technology. He went to the U.S. in 1982 to pursue his graduate education. He received his M.S. and Ph.D. degrees in Materials Science from the University of Wisconsin-Madison in 1984 and 1988, respectively. His thesis research was aimed at better understanding of radiation effects in future fusion reactor materials, with ion beam accelerator and TEM as his major tools. He started to get involved with in situ TEM during ion irradiation when he was a post-doc fellow at Argonne National Laboratory in 1989.

Dr. Wang is a full professor in the Department of Nuclear Engineering and Radiological Sciences, and the Department of Materials Science and Engineering at the University of Michigan-Ann Arbor. His research has been focused on irradiation effects in metals, semiconductors and ceramic materials. His group has a long history of using TEM, including *in situ* TEM during irradiation for the analyses of radiation induced defect formation and evolution, and published extensively in this research area.

Multiscale modelling platform of materials under neutron irradiation - Role of in-situ TEM on the platform development

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The reliability, safety and economics of advanced nuclear energy system are strongly dependent on the performance of materials under neutron irradiation. Due to the long-term and high-cost features of neutron experiments and even the lack of a fusion-relevant neutron source, computer simulation as an indispensable and legitimate physical tool is of great importance to understand microstructure evolution and property degradation of materials under neutron irradiation. Because of the multiscale nature of neutron irradiation damage (from cascade collision to defect evolution and eventually macroscopic property variation), multiscale modelling approaches provide a significant pathway to study irradiation with different temporal and spatial domains. Herein, taking tungsten as a prototypical example, I will introduce our recent progress in developing multi-scale modelling platform of materials under neutron irradiation, which we name as NINUM^3. The developed platform contains two comprehensive databases (i.e., defect property database & displacement cascade database) and a set of computational tools in the multiscale framework (i.e., object kinetic Monte Carlo and cluster dynamics for defect evolution simulation, and dislocation dynamics and crystalline plasticity finite element method for mechanical property calculation). Based on the NINUM^3 platform, we have investigated the microstructure evolution and hardness increase of tungsten under irradiation at different conditions, which are quantitatively consistent with the experimental results from HFIR and BR2 irradiation samples as well as ion implantation with a specific helium-dpa ratio, thus verifying the accuracy and reliability of the present platform. By employing the NINUM^3 platform, we explicitly reproduce the self-assembly process of neutron irradiation defects from chaotic distribution into void lattice, and further predict the irradiated microstructure and corresponding hardness increase/thermal conductivity decrease of tungsten under fusion neutron irradiation based on the neutron spectrum of CFETR up to 5 dpa.

Self-Ion Irradiation Induced Grain Formation

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Nickel polycrystalline foils have been irradiated in-situ using the MIAMI-2 (Microscope and Ion Accelerators for Materials Investigations) system. During these irradiations it was observed that dislocations were generated that over the course of the irradiation accumulated together to form low angle grain boundaries (Figure 1). The irradiations were carried out with 300keV Ni^{+} ions with the same result at various temperatures RT, 250°C, 375°C and 475°C.

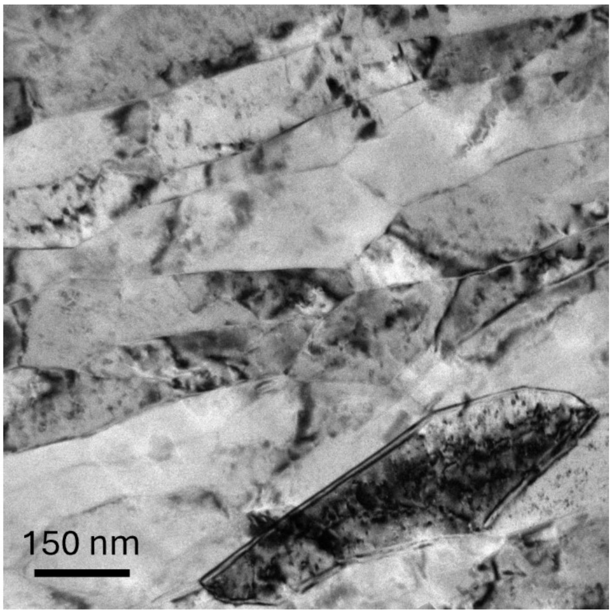
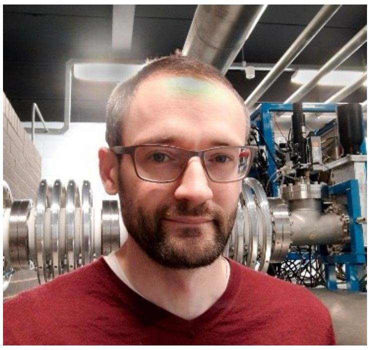


Figure 1: A micrograph showing an example of the low angle grain boundaries formed via irradiation of Ni with 300 keV Ni+ ions at 475°C.

The grains formed are of the order 200 nm wide and microns in length and are oriented with the long axes parallel to the edge of the foil. Following irradiation, analysis using nano beam diffraction of the grain structure made it possible to determine the angle between neighbouring grains with it commonly being less than 1°.

The Ni foils for this work were prepared for transmission electron microscopy via electro chemical jet polishing. This resulted in the sample having a thickness gradient towards the central hole. It is theorized that this thickness gradient combined with the energy of implantation creates a non uniform distribution of the radiation damage, thus inducing stresses within the sample. As the irradiation takes place, defects begin to agglomerate forming larger loops which then also go on to agglomerate into dislocation lines, which due to the stress fields align with the sample edge. By the pile up of these dislocations into larger structures the resulting low angle grain boundaries morphology is formed.

Presented will be an overview of the experimental results and the theory of the mechanisms of this effect, including details from modelling simulations performed by colleagues at the University of Helsinki. Finally, some other preliminary results of interest obtained utilizing the MIAMI-2 system will be presented.



Graeme Greaves is a senior research fellow at the MIAMI facility, University of Huddersfield and has been involved with in-situ TEM with ion irradiation since his PhD (2012). GG currently is responsible for the operation and maintenance of the MIAMI-2 system which he operates for both external and internal projects.

The Vision for Tennessee In-situ Ion Irradiation STEM

Khalid Hattar
Nuclear Engineering-the University of Tennessee, Knoxville

This presentation will serve to highlight both the updates made to the In-situ Ion Irradiation Transmission Electron Microscope (I^3TEM) since the last Workshop on TEM with in-situ Irradiation (WOTWISI) meeting, as well as the vision for the development of an In-Situ Scanning Transmission Electron Microscopy (STEM) facility at the University of Tennessee, Knoxville. The I3TEM facility [1] at the Ion Beam Lab at Sandia National Laboratories underwent several major upgrades shortly after the last WOTWISI meeting. This included the inclusion of a Waviks Gas Injection System (GIS) converting the I^3TEM into an Environmental TEM (ETEM). In collaboration with Professor Lewys Jones at Trinity College Dublin, the I^3TEM also became the first TEM in the world to incorporate a User Adjustable Pole-piece (UAP). The UAP permits the individual microscopists to choose their preferred pole-piece gap and subsequent resolution and experimental flexibility.

In January 2023, the University of Tennessee, Knoxville decided to revitalize the Ion Beam Materials Lab [2] located in Senter Hall, seen in Figure 1A. This includes the addition of a JEOL 2100+ STEM that was delivered in late 2023. The JEOL 2100+ STEM is located in the same lab as a 3 MV NEC tandem accelerator and where a 300 kV NEC, a 20 kV Nonsequitur technologies, and 2 kV Nonsequitur technologies will be delivered before the end of 2024. The ultimate vision of the microscope in the Tennessee Ion Beam Materials Lab (TIBML) is to create a facility that permits testing over a range of controlled environments. This JEOL 2100+ has already undergone several modifications already in order to permit such a vision in the coming years. Working from the top of the STEM down, these include the addition of another condenser lens increasing the potential brightness of the electron beam, an optical port aimed at the STEM filament permitting the potential nanosecond Dynamic TEM (DTEM) experiments, an optical port aimed at the TEM sample (Figure 1C) and an associated 20 W 1064 nm laser from IGP optics, a Waviks GIS, a Waviks optical port [3] and finally an ion port to connect to the ion beam to the STEM.

Acknowledgements: Drs. Miguel L. Crespillo and Sang Tae Park as well as Aaron Graham, Steven Frankowski, Jhovanna Garcia, Jacob Petersen for assistance in developing the Tennessee In-situ Ion Irradiation Ion STEM. This work was performed, in part, at the Center for Integrated Nanotechnologies, an Office of Science User Facility operated for the U.S. Department of Energy (DOE) Office of Science by Los Alamos National Laboratory (Contract 89233218CNA000001) and Sandia National Laboratories (Contract DE-NA-0003525).

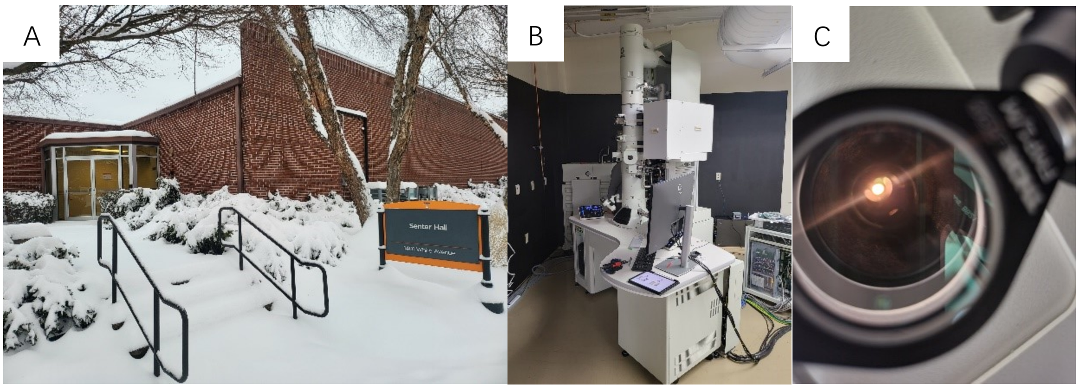


Figure 1: A) The Tennessee Ion Beam Materials Laboratory is located within Senter Hall at the University of Tennessee, Knoxville. B) The JEOL 2100+ STEM has been delivered and is being developed. C) The STEM has been already outfitted with an optical and ion port. Light seen external to the TEM originating from the JEOL IDES Pump Laser Alignment Holder (PLAH).

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[3] Parrish, Riley J., Daniel C. Bufford, David M. Frazer, Caitlin A. Taylor, Jacob Gutierrez-Kolar, Daniel L. Buller, Brad L. Boyce, and Khalid Hattar. "Exploring coupled extreme environments via in-situ transmission electron microscopy." Microscopy Today 29, no. 1 (2021): 28-34.

Khalid Hattar is new Associate Professor in Nuclear Engineering at the University of Tennessee, Knoxville and Director of the Tennessee Ion Beam Materials Lab. He received his doctorate from University of Illinois, Urbana-Champaign and has over 18 years of experience in radiation damage effects and in situ electron microscopy in a large range of materials systems. He has developed a range of in situ techniques to explore the microstructural and property response of materials to combined extreme conditions at both the TEM and SEM length scale.

Combined irradiation study of radiation damage in reactor pressure vessel materials

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The one-dimensional migration behavior of dislocation loops is a hot topic for radiation damage studies. Neutron irradiation produces nanoscale loops in reactor pressure vessel (RPV) materials, resulting in hardening and accompanying embrittlement, which is a critical concern for the life extension of nuclear reactors. Loop study using traditional neutron irradiation is challenging because of the small loop size. Such study is facilitated via experiments combining in situ irradiation in transmission electron microscope (TEM), and pre-irradiation which produces highly-visible loops.

The present study employs combined irradiaiton method to investigate the change of loop Burgers vector under irradiation relevant to the one-dimensional migration of dislocation loops in RPV material. The purpose is to reveal the relationship between the type of loops and irradiation dose, and provide the basis for the prediction of irradiation embrittlement. The pre-irradiation and subsequent in situ irradiation, are carried out with a tandem-type accelerator located at the High Fluence Irradiation Facility at the University of Tokyo (HIT). In the experiment, ferritic model alloy Fe-Ni is used. First, a bulk sample is irradiated to 1 dpa at 400 ºC, and then the foil specimen fabricated from pre-irradiated sample receives in situ irradiation at 400 ºC up to 0.3 dpa under TEM observation (see Fig. 1). It is found that nanometre-sized loop can migrate one-dimensionally, following which the large loop can absorb the small loop. The absorption of loops can disobey the conservation law $\boldsymbol{b} = \boldsymbol{b}_1+\boldsymbol{b}_2$, for both 1/2<111> and <100> loops. This throws light on the further studies of the change of loop Burgers vector under irradiation.

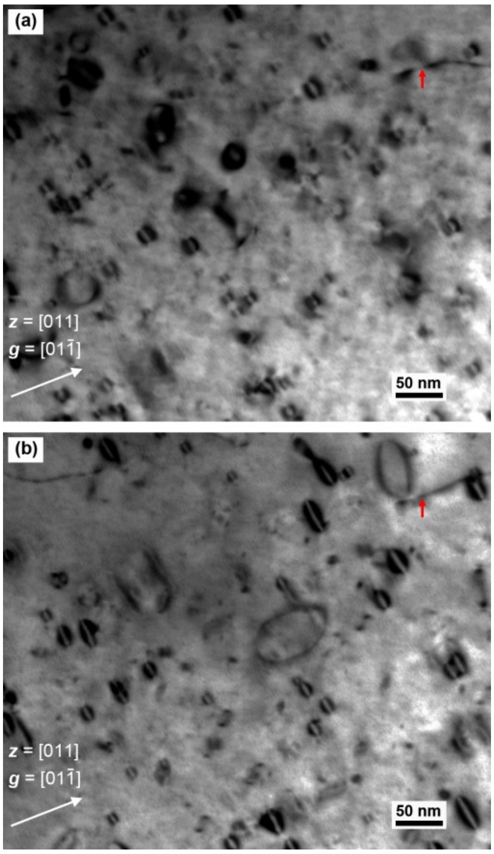

Fig. 1 TEM images of the same area after pre-irradiation and after secondary irradiation under reflections $[01\bar{1}]$.

Dr. Liang CHEN is currently an Associate Professor of Paris Elite Institute of Technology, Shanghai Jiao Tong University, China. He received his undergraduate degree from Tsinghua University in 2012, his master's degree from Tsinghua University and Ecole Centrale de Marseille, France in 2014, and his Ph.D. degree from The University of Tokyo, Japan in 2017. His research areas include radiation effects in nuclear materials, and multi-scale modelling. The primary focus is irradiation embrittlement of reactor pressure vessel steels and irradiation corrosion of zirconium alloys for fuel cladding. He has won the Nuclear Safety Scholarship from World Association of Nuclear Operators, the Young Scientist Award from Atomic Energy Society of Japan, etc. He now leads projects supported by National Natural Science Foundation of China, Natural Science Foundation of Shanghai, and China National Nuclear Corporation.

Electron irradiation in nanomaterials

Bo Da, Jiangwei Liu, CRETU, Ovidiu, Daiming Tang, Hideki YOSHIKAWA, Shigeo TANUMA, Advanced Materials Characterization Field, Center for Basic Research on Materials, National Institute for Materials Science, 305-0044 1-1 Namiki Tsukuba Ibaraki JAPAN

The irradiation effect of electrons, though relatively weak compared to other forms of radiation, can still induce notable changes in nanomaterials due to energy transfer processes. In exploring this phenomenon, two distinct experimental systems have been instrumental in elucidating the effects of electron irradiation on nanomaterials and leveraging these effects to obtain unique properties.

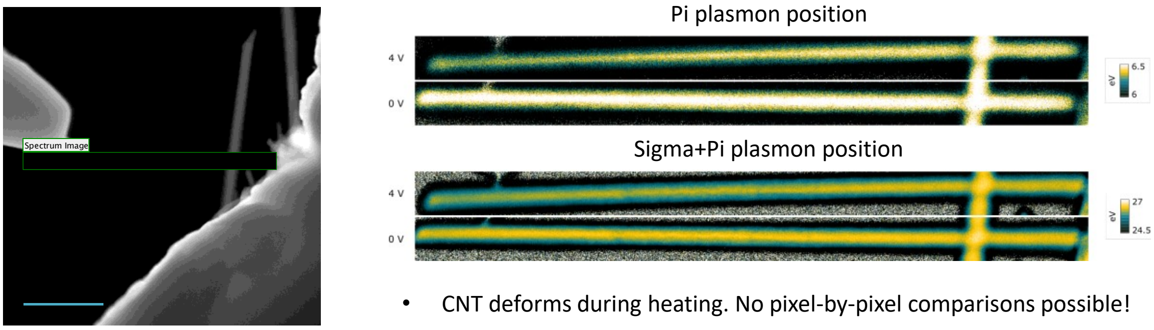

Fig.1 Single wall CNT deforms during heating by TEM.

The first experimental system involves Transmission Electron Microscopy (TEM), a powerful tool for probing nanoscale structures. Researchers have conducted studies on the electron energy loss spectrum of suspended carbon nanotubes using TEM [Carbon. **201** (2023) 1025-1029]. Remarkably, these investigations have revealed a significant rise in temperature along the entire length of the nanotube under electron irradiation. This temperature increase suggests that electron energy is being effectively transferred to the nanotube structure, leading to localized heating phenomena. Such observations provide crucial insights into the thermal dynamics of nanomaterials under electron bombardment and open avenues for controlled manipulation of their properties.

In the second experimental setup, Auger Electron Spectroscopy (AES) is employed to investigate the scattering transport behavior of electrons in two-dimensional (2D) materials [Nature Comm. **8** 15629 (2017)]. By analyzing electron energy loss energy spectra obtained from measurements of 2D materials, researchers can discern intricate details of electron scattering processes within these nanomaterials. This approach offers a comprehensive understanding of how electrons interact with and traverse through 2D materials, shedding light on their electronic properties and potential applications in nanoelectronics and quantum technologies. Through precise control and manipulation of electron scattering behaviors, novel functionalities and enhanced performance can be engineered in 2D materials, paving the way for innovative advancements in nanoscience and technology.

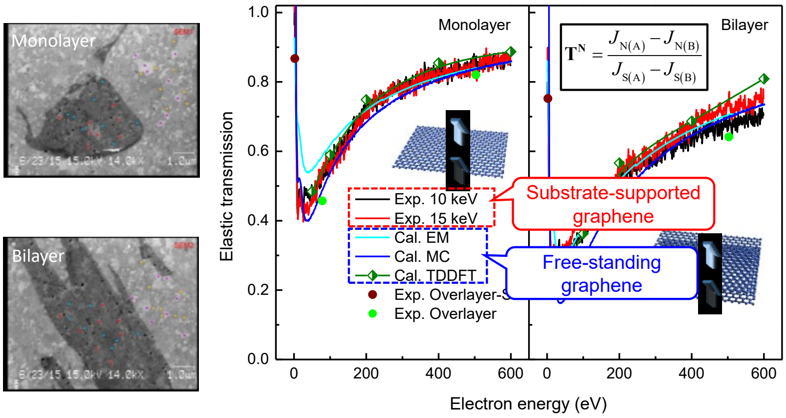

Fig.2 AES was utilized to measure the electron transmittance of monolayer and bilayer graphene. In both experimental systems, the irradiation effect of electrons serves as a fundamental mechanism for inducing structural, thermal, and electronic modifications in nanomaterials. By harnessing this effect, researchers can tailor the properties and behaviors of nanomaterials with unprecedented precision, unlocking their full potential for a wide range of applications spanning from nanoelectronics to energy storage and beyond.

In 2008 Dr Bo Da obtained a BS in Physics from University of Science and Technology of China (USTC) and in 2013 a PhD in physics from the same university. In November 2013 he moved to the National Institute for Materials Science (NIMS) (Tsukuba, Japan) as a Postdoctoral Research Fellow, in January 2015 becoming an ICYS Researcher at their International Center Young Scientists (ICYS), in December 2015 becoming a Researcher in the Center for Materials Research by Information Integration (Mi2) and promoted as Senior Researcher in April 2019.

High Resolution Electron Microscopic Characterisation of Irradiation-induced Elastic Strains in a thin TEM sample

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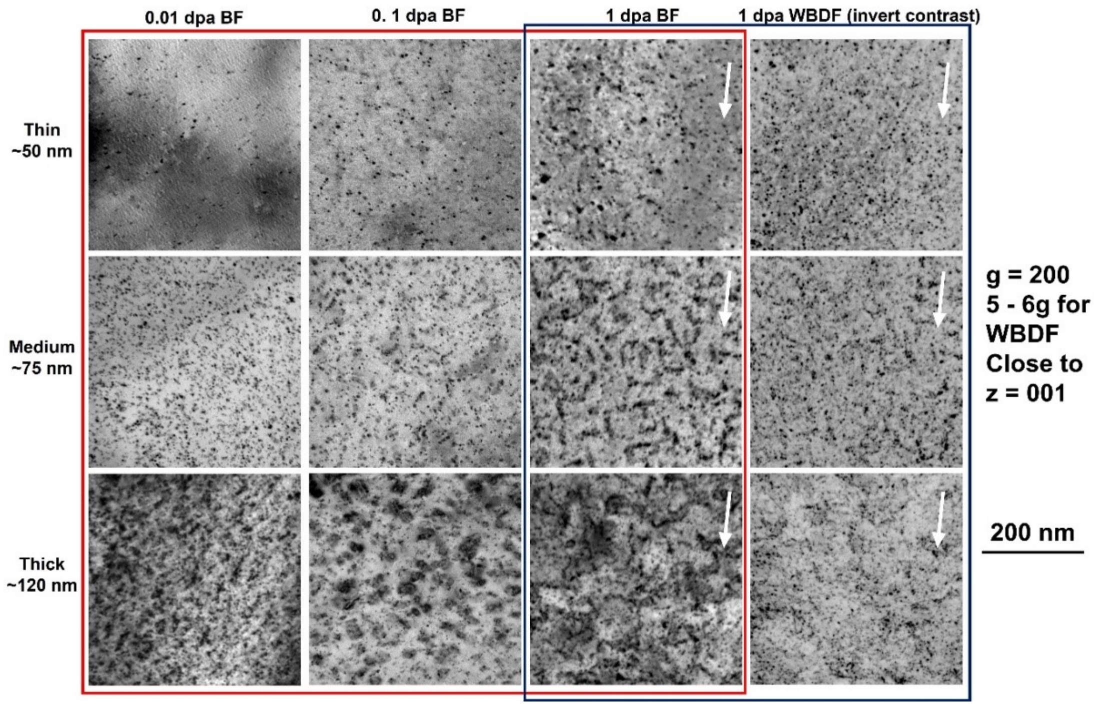
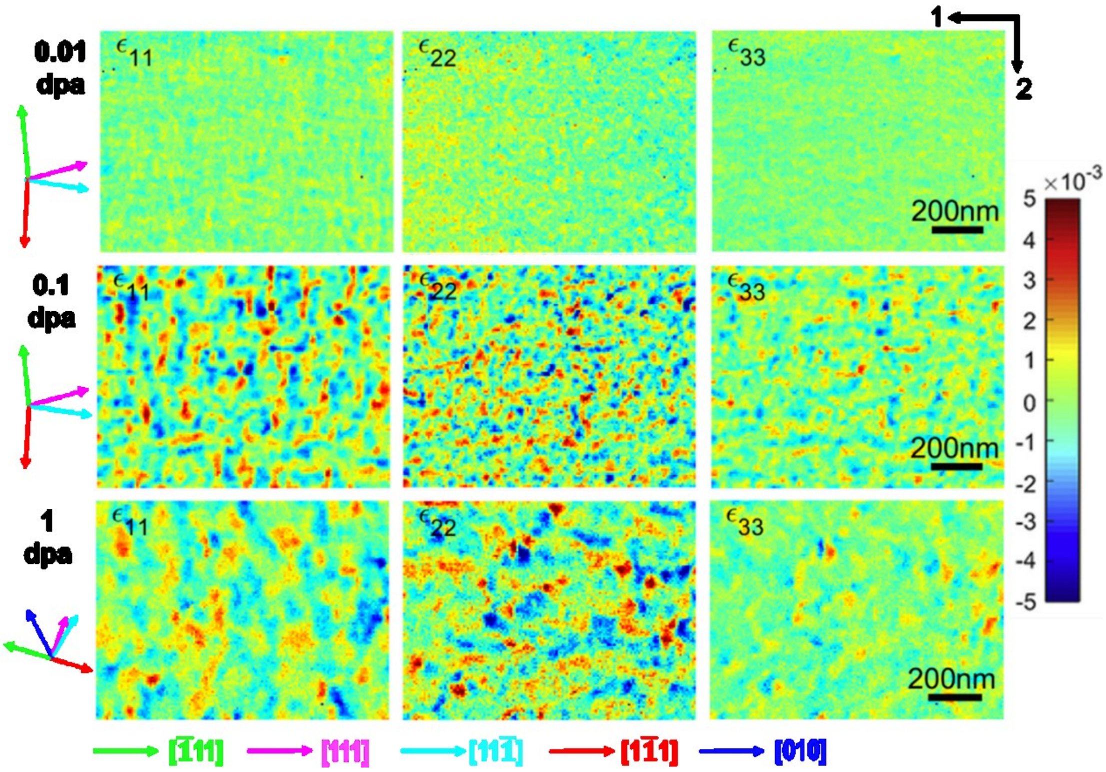
Elastic interactions play an important role in controlling irradiation damage evolution, but remain largely unexplored experimentally. Using transmission electron microscopy (TEM) and high-resolution on-axis transmission Kikuchi diffraction (HR-TKD), we correlate the evolution of irradiation-induced damage structures and the associated lattice strains in self-ion irradiated pure tungsten. TEM reveals different dislocation loop structures as a function of sample thickness, suggesting that free surfaces limit the formation of extended defect structures that are found in thicker samples. HR-TKD strain analysis shows the formation of crystallographically-orientated long-range strain fluctuation above 0.01 dpa and a decrease of total elastic energy above 0.1 dpa. This work has demonstrated a method to use electron microscopic techniques complementarily to investigate the strain in a thin sample.





Dr Guanze He obtained PhD in Materials from the Department of Materials, University of Oxford in 2020, and then worked as a postdoctoral researcher in the Departments of Materials and Engineering Science in the University of Oxford from 2020 to 2023, focusing on the study of radiation damage and corrosion of metallic alloys. Then he joined Shanghai Nuclear Engineering Research and Design Institute as a Senior Research Engineer and pursues the research on the degradation of materials used in the nuclear power plants.

Research on the micro-deformation mechanisms and irradiation effects of high-entropy thin films

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Radiation defects such as swelling and embrittlement will appear in the nuclear materials under the strong irradiation environment, which threatens the safe operation of the reactor. Therefore, developing advanced nuclear engineering materials is crucial for the fourth-generation reactors. Nano-metallic thin films with high-density grain boundaries can effectively absorb irradiation defects, or promote the annihilation of interstitial and vacancy, thereby enhance the radiation damage performance of materials. Nano-metallic multilayer films not only have high-density grain boundaries but the high-density layer interfaces can also improve the mechanical properties and radiation resistance. High-entropy alloys are new candidate materials for fourth-generation reactors due to their excellent thermal stability, high-temperature strength, and corrosion resistance [1-5]. In this work, high-entropy single-layer films and multilayer films are prepared by magnetron sputtering, then the micro-deformation mechanism and irradiation effects are investigated. The CoCrCuFeNi HEA film shows an excellent irradiation resistance, and a nanoscale FCC-to-BCC structure transformation can be observed during irradiation, which effectively alleviates the internal stress caused by irradiation. The deformation mechanism of Cu/TaNbMo medium-entropy multilayer film is investigated, results indicate a strong shear deformation resistance compared to traditional bimetallic multilayer films. Also, it exhibits a good anti-swelling ability, only nanoscale pores are observed in the TaNbMo layer after irradiation.

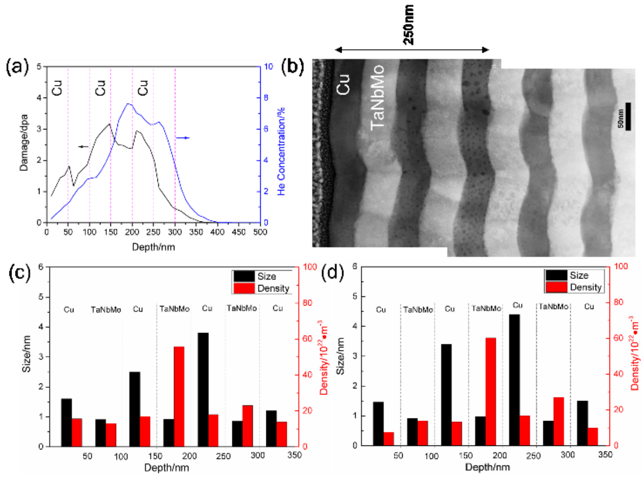

Figure 1 (a) The depth profiles of damage and implanted He concentration calculated for Cu/Ta50Nb25Mo25 50 nm multilayer films after irradiated with 60 keV He^+ to fluence of 1×10^{21} ions/cm^2. (b) The cross-sectional bubble distribution of sample irradiated at 400 °C. Average density and size of helium bubbles along irradiation depth of samples irradiated at (c) 300 °C and (d) 400 °C, respectively.

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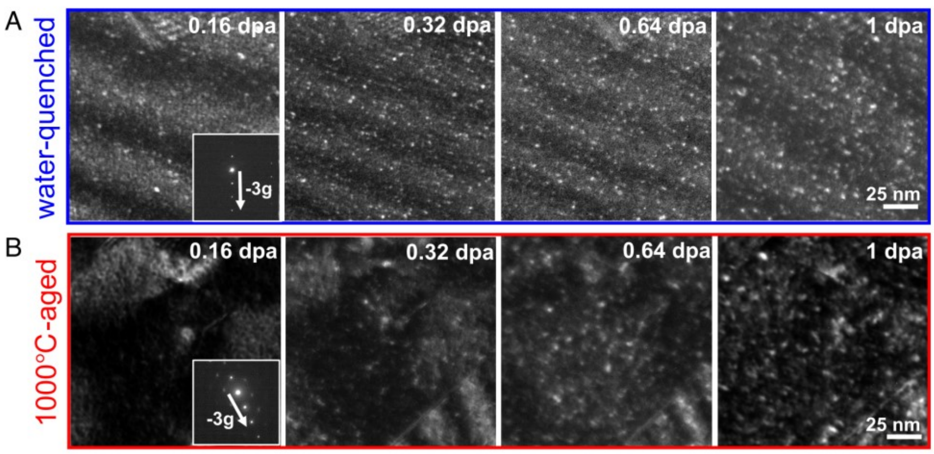
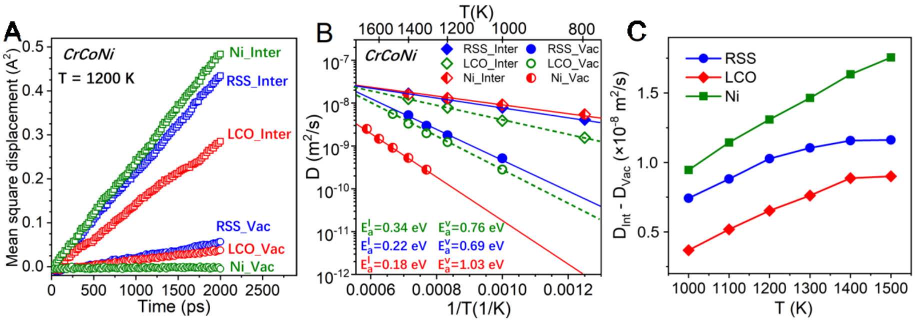
Dr Jiang is an associated professor of Dalian University of Technology. Her research topics are in the field of nuclear materials, including characterization and testing of advanced alloys for structural and cladding applications in nuclear reactors (e.g. high entropy alloys, film materials).

The influence of local chemical order on the radiation behavior of high entropy alloys

Chenyang Lu, Zhengxiong Su, Zhen Zhang, Jun Ding, Tan Shi, En Ma
Xi’an Jiaotong University

In this study, by adjusting the heat treatment process, NiCoCr alloys with varying degrees of local chemical order (LCO) were successfully obtained. Utilizing combined ion beam and transmission electron microscopy for in-situ irradiation revealed that a heightened degree of LCO markedly curtails the early nucleation and growth of irradiation-induced defects. Conjoining first-principles and molecular dynamics simulations, it was discovered that LCO could also impede the diffusion of interstitial atoms, aligning their diffusion rates more closely with those of vacancies. This alignment fosters the annihilation of both point defect types, diminishing the residual concentration of irradiation defects in the alloy. Moreover, the introduction of interstitial C/N atoms into the NiCoFeCrMn alloy adjusted both LCO and fluctuations in chemical composition. Experimental and simulation studies demonstrated that such adjustments significantly enhanced the alloy's LCO area proportion and chemical composition fluctuations, narrowed the migration rate disparity between self-interstitial atoms and vacancies, thereby facilitating defect recombination and suppressing the enlargement of defect clusters. In examining the mechanisms of radiation resistance in HEAs, it was identified that besides the impact of LCO on irradiation defect behaviors, the irradiation dose and temperature reciprocally and significantly influence the stability of LCO within the alloy. This scenario involves a dynamic competition between the radiation-induced disruption and reformation of LCO structures.





Dr. Chenyang Lu is a professor in the Department of Nuclear Science and Technology of Xi'an Jiaotong University. He served as the director of 'Key R&D Project Issues' for the Ministry of Science and Technology. Professor Lu received his PhD in Materials Science from Northeastern University in 2014. He worked at the University of Michigan as a post-doctoral fellow and research scientist in the Department of Nuclear Engineering & Radiological Sciences from 2014 to 2018. He joined Xi'an Jiaotong University at the end of 2018.

Professor Lu engaged in the research of nuclear engineering materials. He dedicated his researches on the preparation of advanced nuclear structural materials, radiation damages in materials, and advanced microstructural characterizations. Professor Lu has published more than 70 SCI peer-reviewed papers, including 4 Nature Communications, PNAS, 7 Acta Materialia, 10 Journal of Nuclear Materials and other related influential papers in this field. The articles have been cited more than 5,000 times, the h-index is 34. In 2013, he won the Presidential Scholar Award from the American Microscopy Society at the conference of M&M 2013.

Study on the microstructure and hardness of Cast ODS steel subjected to 3.0 MeV Fe ion irradiation

Yu Si
Shanghai University

Oxide dispersion strengthened (ODS) steel is a candidate material for cladding of future nuclear fusion reactors. This work aims at exploring the irradiation resistance of ODS steel prepared by casting manufacturing. All specimens were irradiated with 3.0MeV Fe ions at 450 ºC to a influence of 1×10^{16} ions cm^{-2}, corresponding to a peak damage dose of 10.3 dpa. The microstructure, oxide and defects of casting manufacturing ODS steel before and after irradiation were investigate and compared with mechanical alloying (MA) ODS steel.

It is not found the void in all specimens. Y-Ti-O nano-oxide particles have been observed in the MA-ODS matrix, with an increase in the number density and a decrease in size after irradiation. The same phenomenon was observed in cast-ODS steel. It is the main reason for the irradiated hardening. However, unlike MA-ODS steel, cast ODS steel exhibits a higher number density of nano-oxide particles, with a finer and more uniform size. Meanwhile, cast-ODS steel exhibits excellent radiation resistance performance.

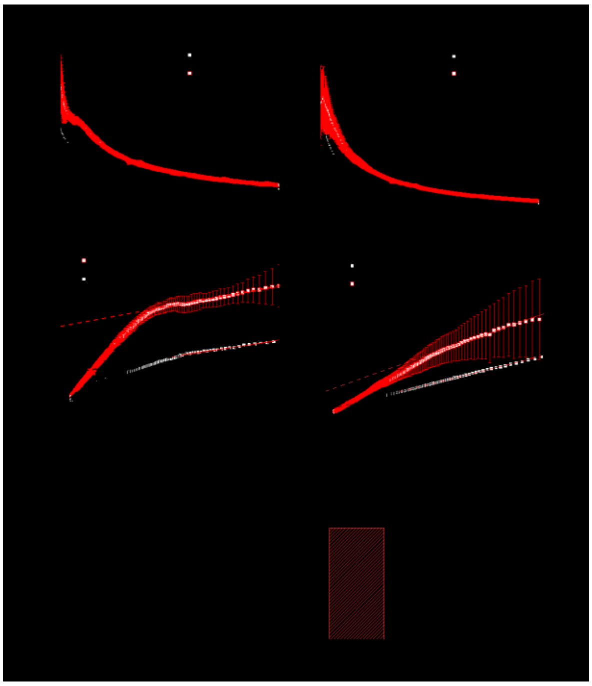


Figure1 Nano-hardness versus indentation depth curves of (a)MA-ODS steel and (b)Cast-ODS steels before and after irradiation; H^2 - 1/h curves of pristine and irradiated specimens: (c) MA-ODS steel and (d)Cast-ODS steels; Hardening increment($\triangle$H) of two specimens before and after irradiated (e).

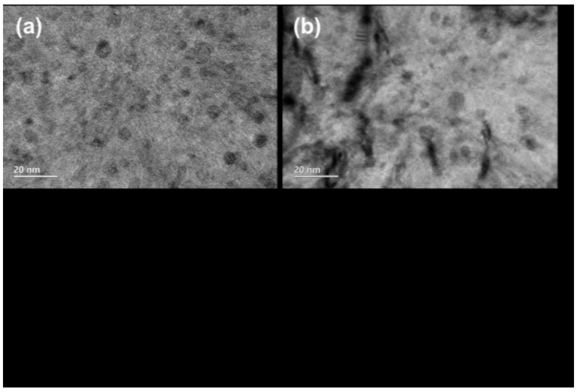


Figure 2 HR-TEM images of nano-oxide particles in MA-ODS steel before (a) and after (b) irradiation. Additionally, it provides a comparison of size distributions of oxide particles in unirradiated (a) ODS steels and after (d) irradiation.

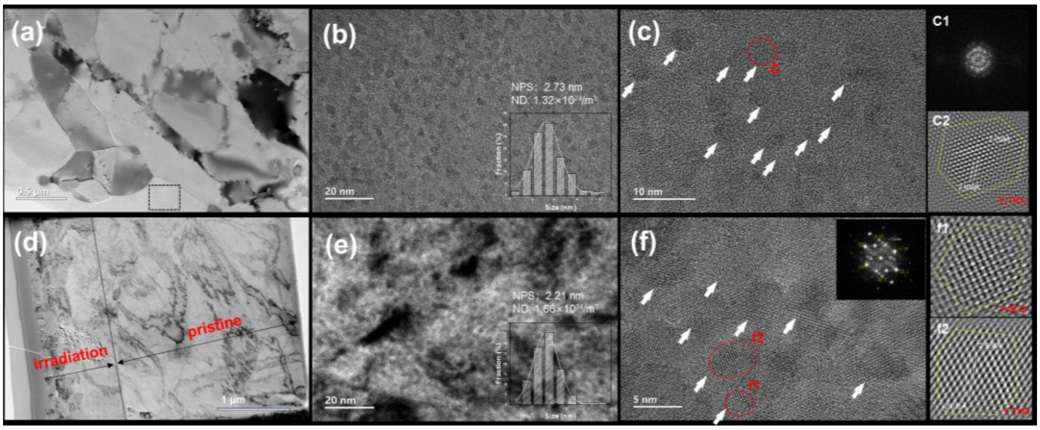


Figure 3 Microstructure and nano-oxide particles of cast-ODS steel before (a, b, c) and after (d, e, f) irradiation as observed by TEM. HR-TEM images (c) and (f) are obtained from (b) and (c), respectively. The FFT pattern (c1) and IFFT figure (c2) of one nano-particle particles marked by the red circle in figure (c) are also shown. Additionally, IFFT figures (f1) and (f2) of two nano-oxide particles marked by red circles in figure (c) are provided for reference.

Name: Yu Si

Education: He is currently working toward the Ph.D. degree in Chemistry with he Center for Nanoscience and Technology at Shanghai University.

Research Interests: Structural materials for nuclear reactors.

He is a proactive person. If you are interested in his research topics, please feel free to contact him via email: sxsiyup@163.com

The *in situ* dual ion beam TEM at JANNuS-Orsay: description of the facility, recent updates and results

Cédric Baumier, Stéphanie Jublot-Leclerc, Aurélie Gentils*
Université Paris-Saclay, CNRS/IN2P3, IJCLab, Orsay, France

In situ Transmission Electron Microscopy (TEM) with ion implantation/irradiation is a key tool to investigate a wide field of topics, such as phase transition far from the equilibrium, solid-state nucleation/growth, and ageing of materials used in nuclear or space industries.
In the 1980's our laboratory was a pioneer in operating this kind of facility [1]. Thanks to this long experience, our *in situ* experimental facility was extended to multiple simultaneous scanned beams, by building an *in situ* dual ion beam TEM setup. Based on a 200 kV Tecnai G^2 20 Twin FEI microscope coupled with two ion accelerators (Figure 1), it offers a large choice of ions and energies (from 10 keV to 6 MeV) and complementary analytical equipment (EDX, STEM and EELS) associated with different specimen holders (from LN_2 temperature up to 1000°C). Launched in 2009 as an open facility, JANNuS-Orsay allows each year several worldwide teams to process their *in situ* TEM experiments through the EMIR&A French accelerators federation [2].
The JANNuS-Orsay experimental hall is part of the MOSAIC facility [3] at IJCLab, an interdisciplinary research ion beam platform supporting many scientific fields ranging from materials science to astrophysics, including geology and nuclear physics. **The JANNuS-Orsay experimental hall groups together the *in situ* dual ion beam TEM, the** IRMA 190 kV ion implanter, and the ARAMIS 2 MV ion accelerator and associated beam lines (implantation/irradiation, ion beam analysis including *in situ* RBS-C, future *in situ* XRD), as shown in Figure 2. The coupling of the Transmission Electron Microscope with ARAMIS and IRMA is unique in the world due to the diversity of elements that can be accelerated inside the microscope in a wide range of energies, with a well-controlled dosimetry, making it possible to characterize *in situ* at the nanometric scale the evolution of structural and chemical modifications of materials subjected to one or two ion beams, at a given temperature.
A description of the JANNuS-Orsay equipment and recent updates will be given in this presentation, with a focus on the *in situ* Transmission Electron Microscope. A selection of scientific results obtained throughout the past years will be given, with examples mostly based on different materials studied for energy applications, and in particular fission and fusion [4].

The JANNuS-Orsay / MOSAIC accelerator staff (in particular C. Bachelet, J. Bourçois, S. Hervé, S. Picard) are gratefully acknowledged for their unfailing assistance during in situ experiments.

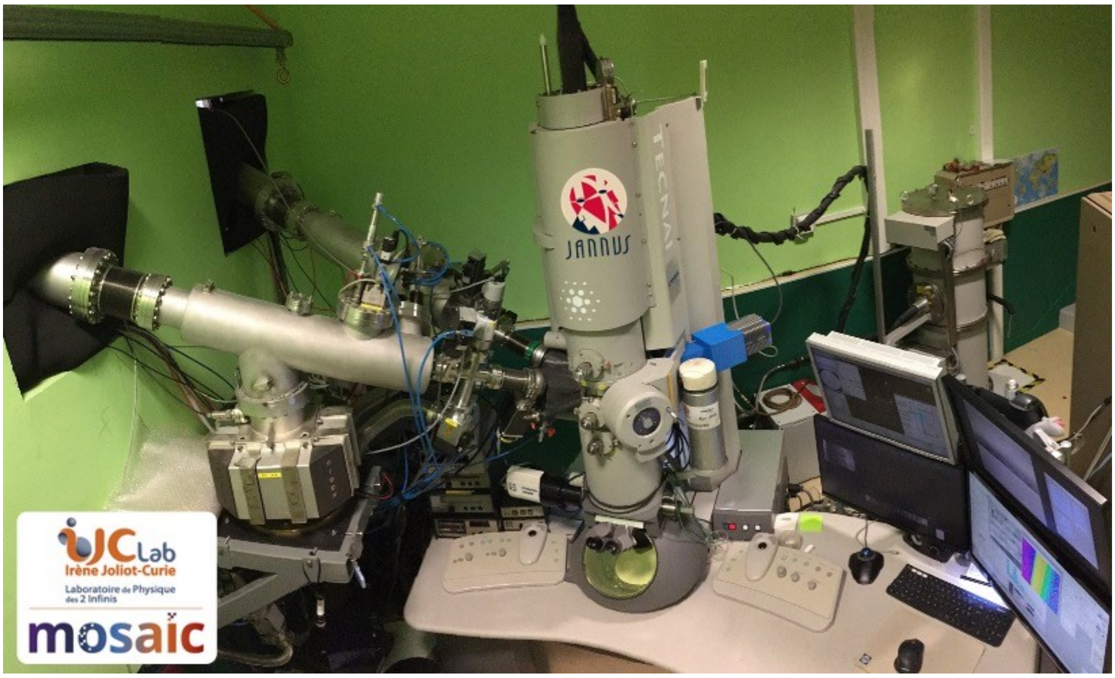


Figure 1 - The in situ dual ion beam TEM at JANNuS-Orsay, MOSAIC facility

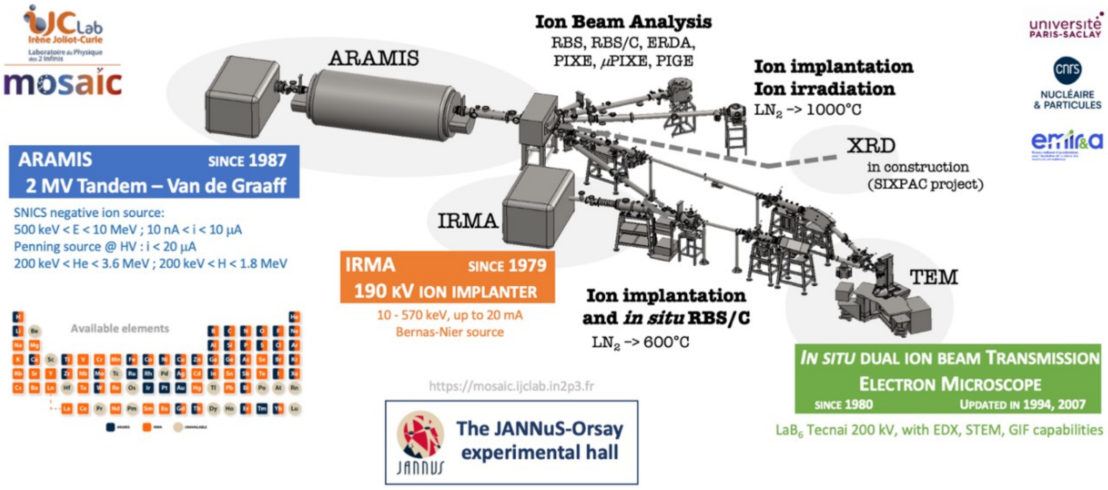


Figure 2 - Overview of the JANNuS-Orsay experimental hall, showing the coupling of IRMA ion implanter, ARAMIS ion accelerator, the Transmission Electron Microscope, and the various ion beam lines dedicated to irradiation/implantation and ion beam analysis

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[4] A. Gentils, C. Cabet, Nucl. Instrum. Methods B **447** (2019) 107

Dr. Aurélie Gentils is currently a research scientist at the CNRS, French National Centre for Scientific Research, assigned to Université Paris-Saclay, CNRS/IN2P3, IJCLab. Her main research interests are related to nuclear materials, using ion beams either for synthesis, modification (implantation/irradiation) or microstructural characterizations of ceramics and metallic alloys. She is the scientific leader of the ion beam MOSAIC facility that includes the *in situ* JANNuS-Orsay dual ion beam Transmission Electron Microscope. She has written over 60 peer-reviewed publications, a book chapter, and has supervised numerous PhD theses and internships.

Sample holders for *in situ* TEM with ions

C. Baumier
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Transmission electron microscopy (TEM) relies heavily on specimen holders, whose stability is crucial for accurate measurements. This importance is amplified when ion beam irradiation is performed inside a TEM, where ion-induced damage on the specimen holder results in significant changes at the macroscopic scale, e.g. thermal, mechanical, or charge effects.
To address these challenges, simplifying specimen holders to single-tilt ones is not a solution that we can accept. Alternatively, adapting specimen holders, such as double-tilt heating or cryogenic stages, can enable them to withstand ion-beam-induced changes under extreme conditions during our *in situ*experiments.
Furthermore, innovation in specimen holder design is essential. Given the limited space in TEM columns, integrating additional functionalities into specimen holders becomes imperative for advanced experimental capabilities.
In our field, specimen holders play a pivotal role in enabling comprehensive advancements, particularly in *operando* experiments. I will present during the WOTWISI-7 conference two examples of specimen holder possibilities enabling new types of experiments under irradiation, being closer to real systems.

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https://mosaic.ijclab.in2p3.fr

Dr. Cédric Baumier obtained his PhD degree in Chemistry-Physics from the University Paris VI in 2011, working on synchrotron instrumentation development. He has been in charge of the JANNuS-Orsay TEM at the MOSAIC ion beam facility since 2014. He improves the TEM for in situ ion irradiations to keep it at the top level of development for the community. He is specialist in making and developing in situ sample holders for electron microscopy (in situ TEM and SEM), to adapt them in a range of operando techniques to study the mechanisms of the materials as close as possible from the real system.

Effects of damage and dose rate on α′ phase formation in oxide dispersion strengthened FeCrAl alloys under Fe ion irradiation studied using in-situ transmission electron microscopy

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Oxide dispersion strengthened (ODS) FeCrAl alloys are promising materials for potential use in accident tolerant fuel claddings for nuclear fission and the first wall of fusion reactors due to their resistance to both oxidation and radiation-induced swelling. However, one challenge is the irradiation-induced formation of the Cr-rich α′ phase which can lead to embrittlement. Whilst this phenomenon has been reported under neutron irradiation, there are limited examples of α′ formation under ion irradiation presented in the literature.

This study has looked at three different ODS FeCrAl alloys with varying Cr contents of 10, 12 and 20 wt.%. These have been 150 keV Fe ion irradiated in-situ of a transmission electron microscope (TEM) using the Microscopes and Ion Accelerators for Materials Investigations (MIAMI-2) facility at the University of Huddersfield (United Kingdom) which comprises a Hitachi H-9500 TEM and dual ion beamlines as shown in Fig. 1(c). Experiments have been performed at room temperature and 300°C to explore the radiation response, and in particular the formation of additional phases, as a function of the irradiation temperature, ion fluence (damage dose) and ion flux (damage dose rate). Additional analysis using scanning TEM (STEM) with energy dispersive spectroscopy (EDS) and electron energy loss spectroscopy (EELS) has been performed using an FEI Titan3 G2 60-300 at Linköping University (Sweden). The results of these experiments will be presented demonstrating that precipitation formation was more prominent at the lower temperature, lower damage dose rates and higher chromium contents. Examples of the results are shown in Figs. 1(a) and 1(b) and will be discussed in terms of possible explanations for this behaviour as well as implications for the neutron case.

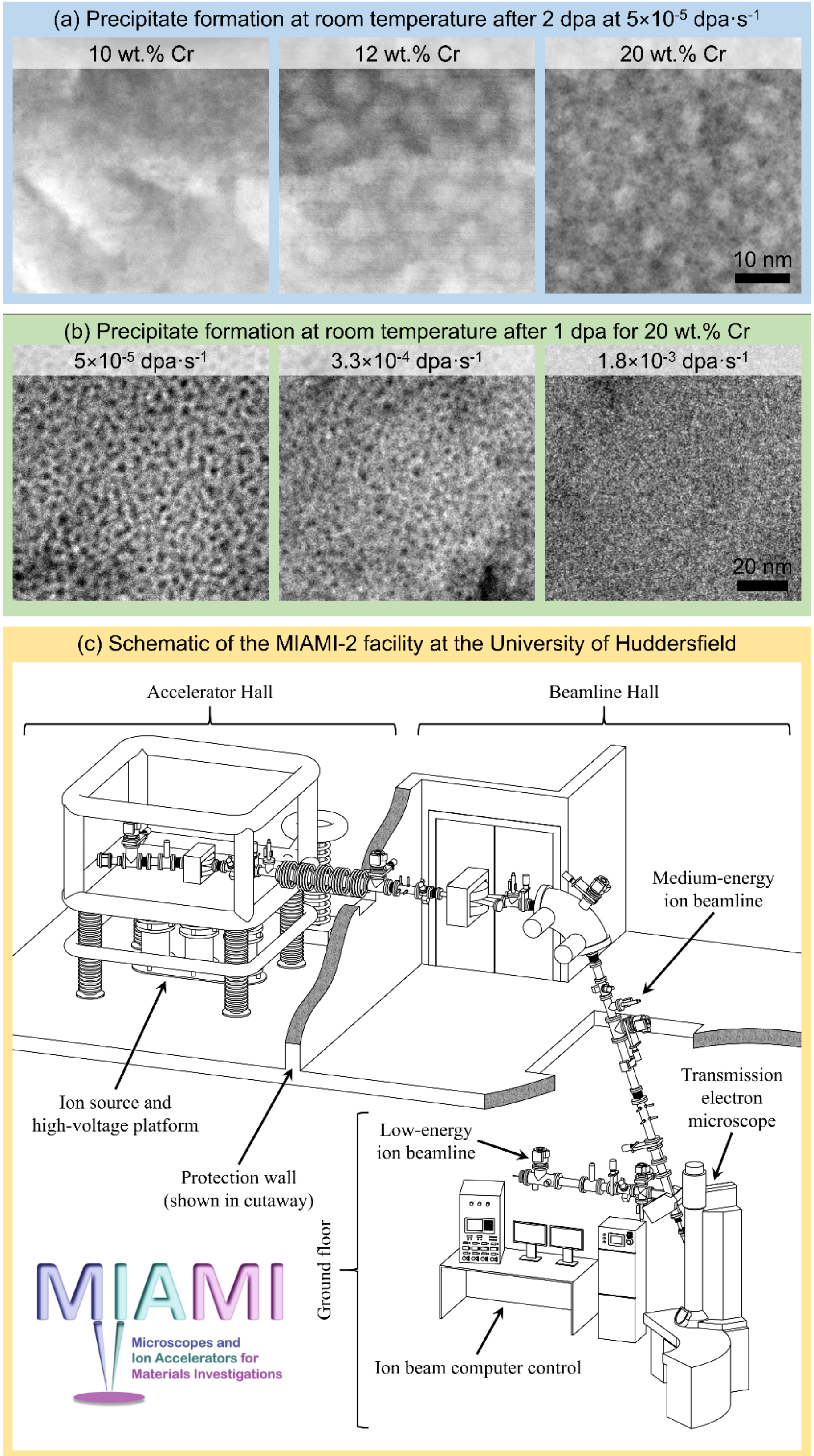

Figure 1: Examples of differences in precipitate formation observed via (S)TEM in ODS FeCrAl with (a) different Cr contents and (b) at different damage dose rates resulting from irradiation with 150 keV Fe ions in the MIAMI-2 facility (c).

Jonathan Hinks is Professor of Radiation Damage in Materials in the School of Computing and Engineering at the University of Huddersfield. He leads the Electron Microscopy and Materials Analysis (EMMA) research group which is home to the Microscopes and Ion Accelerators for Materials Investigations (MIAMI) facility dedicated to the in-situ study of the modification of materials under irradiation.

The TEM-Accelerator Facility at HIT, The University of Tokyo

Hiroaki ABE
The University of Tokyo, Japan

The purpose of this presentation is to introduce the renewed High Fluence Irradiation Facility at The University of -Tokyo (HIT). The HIT facility was first installed on the Tokai Campus (Tokai-mura, Ibaraki, Japan) in 1985. This facility has been modified to have a 1.7 MV tandem Cockcroft-Walton accelerator and a 180 kV ion accelerator with a PIG ion source. Irradiation capabilities are single/dual irradiation at temperatures below 770 K as shown in Figure 1, and an interface with a transmission electron microscope (JEM 2000 FX) as shown in Figure 2.
In the presentation, the details of the facility as well as its recent applications in accident-tolerant fuels, fusion blanket materials, and related irradiation-induced phenomena including in-situ observation experiments will be reviewed.

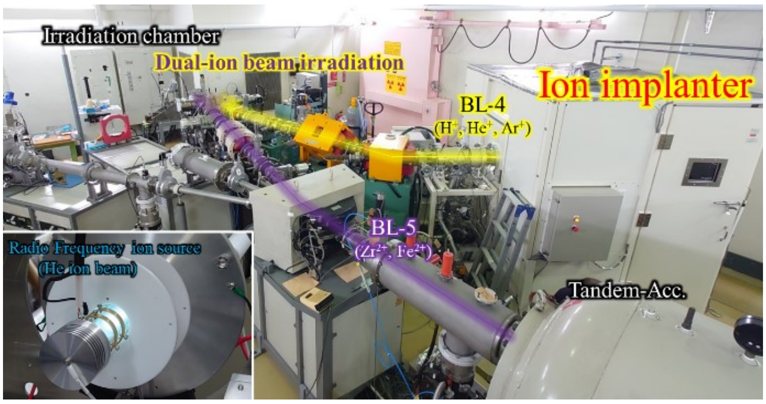

Figure 1. HIT facility at the University of Tokyo


Figure 2. Interface with TEM

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Using In-Situ Transmission Electron Microscopy to Characterize the Microstructure Evolution Under Irradiation: Research highlights from the "MATerials Under Reactor Extremes" (MATURE) research group at NCSU

D. Kaoumi,
Department of Nuclear Engineering, North Carolina State University, USA

Structural alloys used in nuclear reactors are subject to a harsh environment comprised of irradiation, mechanical stress, high temperature and a corrosive medium. Understanding how these materials degrade under such harsh environment is crucial to (i) qualify newly developed materials for in-reactor use or (ii) to predict the lifetime of existing components. Since the changes in macroscopic properties find their roots in the changes in microstructure, characterizing the microstructure evolution is therefore key to understanding and mitigating the degradation of material properties. One of the difficulties of studying processes occurring in a system under external stimulus such as irradiation and/or mechanical stress is the lack of kinetics information, since usually samples are examined ex situ (e.g. after irradiation or after mechanical testing) so that only discrete snapshots of the process are available. Given the dynamic nature of the phenomena, direct in situ observation is often necessary to better understand the mechanisms, kinetics and driving forces of the processes involved. For this matter, using in situ Transmission Electron Microscopy (TEM) can be of great help. Indeed, the spatial resolution of the TEM makes it an invaluable tool in which one can continuously track the real-time response of the microstructure to external stimuli, which can help discover and quantify the fundamental rate-limiting microscopic processes and mechanisms governing the macroscopic properties. In this presentation, various examples will thus be given of how in-situ TEM can be used to answer material science questions for nuclear engineering applications. These examples will include (i) In-situ Ion-irradiation in the TEM for studying the basic mechanisms of irradiation-induced defects (a.k.a radiation damage) formation and evolution as a function of dose, dose rate, temperature in materials of interest and (ii) In-situ straining experiments in the TEM used to investigate deformation mechanisms such as dislocation dynamics in Ni-based alloys and the deformation induced martensitic transformation in 304 Stainless Steels.

Ph.D. Nuclear Engineering, The Pennsylvania State University, 2007.
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B. Sc. Physics, Institut National Polytechnique de Grenoble (INPG), Grenoble, France, 1999.
Dr Kaoumi is a full professor of Nuclear Engineering at North Carolina State University. His research topics are in the field of nuclear materials with an emphasis on development, characterization and testing of advanced alloys for structural and cladding applications in nuclear reactors (e.g. advanced F/M steels, nanostructured ODS steels, and high temperature Ni-based alloys). His interest is to develop an understanding of microstructure property relationships in these metallic systems, especially microstructure evolution under irradiation, high temperature, mechanical load and/or corrosion and how it can impact macroscopic properties which govern the

materials' performance. He specializes in the use of in situ characterization techniques (particularly in situ TEM and in-situ XRD).

Development of engineered nano-nuclear materials using particle beam linked Electron Microscopy

Liviu Popa-Simil
OLAVM LLC

Nuclear power faces a decline amidst the complexities of global warming, economic sanctions, and the transition from hegemony to multi-polarity, despite the nuclear industry's optimistic projections of a "Nuclear renaissance." This optimistic view often overlooks the real reasons behind nuclear power's waning influence. Resistance to change within the industry, primarily due to safety concerns, limits the adoption of innovative approaches that lie outside traditional practices.

A novel concept, dating back to the 1980s, seeks to address these challenges through a philosophy of Harmony, drawing on the traditional concept of "Hehe" or Harmony and Cooperation, but applied in a modern, quantum nuclear context. This approach aims to achieve harmony within the quantum nuclear environment, optimizing outcomes as illustrated in Fig. 1. It envisions nuclear agents and their environment in a multi-scale, multi-dimensional alignment, a perspective that has seen its technology readiness level reach TRL 3 to 3.5, [1].

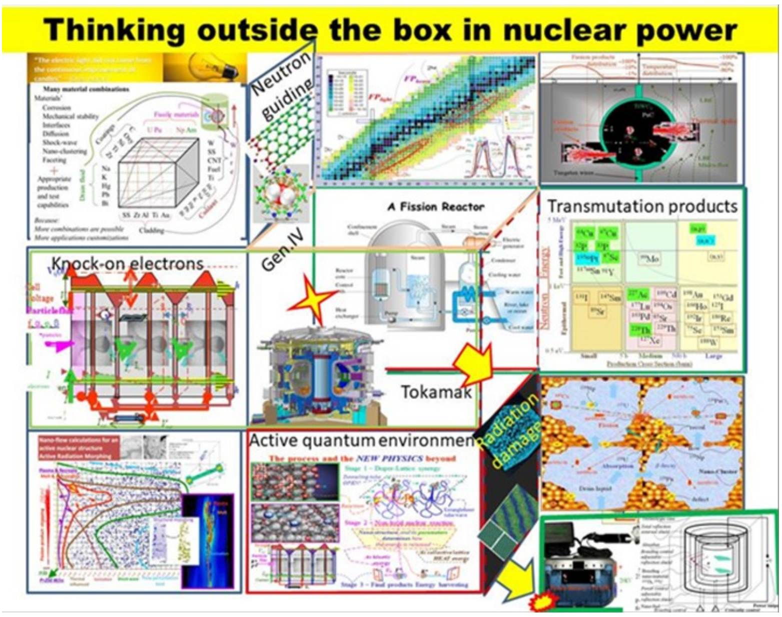

Fig. 1 – the Engineered 6 nuclear-nano materials types

This harmony-based approach has led to the exploration of engineered micro-nano heterostructures, culminating in the identification of six families of nuclear nano materials that could revolutionize nuclear processes and lead to the development of new types of nuclear batteries, [2]:

1. **Fuel Micro-hetero-structures ("Cer_Liq-Mesh")**: These structures utilize microspheres in a liquid to improve nuclear fission efficiency, simplifying fuel processing and enhancing fuel cycle performance.
2. **Nano-beaded Structures**: Leveraging nano-clusters, these structures aim to improve isotopic material purity and address transmutation products.
3. **Fractal Materials**: Designed to recover from radiation damage, maintaining consistent properties regardless of radiation dose.
4. **Nano-hetero Structures as Supercapacitors**: These structures convert kinetic energy from nuclear particles into electricity, enabling more compact reactor designs.
5. **Radiation Guide Nano-hetero Structures**: Aimed at lightweight shielding and reactor control, these materials work with NEMS elements for near-instantaneous response times.
6. **Nano-active Quantum Nuclear Environments**: Potential for triggering high-energy quantum states through low-energy excitation, exploring phenomena like quantum entanglement and

teleportation.

These technological advancements not only pave the way for compact, solid-state nuclear reactors but also open new avenues for applications in space, underwater, and terrestrial environments. They address many challenges of current nuclear technologies and could lead to fusion batteries, among other innovations. This progress could ultimately contribute to a peaceful, balanced cosmic civilization, with ongoing research in particle beam microscopy, computer simulation, and experimental testing playing a crucial role in the development of these energy structures

Keywords: Nano-Micro Engineered Hetero-Structures, Direct Nuclear Energy, Conversion, Nano, Micro, Radiation Guiding, Nuclear Entanglement, Self-repairing, Fractal, Fission, Fusion, Transmutation, Directed Energy, Electric Power, Isotopes

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Dr. Liviu Popa-Simil, is the Executive Director of LAAS - Los Alamos Academy of Sciences, and the president of LAVM LLC. He is a nuclear engineer physicist, graduating from the Nuclear Engineering Faculty in Bucharest, Romania. He worked for Los Alamos National Laboratory, developing Real Time Radiography methods, and then, developed advanced nuclear fuel cycle as part of AFCI program, previously being a senior researcher, program manager at NIPNE-HH, in Bucharest Romania.

Using Proton Irradiation to Study Stress and Temperature Dependence of Irradiation Creep

Brodie Moore, Matt Topping, Fei Long, Mark Daymond
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Zirconium alloys are commonly used in nuclear reactors, and their deformation arising from irradiation creep can be component life limiting. This presentation describes a series of experiments conducted to evaluate the in-situ irradiation creep behavior of Zircaloy-4, a common commercial alloy. We use proton irradiation to emulate the irradiation damage produced by neutron irradiation. Experiments have been conducted at a range of stresses (65 – 105 MPa) and temperatures (250 - 350 °C), and results are in good agreement with results from neutron reactor irradiations, once creep rates are normalised for the different damage rates. Starting with recrystallised material, the impact of early irradiation damage development on subsequent creep rates is also examined. Irradiation damage structures as determined by transmission electron microscopy are comparable to neutron irradiation produced structures. The results are a first set of comprehensive tests carried out at the new irradiation creep facility at the Queen's University tandem accelerator; they demonstrate that proton irradiation can be used for mechanistic investigations of irradiation creep phenomena, which will in the future be applied to material systems for which neutron irradiation creep data is not yet available.

Mark Daymond is a Professor of Mechanical and Materials Engineering at Queen's University in Canada, and Director of the Reactor Materials Testing Laboratory. He holds a Canada Research Chair in Mechanics of Materials, and a UNENE Research Chair in Nuclear Materials, and sits on multiple advisory committees to the Canadian nuclear industry. He obtained his degrees from the University of Cambridge, UK. His research focuses on microstructural degradation mechanisms of nuclear materials, and their impact on thermomechanical properties, with a particular interest in radiation damage and light element effects in zirconium and nickel-based alloys.

Evaluation of hydrogen and helium retention behavior in fusion reactor materials using in-situ TEM-QMS

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Department of Material Science, Shimane University, Matsue, Shimane 690-8504, Japan

In future fusion reactors, plasma facing materials (PFMs) will be exposed to the burning plasma with high density helium, produced by fusion reactions, besides hydrogen isotopes. Many previous studies have shown that retention properties of hydrogen isotope in PFMs are significantly affected by helium pre-irradiation, suggesting the contribution of helium bubbles [1,2]. However, direct evidence of the effect of helium bubbles on hydrogen retention remains elusive. In this study, the effects of helium irradiation on hydrogen isotope retention in tungsten and beryllium were precisely evaluated from a microscopic viewpoint using in situ ion irradiation TEM with a high-resolution quadrupole mass spectrometer (in situ TEM-QMS) at Shimane University.

To examine the dynamic behavior of the microstructure evolution under ion irradiation and post-annealing, in situ observations were carried out with a TEM (JEOL, JEM-2010) equipped with a low-energy ion gun [3]. The desorption behavior of deuterium (D) and He were simultaneously examined under post-annealing using a high-resolution QMS (MKS-Microvision 2) newly installed to the TEM for the present study.

The in-situ TEM-QMS enabled separation and quantitative analysis of D_2 (4.0282 amu) and He (4.0026 amu) released from a *f* 3 mm sample with irradiated area of ϕ 2 mm and simultaneous observation of microstructural evolution under annealing. For tungsten, it was shown that helium pre-irradiation causes a significant increase in deuterium retention and that most of the deuterium is desorbed without any apparent change in the bubbles up to an annealing temperature of ~800 K. Furthermore, for beryllium, significant differences were found between the behaviour of deuterium and helium bubbles. While deuterium bubbles gradually shrunk and disappeared at around 800 K, helium bubbles instantly vanished from the sample surface with bubble migration at a relatively high temperature of ~1000 K.

In the conference, quantitative data on deuterium and helium retention in tungsten and beryllium will be also presented and discussed. The results clearly indicate the direct impact of helium bubbles on the deuterium retention behavior in these materials.

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[2] M. Ialovega et al., Phys. Scripta 2020, 014066 (2020)
[3] M. Miyamoto et al., Phys. Scr. 2014 (2014) 014028

Ph.D. Engineering, Kyushu University, 2004.
M.Eng. Kyushu University, 2001.
B. Sc. Physics, Kyushu University, 1999.
Dr Miyamoto is a full professor of Department of Material Science, Shimane University. His research topics are in the field of plasma-material interactions in future fusion reactors. In particular, he is studying the surface modification of divertor materials exposed to high-density mixed plasma and the tritium inventory (storage) therein from the viewpoint of microstructure.

Combined Irradiation Method for In-situ and Ex-situ Study of Radiation Defect Evolution in Alloys

Dongyue CHEN
Tsinghua University

Thanks to the rapid development of computing technology, the modeling of radiation defect evolution has largely progressed in the recent years. On the other hand, the multi-scale modeling results lack enough experiment data for comparison, especially the one with time resolution. In-situ irradiation is a powerful tool to provide data directly comparable to modeling results, and has been developed for decades. The utilization of combined irradiation method in in-situ irradiation would further extend the capability and flexibility of this irradiation tool. This talk presents our efforts to establish and improve the in-situ and ex-situ irradiation facility and the contribution of the combined irradiation method. Using stainless steels and reactor pressure vessel steels as examples, the growing process of dislocation loops was tracked under low and high doses as well as different dost rates. The evolution of dislocation loops was interpreted to reveal the behavior of point defects, and the influence of irradiation and material parameters on point defect behavior was investigated. The current weakness and the future development of the combined irradiation method was discussed.

Dongyue CHEN, Associate professor, School of Material Science and Engineering, Tsinghua University.
Previously in charge of the HIT-TEM in-situ irradiation facility in Tokai-campus, The University of Tokyo. Enrolled in Tsinghua University in August 2023.

TEM characterization of dislocation loops in post-irradiated and in-situ irradiated pure Cr

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The application of Cr-based coating onto Zircaloy fuel rods has been widely proposed to enhance accident tolerance in water-cooled reactors. Compared to the knowledge that existed in the Zircaloy matrix, the response of Cr against irradiation has been less studied. The purpose of this study is to supplement the fundamental knowledge of radiation damage in Cr, for achieving a better understanding of the in-pile degradation behaviors of Cr-coated Zircaloy fuel rods.

Pure-Cr specimens were subjected to 2.8 MeV Fe ions irradiation at the temperature of 550 ℃ up to 15 dpa. To analyze the migration of dislocation loops, an in-situ irradiation was carried out in a post-irradiated TEM thin foil specimen at 300 ℃.

An improved inside-outside method through combining the consideration of edge-on view and inside-outside analysis for determining the habit plane was proposed. TEM observation showed that the distribution of dislocation loops extends the SRIM simulated irradiation depth to several times, which suggests the high mobility of interstitials in Cr. The dislocation loops were determined including both interstitial and vacancy nature types with ½<111> Burgers vector in all regions. The ratio of the interstitial loops to vacancy loops varied with irradiation dose and the depth in the specimens. An interesting finding is the nested loops, either vacancy-type loops called 'trough' or interstitial-type loops called 'island', which are composed of a large dislocation loop containing multiple small and coplanar loops with inverse nature. In-situ irradiation results indicate that the nested trough and island dislocation loops stem from the 1D migration of dislocation loops. This loop formation mechanism is an alternative mechanism of recombination and annihilation of point defects, which in turn can explain the low swelling rate (0.03-0.04%/dpa) in Cr.

Lijuan Cui, et al., *Acta Materialia*, 2024, 267, 119700;

Lijuan Cui, et al., *Journal of Nuclear Materials* 569 (2022) 153920

Lijuan CUI, is an Associate scientist in Sichuan University since October 2023. She has been working in the field of nuclear materials and science technology since 2013 as a master student in the university of science and technology Beijing with the guidance of Prof. Wan Farong. After that she went to Paul Sherrer Institut as a PhD. Student under Dr. Yong Dai working on "APT/TEM characterization of irradiation-induced segregation in ferritic/martensitic steels after irradiation in SINQ Target". In August 2021 she joined in Abe Hiroaki-lab in the University of Tokyo as a project researcher and studied the irradiation damage effect in chromium and chromium alloys in the application of ATF cladding coatings. Based on her studies, she gained experience on neutron irradiation, ion irradiation and in-situ ion irradiation and published 20 papers in Acta Materialia, Scripta Materialia and Journal of nuclear materials and 7 of them being the first author. In April 2024, she won the Young Scientist Award of Atomic Energy Society of Japan, Materials Science & Technology Division (AESJ-MSTD). Her APT work was chosen to be the most representative work in 2020 by CAMECA and posted on CAMECA Calendar Image Gallery of January. She got the Best Ph.D. presentation award in the E-MRS-2019 fall conference.

H/He mixed beam induced microstructure and nano-hardness evolution in tungsten

Minghuan Cui, Xuexin Ren, Jianlong Chai, Linqi Zhang, Peng Jin, Jing Li, Tielong Shen, Zhiguang Wang
Institute of Modern Physics, Chinese Academy of Sciences; University of Chinese Academy of Sciences; Advanced Energy Science and Technology Guangdong Laboratory

Plasma-facing component in fusion reactor will work under harsh service environment conditions, which will affect the microstructure and the safe operation of the device. Due to the advantages of high melting point, high thermal conductivity, low sputtering rate and low activation, tungsten is one of the important candidate materials for plasma-facing material and is expected to be fully used in future magnetically constrained fusion reactors. Therefore, the influence of irradiation or plasma bombardment on tungsten's structure and properties has been widely studied. However, investigation under a mixed H/He beam is not sufficient. In this study, implantation experiments of a H/He mixed beam irradiation were conducted in pure tungsten, and the irradiation induced hardness change and microstructures were measured by nano-indentation and transmission electron microscopy. At all irradiation conditions, ion irradiation always causes hardening. With increasing fluences, the hardness values show a decrease trend. TEM observations show that irradiation damage leads to the generation of bubbles and dislocation loops. With the increase of fluence, the bubble's size first increases and then decreases, and the density first increases slightly and then remains unchanged. The size of the dislocation loops increases with fluences, and its density decreases first and then remains unchanged. An analysis between the micro-defects and hardness value suggests that hardness increment caused both by the bubble and the dislocation loop but the hardness increase caused by the dislocation loop is much larger than the bubble.

Microstructural characterization of irradiated ferritic/martensitic steels after irradiation at Swiss Spallation Neutron Source

Yong Dai, Peng Bi, Lijuan Cui
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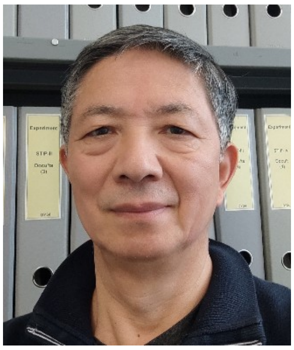
Ferritic/martensitic (FM) steels are the one of major classes of materials included in the SINQ Target Irradiation Program performed at the Paul Scherrer Institute. A large matrix of FM steels from various international nuclear materials programs were irradiated at SINQ in a wide range of irradiation dose (5-30 dpa) and temperature (80-600 °C). Owing to high energy protons producing helium and hydrogen at high rates (up to 80 appm He and 400 appm H per dpa), the irradiation-induced microstructural and mechanical changes in the FM steels are significantly different from that of fission neutron irradiations.
Extensive microstructural characterization of these irradiated FM steels has been conducted by employing transmission electron microscopy (TEM), atom probe tomography (APT) and positron annihilation spectroscopy (PAS) analyses. Irradiation-induced microstructural features such as defect clusters/dislocation loops, cavities/helium bubbles and precipitates of alloying-elements and transmutation products have been studied. In this contribution the results of the microstructural investigations obtained from FM steels irradiated to 5-20 dpa are reviewed.



Yong Dai is a senior scientist at the Nuclear Materials Laboratory of the Paul Scherer Institute (PSI) in Switzerland. He received his Ph.D. at the Swiss Federal Institute of Technology (EPFL) in Lausanne in 1995. He has been working at PSI since 1995, leading a team studying materials for spallation neutron sources and fusion/fission reactors applications. He is responsible for the irradiation experiments at the Swiss Spallation Neutron Source (SINQ), in which some 20 international institutions have participated. His research work focuses on the effects of radiation damage, helium and hydrogen, and liquid metals on various structural materials and pure metals. From 2016 to 2022, he served as associate editor of Journal of Nuclear Materials.

Void evolution mechanisms under in situ TEM heavy-ion irradiation environment

FAN Cuncai
Department of Mechanical Engineering, City University of Hong Kong

Understanding the void evolution in radiation environment is of great interest for material performance in nuclear energy systems at elevated temperature and high radiation dose. Unlike dislocation loops that have been extensively studied by in situ transmission electron microscope (TEM) irradiation, the voids have rarely been reported, presumably due to the free surface effect. To deal with this issue, we deliberately introduced into single-crystalline Cu (110) specimen the faceted voids by using magnetron sputtering deposition technique. We then conducted systematic in situ TEM irradiations on these voids with a single beam of 1 MeV Kr++ and a dual beam of 1 MeV Kr++ and 14 keV He+. Our studies revealed intriguing morphological evolutions of void spheroidization, shrinkage and migration, as well as the influence of helium on the kinetics of void/bubble growth under heavy ion irradiation environment at low and high temperatures. The experimental results were discussed within the framework of a proposed critical bubble model and bubble coarsening model. The current work might be extended to investigate the void swelling behavior of other advanced nuclear materials.

Dr. FAN Cuncai is now an Assistant Professor (since 12/2022) at Department of Mechanical Engineering, City University of Hong Kong, Hong Kong China. He obtained his PhD from School of Materials, Purdue University in 2019. Before joining CityU, he was working as a postdoctoral research associate at Oak Ridge National Laboratory and The University of Hong Kong. His research areas include nanometals, radiation damage effects and helium effects in nuclear fusion materials.

Nature of Nano-scale Defect Clusters Under Irradiation: Computer Simulations versa Experimental Observations

Fei Gao[1], Angel Chavira[1], Long Guo[2], Huiqiu Deng[2], Kan Ma[3]
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Irradiation in metals produces a variety of point defects and nano-scale clusters, including voids, precipitates and dislocation loops, and results in macroscopic void swelling, radiation-induced hardening, and embrittlement. Understanding the nature of nano-scale defect clusters and their long-time evolution with microstructures is crucial for unraveling the mechanisms of radiation damage in fission and fusion reactor environments. MD method is widely used for simulating defect migration and defect interaction, but it is unavailable for the long-time defect evolution due to its limited time-scale, while kinetic Monte Carlo (KMC) requires prior knowledge of defect properties with no atomic detail. Here, several recently developed simulation methods, including Self-adaptive Accelerated Molecular Dynamics (SSAMD), kinetic Activation Relaxation Technique (k-ART) and Object Probability-based Long-time Dynamics (OPLD) method, will be reviewed, and their possible applications to the fundamental rate-limiting microscopic processes and the corresponding mechanisms governing the macroscopic properties will be highlighted. Molecular dynamics simulations are used to understand the detailed mechanisms of loop formation and elucidate a distinct correlation of loop nature to the stacking fault energy, as explored by high-resolution scanning transmission electron microscopy in fundamental fcc metals Al, Ni and Cu. In addition, we have used SSAMD to extend timescale to study the diffusion of <100> interstitial dislocation loops in Fe, which reveals a new diffusion mechanism that is verified by *in-situ* TEM measurements. Finally, our OPLD has been employed to simulate the defect accumulation and microstructure evolution of electron irradiated Fe at dose rates comparable with experimental values. The microstructure features investigated by traditional MD and OPLD up to 0.1 DPA are compared, and the dose rate accounts for significant differences in Frenkel pair accumulation, clustering, and dislocation density. A new mechanism of forming <100> dislocation loops, namely pseudo Burgers vector conservation of loop reaction, is proposed, which is significantly different from all previously proposed mechanisms.

Dr Fei Gao is a full professor of the Department of Nuclear Engineering and Radiological Sciences at University of Michigan. Prof. Gao's current work mainly focuses on ion-solid interaction, irradiation damage, detector materials, nanostructures properties, Li^+ ion battery, multi-scale modeling of materials behavior under extreme condition. Prof. Gao has published more than 350 refereed journal papers, including *PRL, Nature Communication, Energy & Environmental Science, Nano letter, ACS Nano, PNAS, Advanced Materials, Angewandte Chemie International Edition, Acta Materialia, PRB, APL, JAP, etc.*He has been an associated editor of *AIP (American Institute Physics)*

Advances, editorial board of Scientific Reports and a guest editor for a number of journals. He is a member of international advisory committee for international conference on Computer Simulations of Radiation Effects in Solids, one of the program committee members for international conference on hard x-rays, γ-rays and neutron detector physics, and fellow of international association of advanced materials.

Coupled effect of Cr and Al on interactions between a prismatic interstitial dislocation loop and an edge dislocation line in Fe-Cr-Al alloy

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Segregation of alloying elements to a prismatic dislocation loop under irradiation is an important phenomenon in understanding the role of loops in radiation effects in alloys. In this presentation, various segregation sequences of Cr and Al atoms to a <100> and a 1/2<111> prismatic dislocation loop in a Fe-Cr-Al alloy explored using molecular dynamics method and ab initio energy calculation would be presented. The energy calculation results reveal that Cr can readily segregate to the loops while Al atoms cannot by themselves, but the segregated Cr atoms are able to promote Al segregation at the loops, defined as a coupled segregation of Cr and Al, as shown in Fig. 1.

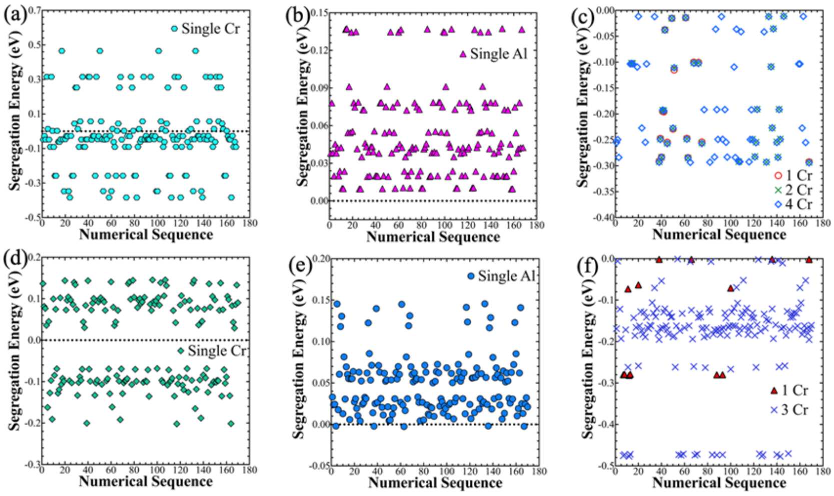

Fig.1 Segregation energy of a Cr atom and an Al atom to a <100> loop and a 1/2<111> loop.

After interacting with Cr and Al, a previously mobile 1/2<111> loop becomes sessile. Al segregation through the coupled segregation process can further affect the pinning behavior of a dislocation loop to an edge dislocation motion, as shown in Fig.2. By varying the Al fraction in the segregated solutes, a nonlinear dependence of the pinning effect on dislocation motion is determined. All these results suggest that, in addition to the effect of Al on the surface resistance to oxidation, Al effect on the dislocation loop behavior is important to optimize the composition of Fe-Cr-Al alloys for their applications in accident tolerant fuel concepts.

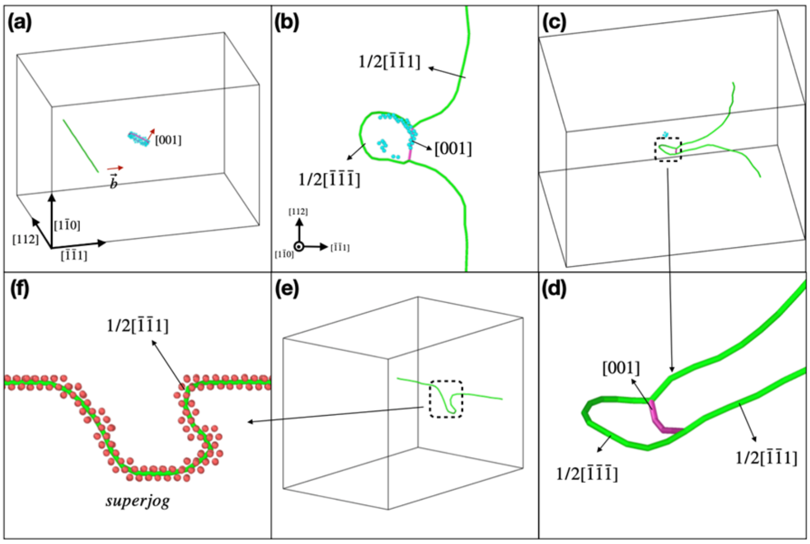

Fig.2 Unpinning process of a $1/2[1\bar{1}\bar{1}]$ edge dislocation from a [001] loop segregated with 71 Cr atoms through the formation of a superjog in a binary Fe-Cr system under a 400 MPa shear stress.

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Assistant researcher, Paul Scherrer Institute, 2007-2011

Utilizing Cluster Dynamics Modeling to Interpret TEM Observations of In-situ and Ex-situ Irradiation

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The effects of irradiation damage in nuclear reactor materials encompass a dynamic process involving diverse microstructural mechanisms across multiple temporal and spatial scales. Integrating multi-scale simulations with experimental approaches offers an effective means to delve into the mechanisms underlying material irradiation damage and predict its performance in service. Among these approaches, cluster dynamics modeling, grounded in diffusion-reaction rate theory, serves as a vital conduit linking atomic-scale and macroscopic-scale models. This modeling relies on fundamental defect properties garnered from first-principles calculations or molecular dynamics simulations as inputs. Employing mean-field approximations, it simulates the dynamic evolution of irradiation defects across extensive temporal and spatial scales while providing invaluable inputs such as defect distributions for higher-scale simulations, thereby facilitating the prediction of material macroscopic performance.

This report focuses on the application of cluster dynamics modeling in interpreting TEM observations of in-situ and ex-situ irradiation. It commences with an introduction to the construction of cluster dynamics models, followed by an analysis of the selection of key input parameters and numerical solution methods. Subsequently, it explores the applications of cluster dynamics models in two cases. Firstly, it investigates the evolution of visible defect clusters in nanometer-thick molybdenum foils through in-situ TEM irradiation experiments and coordinated cluster dynamics modeling. The in-situ TEM observation of molybdenum foil was conducted under 1 MeV krypton ion irradiation at 80℃ to various radiation damage levels, performed using the IVEM. The optimized model demonstrates good qualitative and quantitative agreement with the in-situ TEM experiments. Secondly, it examines helium-defect evolution in helium-implanted iron. Thermal helium desorption spectra (THDS) were effectively reproduced by employing cluster dynamics modeling alongside the underlying helium-defect cluster distribution at different temperatures. This integration of THDS and cluster dynamics modeling aids in identifying the primary defect evolution contributing to observed helium desorption peaks.

Lastly, the report scrutinizes the limitations of cluster dynamics modeling and discusses its advantages in interpreting experimental observations of in-situ and ex-situ irradiation.

Dr. Xunxiang Hu is a professor in the College of Physics at Sichuan University. Prior to his current position, he served as a staff scientist at the Materials Science and Technology Division of Oak Ridge National Laboratory from 2014 to 2021. He earned his Ph.D. from the University of California, Berkeley (2013), his M.S. from Tsinghua University (2009), and his B.S. from Shanghai Jiao Tong University (2007).With a profound expertise in high temperature moderator materials and neutron irradiation effects of nuclear materials, Dr. Hu's research is pivotal in extending the lifetime and enhancing the accident tolerance of current nuclear fission reactors, designing advanced nuclear reactors, and realizing future fusion reactors. He held key roles in various R&D programs of the US Department of Energy, including TCR Program, ATF

Program, Fusion Materials Science Program, and Fusion Blanket/Fuel Cycle Program. Currently, Dr. Hu leads a National Natural Science Foundation of China - “Qisun Ye” Science Foundation Program, in addition to two thrusts of the National Key Research and Development Project of China. He also serves as an Editorial Board Member of the esteemed Journal of Nuclear Materials. He was nominated for the 2021 Early Career Award of the US DOE, selected for the Overseas High-level Talent Recruitment Program in 2018, and honored with the Eugene P. Wigner Fellowship in 2014. He was further recognized as a JNM Rising Star in Nuclear Materials in 2022 and awarded Outstanding Research Scientist of Sichuan University in 2023.

Superior hydrogen permeation resistance via Ni–graphene nanocomposites: Insights from atomistic simulations

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[1]Key Laboratory of Material Physics of Ministry of Education, School of Physics and Microelectronics, Zhengzhou University, Zhengzhou 450001, China.
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Designing hydrogen-resistant Ni-based alloys from the perspective of the Ni/graphene interface (NGI) provides the potential to increase hydrogen trapping away from potential fracture paths. Nonetheless, numerous essential mechanisms of hydrogen penetration behaviors in the Ni-graphene nanocomposites (NGNC) are presently not well understood. Here we investigate the influence of Ni/graphene interfaces (NGIs) on the behavior of hydrogen diffusion and trapping in their vicinity using atomistic simulations. The thermodynamic results showed that as the temperature increases, the individual diffusion distance of solute H exhibits an increase, while its total diffusion distance and frequency experience a decrease. When the temperature exceeds 900 K, the solute H ceases to exhibit intermittent jumps after a certain duration and is instead trapped by the NGI. This circumstance leads to a gradual transition of the diffusion mechanism from two-dimensional to one-dimensional over time, manifested specifically as diffusion along the [111] direction. There is a noticeable difference in the MSD for solute H diffusion between NGNC and pure Ni. This difference can be substantial, spanning two orders of magnitude, highlighting the hindered rate of solute H diffusion in the NGNC. The presence of NGI in the composites leads to reduced hydrogen diffusivity in their vicinity compared to single-crystal Ni. This effect is particularly pronounced at elevated temperatures, leading to slower hydrogen diffusion, and consequently facilitating its entrapment in the interfacial regions. Energetic and kinetic calculations have been employed to comprehend the dynamic behaviors of solute H further. The segregation energy of solute H near the NGI demonstrates a propensity to rise and subsequently decline, peaking at 2.484 eV. Notably, the minimal segregation energy near the interface is comparable to the maximum value near the Ni grain boundaries. Discerning the disparity in formation energy clarifies the existence of three distinct and stable sites for H atoms within the core of NGI. At the S_{III} sites, solute H readily forms a sp^3 C−H bond with the C atoms on the surface of graphene as the distance between C−H falls within the typical bond length range. This bond formation produces significant heat release in the system and causes local configuration changes in graphene. The robust covalent bond energy guarantees the utmost stability of the soluble H at the S_{III} sites, thereby impeding its detachment from graphene and subsequent entry into a non-diffusible state. Upon approaching the NGI, the diffusion barrier gradually diminishes to 0.122 eV, affirming the solute H's distinct inclination to migrate toward the interface within a specified NGI range. The NGI has the potential to provide two distinct migration paths for solute H. Nonetheless, the practical activation likelihood for both paths is extremely minimal. The present study advances

understanding of the behaviors of hydrogen permeation in NGNC. Achieving an appropriate dispersion of graphene within the matrix can considerably reduce the diffusion coefficient of solute H in the material. This, therefore, guides the active H atoms towards non-diffusible H atoms. Such a process can successfully provide an efficient strategy to impede hydrogen barrier. This approach may enable the successful avoidance of hydrogen embrittlement failures in high-strength Ni-based alloys, ultimately ensuring the reliability of nuclear reactor operation.

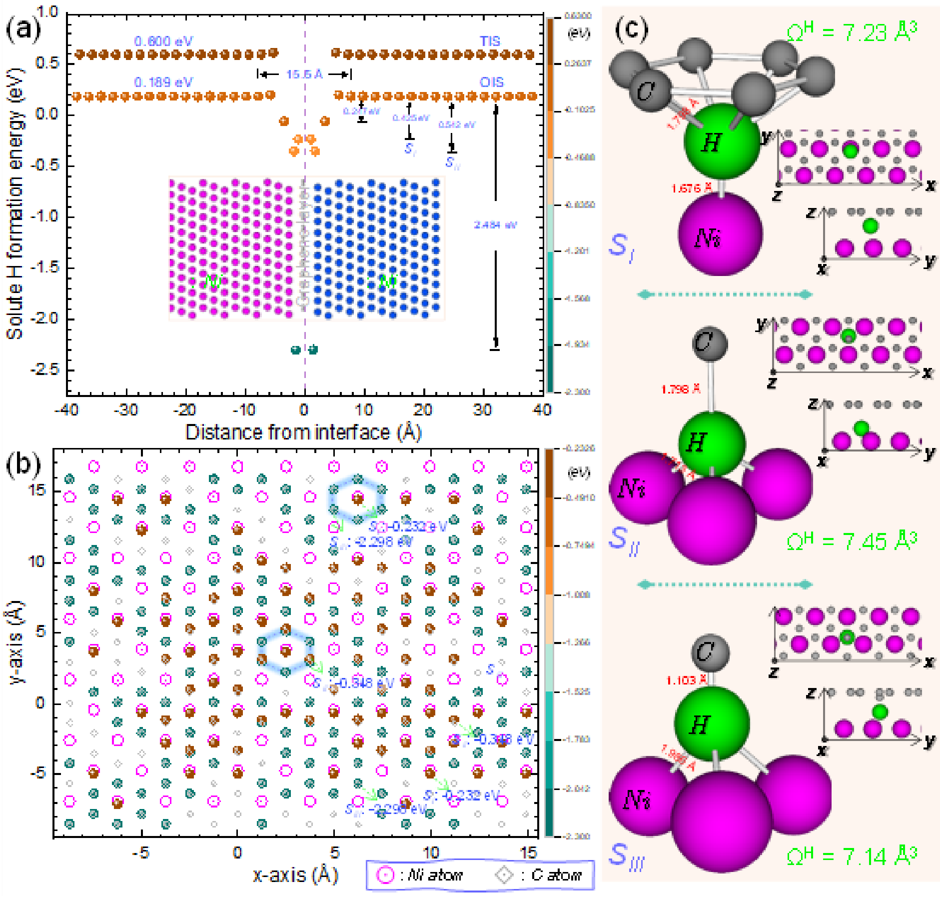

Fig. 1. Distributions of the formation energies of solute H near the NGI.

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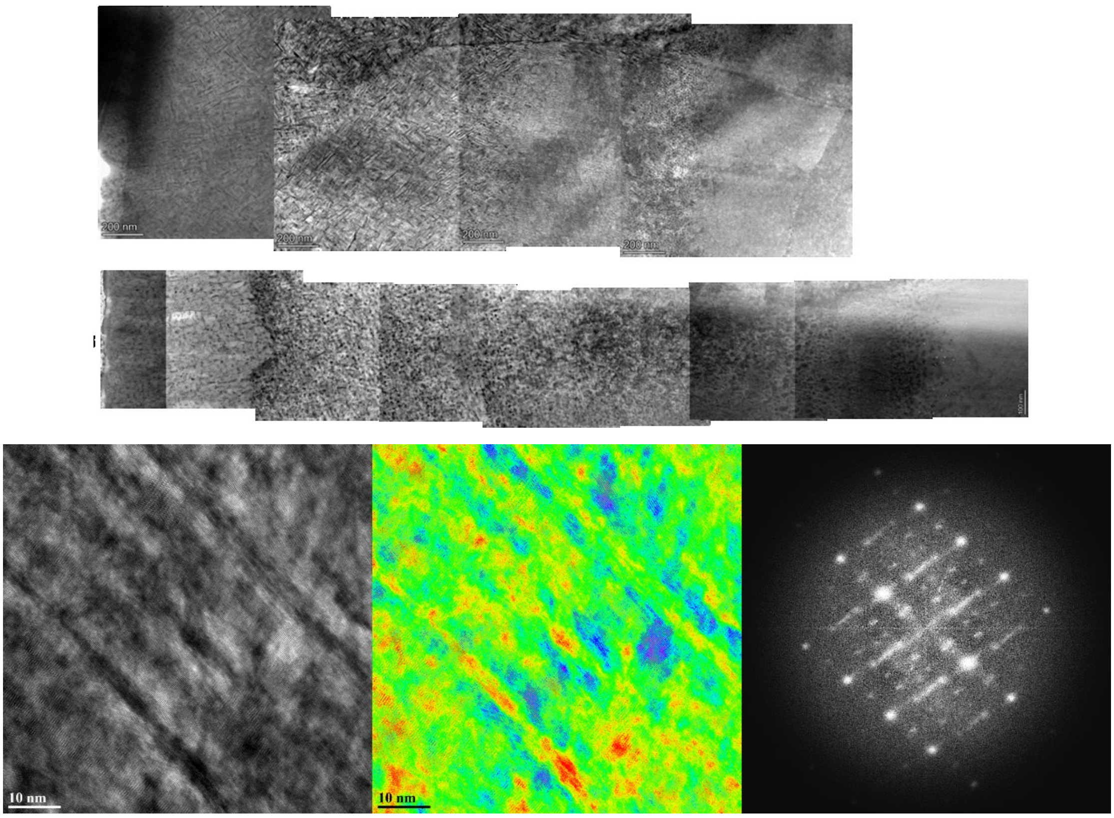
Research Interests: Radiation effects of materials

Precipitation behavior of Mo-Re alloy under ion irradiation with 20 MeV nickel ions at a dose level of 140 dpa at 580°C

Hongtao Huang
Sun Yat-sen University

Molybdenum-rhenium (Mo-Re) alloy is one of the crucial candidate structural materials for space nuclear reactors. Therefore, the stability of Mo-Re alloys under irradiation conditions has long been a subject of concern. Over the past few decades, radiation-induced segregation and precipitation have been discovered in Mo-Re alloys, yet researchers have not reached a unified understanding of the types and structures of the precipitated phases. In this study, Mo-14Re and Mo-42Re alloys were irradiated with 20 MeV nickel ions at a dose level of 140 dpa at 580°C. Re-riched precipitates were observed within the depth range of 0 to 3 micrometers in both alloy compositions. These precipitates exhibited two morphologies: elongated needle and disc-shaped structure. The disc-shaped precipitates were identified as the χ phase, characterised by a complex bcc lattice with lattice parameters three times those of the Mo lattice. Previous research on the crystall structure of needle-like precipitates has been contentious, with some suggesting a solid solution of rhenium in the hcp structure, while others proposing the presence of the χ phase. The needle-like precipitates are more likely the χ phase according to this study.



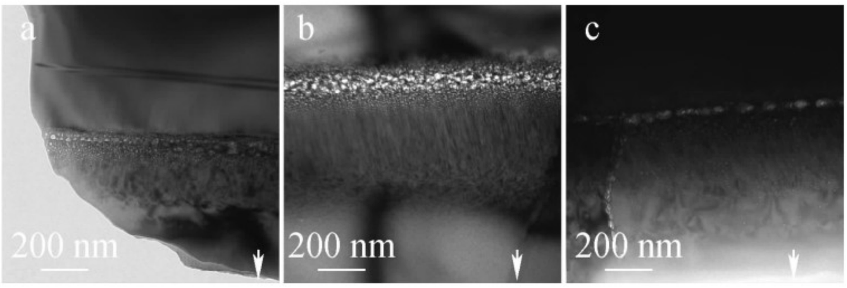
I received my PhD in engineering from Tsinghua University in July 2013. From July 2013 to December 2019, I worked at China Institute of Atomic Energy, serving successively as assistant researcher, associate researcher and group leader. In January 2020, I was introduced to the Sino-French Institute of Nuclear Engineering and Technology of Sun Yat-sen University. I am currently an associate professor and doctoral supervisor at Sun Yat-sen University. I have published more than 30 SCI papers in journals such as Journal of Nuclear Materials, Fusion Engineering and Design, Materials Science and Engineering A, etc.

Irradiation damage in sintered SiC irradiated by He ions

Bingsheng Li
Southwest University of Science and Technology

We reported He ion irradiation of sintered SiC. He ions supplied by the 320 kV Multi-disciplinary Research Platform for Highly Charged Ions of the Institute of Modern Physics, Chinese Academy of Sciences. The irradiations were performed at room temperature (RT) to 1000℃, to fluences from 10^{15} to 10^{17} ions/cm^2. Post irradiation, the sample was annealed at different temperatures. A combination of transmission electron microscopy, Raman spectroscopy, Nano-indenter and thermal conductivity were used to characterize irradiation damage. The threshold fluence for helium bubble formation in the sintered SiC is lower than that in single crystal SiC. What's more, morphology and size of helium bubbles depend on irradiation temperature and annealing temperature. We also investigated irradiation-induced defects and elemental segregation. Lots of defect clusters can be formed by He ion irradiation. Meanwhile, the growth of helium bubbles can emit self-interstitials, resulting in the formation of interstitial-type dislocations. Carbon enrichment occurred on grain boundaries. Irradiation hardening increased initially, and decreased finally with irradiation dose. The thermal conductivity of sintered SiC decreased after He ion irradiation.

Keywords: Sintered SiC; He ion irradiation; Microstructural defect; Helium bubble



a——1 000 ℃,30 min; b——1 200 ℃,30 min;
c——1 500 ℃,30 min

Fig. 1. 230 keV He irradiated sintered SiC at RT and annealing at different temperatures for 30 min.

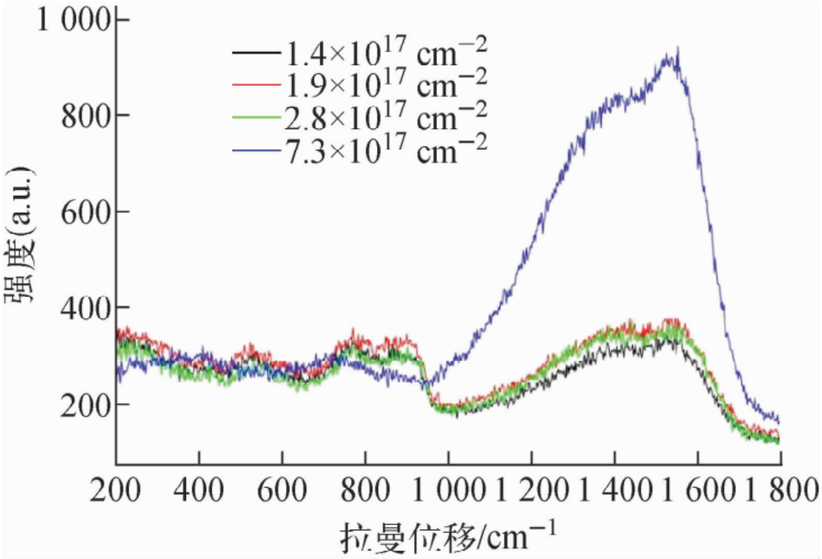


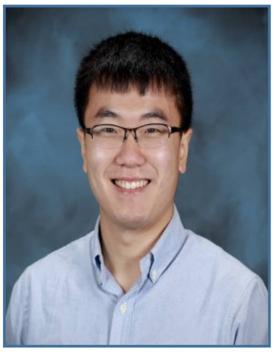
Fig. 2. Raman spectroscopy of He irradiated sintered SiC to different fluences at RT

Bingsheng Li, Professor, Southwest University of Science and Technology, P.R. China. I studied modern physics in the University of Lanzhou and did a PhD in particle physics and nuclear physics with Dr. Chonghong Zhang at IMP-CAS. The field of I research lies in the investigation of the evolution of microstructure and properties of nuclear structural materials, such as silicon carbide and ferritic/martensitic steels induced by ion-irradiation.

Theoretical investigation of positron annihilation spectroscopy of neutron irradiated CrMnFeNi high entropy alloy

Junyi Fan, Congyi Li
Shanghai Jiao Tong University

Short range order (SRO) is critically important for the mechanical performance of high entropy alloy (HEA). Recent studies showed that SRO may alter the migration energetics of defects, thus improving the radiation resistance of HEA. However, nano-scale SRO is extremely challenging to characterize. The evolution of SRO under irradiation, especially under neutron irradiation, is still not well understood. In this work, positron observable calculation based on two-component density functional theory (TCDFT) is utilized in combination of experimental positron annihilation spectroscopy (PAS) to investigate the evolution of SRO and defect microstructure after neutron irradiation and post-irradiation examination. First, the computed PAS of pure elements in FeCrNiMn are benchmarked with experimental measurement to verify the accuracy of TCDFT method. After that, the calculated FeCrNiMn PAS, including annihilation lifetimes and CDB spectra, are compared with experimental measurement to investigate bulk and defect microstructure of neutron irradiated FeCrNiMn. The calculated annihilation lifetime of vacancy-type defects revealed the variation of vacancy cluster size during post-irradiation annealing. The computed CDB spectrum, on the other hand, revealed the chemical identity of SRO. Our preliminary calculation identifies the existence of SRO in the bulk structure. Also, our results showed that Mn may play a unique role in the SRO of the bulk structure and the SRO near radiation defects of FeCrNiMn HEA.



Congyi Li is an assistant professor from Shanghai Jiao Tong University. He obtained his PhD from University of Tennessee, Knoxville supervised by Prof. Steve Zinkle. His current research focus on degradation mechanism of materials for nuclear technology applications, including advanced nuclear fuel, reactor structural component and neutron radiation detector.

In Situ and *Ex Situ* TEM Study of Microstructural Evolution in Hollandite Ceramics under Ion Irradiation

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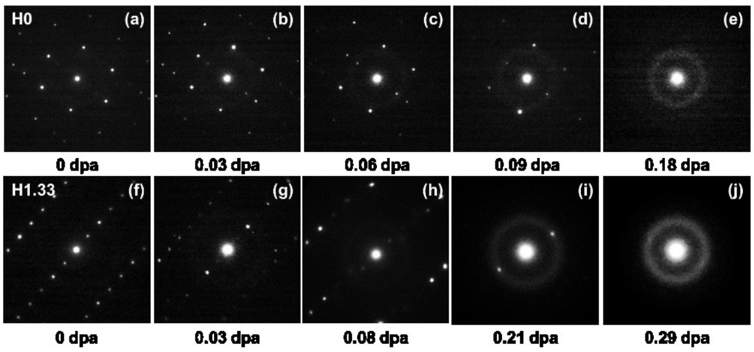
The ceramic materials with the structure of hollandite mineral have been proposed to serve as the nuclear waste form to host radionuclides (e.g., Cs) in the high-level nuclear waste (HLW) due to high waste loading capacity and excellent chemical durability. The waste form has demonstrated high structural stability under ionizing radiation from 200 and 300 kV electron beam irradiation. The purpose of this work is to study the effects of displacement damage that may be induced by alpha-decay recoils of the transuranic elements (TRU) in the Ba- and Cs-end members of the Ba/Cs-Fe/Ti hollandite waste form with *in situ* TEM analysis to see if the material is also suitable for hosting TRUs. Pre-irradiation characterization was conducted with atomic resolution TEM, combined with elemental mapping to establish a baseline understanding of the materials' microstructure and composition. 1.2 MeV Kr^{3+} ion irradiation was conducted at 200°C for the study using a TEM-ion beam linked facility at the Michigan Ion Beam Laboratory (MIBL). The two samples tested have the chemical compositions of $Ba_{1.33}Fe_{1.33}Ti_{6.67}O_{16}$ (H0) and $Cs_{1.33}Fe_{1.33}Ti_{6.67}O_{16}$ (H1.33), respectively. Figure 1 compares the selected area electron diffraction patterns (SADP) of the two specimens obtained by in situ TEM during the ion beam irradiation. The critical amorphization dose, when the Bragg spots completely disappeared in the diffraction pattern, are 0.176 and 0.289 dpa, for H0 and H1.33 respectively. The curves of accumulation of amorphous fraction with increasing irradiation dose for the two hollandite compositions are plotted in Figure 2. The Cs-end member of hollandite took 60% higher dose than the Ba-end member to become fully amorphous. Supplementary to the *in situ* TEM studies, *ex situ* irradiation experiments were carried out using 8 MeV Fe^{3+} ions on bulk samples at the room temperature for both Ba- and Cs-hollandite, followed by cross-sectional TEM imaging to elucidate depth-dependent radiation effects. These investigations provided key insights into the microstructural evolution, and phase stability for the hollandite matrix under different irradiation dosage.



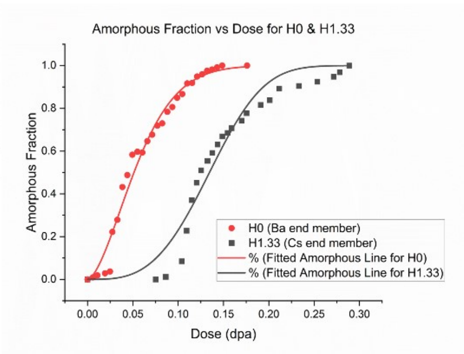


Fig. 1 Progressive amorphization process of Ba (H0) and Cs (H11.33) end members of hollandite composition irradiated with 1.2 MeV Kr^{3+} ions at 200ºC.
Fig. 2 Amorphous fraction vs. dose curves for the two hollandite samples under the Kr^{3+} ion irradiation.

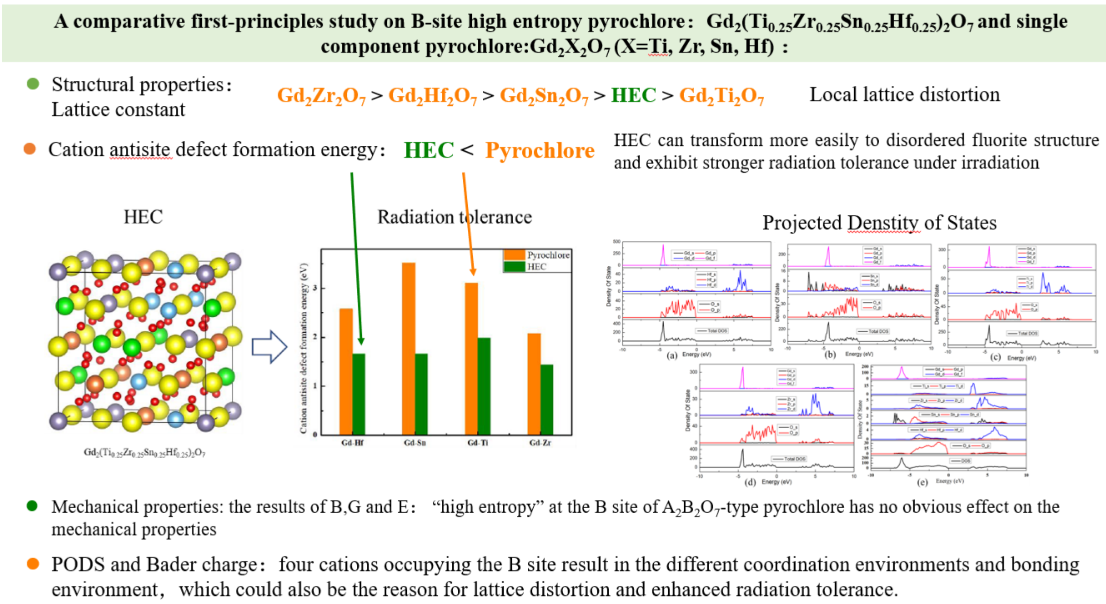
Yuhan Li is currently a Ph.D. student in the Department of Materials Science and Engineering, University of Michigan-Ann Arbor. She has master's degrees in Materials Science and Engineering, as well as in Industrial and Operation Engineering from the University of Michigan (2021). She obtained her B. Sc. Degree in Chemistry (minor in Mathematics) from Pennsylvania State University (2020). Her research interests have been in TEM study of irradiation effects in materials.

This work was supported as part of the Center for Hierarchical Waste Form Materials (CHWM), an Energy Frontier Research Center funded by the U.S. Department of Energy, Office of Science, Basic Energy Sciences under Award No. DE-SC0016574.

Physical properties and radiation tolerance of high-entropy pyrochlores $Gd_2(Ti_{0.25}Zr_{0.25}Sn_{0.25}Hf_{0.25})_2O_7$ and individual pyrochlores $Gd_2X_2O_7$ (X= Ti, Zr, Sn, Hf) from first principles calculations

Chenguang Liu
Yantai University

The physical properties of high-entropy ceramics (HEC) of $Gd_2(Ti_{0.25}Zr_{0.25}Sn_{0.25}Hf_{0.25})_2O_7$ and $Gd_2X_2O_7$ (X = Ti, Zr, Sn, Hf) are investigated using first principles method. The structural properties, anti-site defect energetics, mechanical properties and electronic properties are compared. The lattice constant of HEC is within the range of its constituent pyrochlores, and the local lattice distortion presents in HEC. The cation anti-site defect is much easier to form in HEC than in pyrochlore, which implies that HEC can easily transfer to disordered fluorite structure and exhibit stronger radiation tolerance. The "high entropy" at B site of $A_2B_2O_7$-type pyrochlore had no obvious effect on mechanical properties. The electronic properties of HEC and pyrochlore are characterized by projected density of states distribution and Bader charge, and the causes of HEC lattice distortion are inverstigated.



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Synergistic effects between He bubbles and Mg in the silicon carbide materials

Min Liu, Qiqi Li, Xiaoyue Li
Sun Yat-sen University

Silicon carbide (SiC) has been considered as a promising candidate material used as the structural materials for nuclear reactors. He atoms will be generated in SiC through (n,α) reaction, which can be trapped by vacancies to form bubbles. In addition, metallic transmutant Mg will also be introduced. In this talking, the evolution of He bubbles in CVD-SiC, the diffusion behaviors of Mg in SiC and their synergistic effect will be presented.
The studys indicate that helium ion irradiation can cause SiC to form helium bubbles and dislocation loops, leading to the irradiation swelling and irradiation hardening. Helium bubbles can accelerate the recrystallization of the amorphous layer and thus inhibit the diffusion of Mg in the SiCmaterial. Meanwhile, Mg can increase the diffusion energy barrier of He atoms and make them in a high-energy unstable state, which prompted He atoms to move to vacancies far away from Mg, and thus inhibited the diffusion of He atoms to the surface of SiC, resulting in the formation of helium bubbles in the internal region far away from Mg.

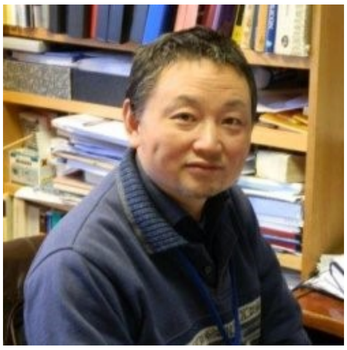
Min Liu, PhD and Associate Professor at Sino-French Institute of Nuclear Engineering and Technology, Sun Yat-Sen University. His research focus is the irradiation damage of nuclear materials under service environments. He has published over 20 SCI papers in the journals such as Scripta Mater, J Adv Ceram, Corros Sci, J Nucl Mater, J Eur Ceram Soc, Ceram Int, Mat Sci Eng A, and Mater Charact.

UK National Ion Beam Centre and Ion Beam for Novel Materials Research

Surrey Ion Beam Centre, ATI, FEPS, University of Surrey, Guildford GU2 7XH, UK

The UK National Ion Beam Centre is a national research facility supported by UKRI/EPSRC, providing ion beam services for researchers based both in UK and overseas, with laboratories located at Universities of Huddersfield, Manchester and Surrey.

At Surrey, surface modification of materials has been one of our major research areas since 1960s, such as ion implantation and analysis for semiconductors. More recently, activities on ion irradiation and analysis of other materials like ceramic superconductors and solid-state electrolytes (SSEs) are on the way up. In this presentation I will describe our research on in-situ He ion irradiation of ReBCO conductors, Xe ion irradiation for suppressing Li dendrite growth and elastic recoil detection (ERD) measurements of H and O concentrations in oxide thin films.



Nianhua Peng, former Semiconductor Physics major of Wuhan University in 1978, is currently the liaison officer of Ion Beam Center, University of Surrey, UK. He has long been engaged in the regulation and analysis of physical and chemical properties of solid materials under extreme conditions.

On the Study of Helium Bubble Evolution in Metal Tritide by Advanced Transmission Electron Microscopy

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Keywords: titanium tritide, helium bubble, growth mechanism, microscopic study, electron-energy loss spectrum

Tritium is the key nuclear fuel for nuclear fusion reaction and metal tritides are important functional materials in the field of nuclear fusion energy. The helium atoms generated by the decay of tritium atoms and their evolution seriously affect the macro performance of metal tritides. The formation and residence of helium bubbles will affect the tritium storage properties of metal tritides. The research on the growth mechanism of helium bubbles in metal tritides is an important scientific problem and attracts more attentions. Limited by the radioactivity of tritium and the lack of microanalysis technology for light element atoms such as tritium and helium, the study of helium bubble growth model has been in the stage of phenomenological theory for a long time. With the progress of electron microscopic analysis technology, a new transmission electron microscopic technology for more accurate analysis of helium bubble information is proposed, and a new breakthrough has been made in the research on the microscopic mechanism of helium bubble growth.

Understanding the helium bubble evolution is the key basis to solve the scientific problem in the practical application of metal tritides, which is of great significance to promote the development of science and technology of nuclear energy engineering. In recent years, the micro analysis technology has made remarkable progress. The focused ion beam microscope has solved the difficulty of preparing electron microscope samples of radioactive tritides [1]. The resolution of double spherical aberration transmission electron microscope is better than 60 pm, which helps to reveal the mechanism of helium atom migration and helium bubble growth at the atomic scale [2,3]. The three-dimensional reconstruction technology of transmission electron microscope provides the three-dimensional spatial number density and interval size distribution information of helium bubbles [4,5], which avoids the possibility that the two-dimensional plane information may mask the three-dimensional information of helium bubble growth. The resolution of electron energy loss spectrum has reached 0.1 eV, which solves the difficulty of analyzing the pressure in helium bubble and the number density of helium atoms in helium bubble [6-7]. In addition, the preparation technology of ultra-thin electron microscope specimens is necessary to obtain microscopic information at the atomic level. The progresses of low-energy ion thinning technology and electrochemical flash polishing [9] all increase the feasibility of studying the microscopic mechanism of helium bubble evolution. This report will focus on introducing the developed technology for the microscopic analysis of helium bubbles in metal tritide. Some of the major results on the helium bubble evolution in the metal tritide will also be summarized.

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Effect of post-treatments on radiation damage of additively manufactured 316L stainless steels

Yingxi Li, Miao Song*
Shanghai Jiao Tong University

The AISI 316L austenitic stainless steel (SS) fabricated by laser powder bed fusion presents significant potential for applications in light water reactors and liquid metal-cooled fast reactors. Radiation damage in additively manufactured (AM) 316L SS has been widely investigated, yet comprehension of the mechanisms behind radiation resistance in AM 316L SS are insufficient. Despite this, the understanding of its intrinsic microstructures, such as textures, dislocation cells, elemental segregations and oxide inclusions, and their impacts on radiation damage remains unclear. This study aims to investigate the thermal stability of the microstructure and its effect on the radiation behavior of AM 316L SS. Four different heat treatments, namely 650℃ / 1 h, 800℃ / 1 h, 950℃ / 1 h, 1200℃ / 2 h, and the as-built condition were applied to the radiation specimens. The as-built specimen and post-treatment specimens were irradiated by 5 MeV Fe2+ ions to a damage level of 50 displacements per atom (dpa) at 550℃. Microscopic characterization demonstrates a remarkably suppression of void swelling in full recrystallization (RE) AM 316L SS compared to other specimens. Additionally, it was observed that stripe-shaped clusters enriched with Ni-Si were present. The Ni3Si type precipitates were more likely to segregated along with dislocation loops in RE specimen but segregated along with voids in other specimens. These distinct responses indicate the radiation damage is sensitive to heat treatments and the isolation of structural factors help to shed light on the understanding of radiation damage in AM 316L SS.

Song Miao, Ph.D., graduated from Texas A&M University. He joined the Department of Nuclear Science and Engineering at Shanghai Jiao Tong University in 2021 as a tenured associate professor. His research focuses on irradiation damage and radiation effects, as well as the application of additive manufacturing materials in reactors. To date, he has published nearly 70 SCI papers in journals such as the Journal of Nuclear Materials and Acta Materialia, with over 1200 citations. He is the principal investigator of a National Natural Science Foundation of China. He also serves as a reviewer for the Journal of Nuclear Materials and Corrosion Science et. al.

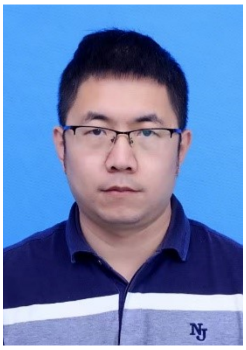
Computer simulation and modeling of radiation effects

Dan Sun, Yong Xin, Xun Lan, Yuanming Li, Xi Qiu, Jijun Zhao
Nuclear Power Institute of China; South China Normal University

The doped large-grain UO2, is considered as a promising nuclear fuel, owing to its excellent properties. It delays the release of fission gas by increasing grain size to lengthen the migration of bubble to grain boundaries. However, the irradiation defect behavior is different in UO2 doped with different oxides.

To understand the effect of different dopant on the defect behavior, we first compare the stable site of different foreign atoms (Cr, Al, Si) and calculate the change of bulk modulus, shear modulus, Young's modulus and Possion's ratio with dopant atoms. Then we systemically investigated the stability of point defects (vacancy and interstial atoms) as well as the structure and energetics of small defect clusters in UO2 doped with Cr, Al, Si. From energetic point of view, the doped atoms are most sable at the interstitial site. The bulk modulus, Young's modulus, shear modulus and Possion's ratio of all doped UO2 are lower than undoped UO2. The point defects in doped UO2 are also different with undoped UO2.

In addition, Understanding the microstructure evolution of nuclear fuels under high burn-up is a critical concern to prolong fuel refuelling cycle and improving the safety of nuclear reactor. Hence, a phase-field model was proposed to study the high burn-up structure evolution in UO2 with different grain size. The variation of recrystallization kinetic and porosity with burn-up is in good agreement with experimental measures. Increasing grain size reduced the porosity and the fraction of high burn-up structures.



Yong Xin is engaged in research related to the mechanisms of complex nuclear fuel behavior, numerical simulation of fuel behavior and specialized fuel research. He currently serves as the Director of the High Burnup Center at the Nuclear Power Institute of China and is a leading expert in ceramic fuel element design.He has published more than 20 papers and granted 7 invention patents.

Atomistic simulations of the clustering of vacancy and rhenium in Mo-Re alloys

Lu Sun, Li Chen, Ming-Jun Li, Meng-Lu Qin, Zheng-Feng Tong
North China Electric Power University

Refractory molybdenum-rhenium (Mo-Re) alloys are considered as the best core structural materials of heat pipe cooled space nuclear reactors. However, the irradiation induced embrittlement and creep will lead to the mechanical property degradation of Mo-Re, which is predominantly associated with the interactions of irradiation defects such as vacancies, interstitials, and their clusters.

In the present study, the clustering behaviors of vacancy and Re and the corresponding binding energies were investigated based on atomistic simulations. In particular, a newly developed Mo-Re interatomic potential by machine learning was used in the molecular dynamic calculations. The rule underlying the vacancy clustering in pure Mo was first revealed. The vacancy binding energies increase with fluctuation with the increasing vacancy number (Fig.1). This trend was demonstrated to be related with the coordination environment of vacancies, to be specific, the total number of the formed 1st nearest neighboring (1NN) and 2NN vacancy pairs in the V_n cluster. Therefore, the vacancy binding energies could be described as the function of the 1NN and 2NN vacancy-pair numbers (N1, N2). Using this general expression, the vacancy binding energies in more large-size clusters could be derived and applied to OKMC and CD simulations to obtain the number density and size of vacancy clusters.

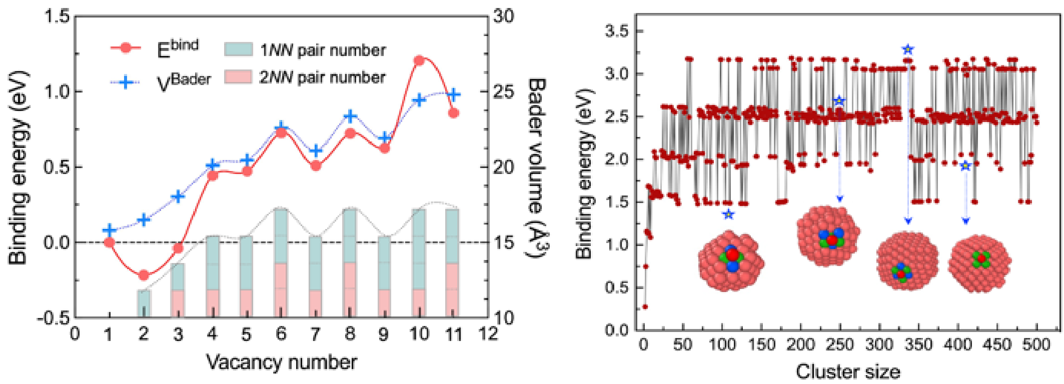


Fig.1. Vacancy binding energies in the cluster composed of different vacancy numbers

The interactions of vacancy and Re were then calculated. Re atoms show a disperse distribution in the perfect Mo supercell and form a most stable <111>Re-Re pairs. When vacancies were introduced, it prefers the site adjacent to the Re atom, and the favorable configuration of vacancy-Re complex is the 2NN-Re-Vac pair along the <001> direction. Following this pattern, the configurations of vacancy clusters in the presence of *n*Re atoms were determined (Fig.2a). Vacancies will be first separated from each other but bind with Re atoms, and the negative vacancy binding energies continuously decrease with increasing vacancy number (Fig.2b). Interestingly, at a certain vacancy number, the clustering of vacancies becomes energetically favorable and show a similar behavior to that in pure Mo. Furthermore, if vacancy clusters are pre-formed, Re atoms will tend to segregate to the vacancies, leading to the formation of Re_nVac_m clusters.

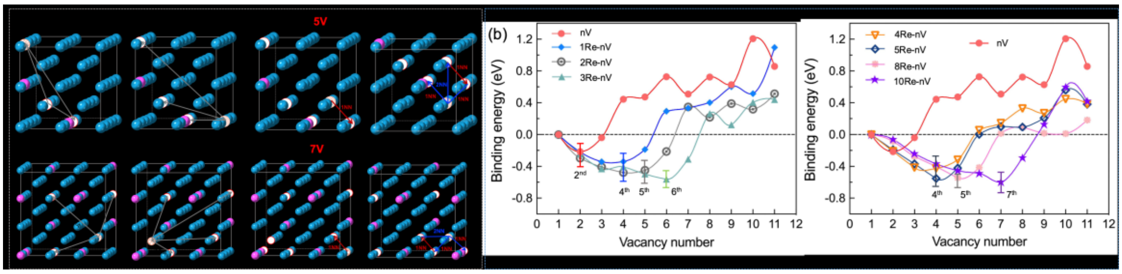


Fig.2 (a) Atomic configurations and (b) binding energies of vacancy clusters in the presence of Re atoms

Our work sheds a new light on the clustering behavior of vacancy and Re in Mo-Re alloys, and provides fundamental energy parameters for further investigation on the evolution of vacancy-Re clusters.

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Research Interests: Multi-scale simulations on the nucleation,diffusion,and evolution behaviors of irradiation defects in nuclear metallic materials.

High-entropy complex ceramics under in-situ irradiation

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Complex ceramics, such as MAX phases and pyrochlore oxides, exhibit unique atomic structures and excellent properties under extreme environments. Due to their unusual characteristics, these materials have been proposed for use in a wide variety of industrial applications, such as accident-tolerant fuel cladding and nuclear waste forms. In the past ten years, it was found that chemical complexity significantly affects structures and properties in materials, such as high-entropy alloys and ceramics. In this talk, I will summarize our recent experimental and theoretical works on the behaviors of the high-entropy Mn+1AXn phases and high-entropy pyrochlores under in situ ion irradiation. (TiVNbZrHf)2SnC was compared with (TiVNbZrHf)2SnC, and Gd2(Ti0.2Zr0.2Sn0.2Hf0.2Ta0.2)2O7 and Gd2(Ti0.2Zr0.2Sn0.2Hf0.2Nb0.2)2O7 were compared with Gd2Sn2O7. In both series of materials, irradiation drives similar order-to-disorder phase transformations in high-entropy ceramics to their corresponding single-component materials. However, these high-entropy ceramics exhibit better tolerance to irradiation-induced phase transformation and amorphization than the corresponding single-component materials. The roles of chemical complexity on the susceptibilities of these materials to structural evolution were elucidated by ab initio calculations. However, the high entropy effect still needs to be further studied. This talk will provide an understanding of structural evolution in high-entropy ceramics and propose a new strategy for designing novel complex ceramics with enhanced performance under extreme irradiation conditions.

Dr. Chenxu Wang is an Assistant Professor in the School of Physics at the Peking University, China. He received his Ph.D. in Nuclear Technology and Application from Peking University in 2016. Dr. Wang worked at Stanford University as a postdoctoral researcher from 2016 to 2020. He has published 50 peer-reviewed papers and received the ICACC Global Young Investigator Award, Excellent Ph.D. Dissertation Award and the Best Oral Presentation Award at the CICC. Dr. Wang's research interests are in materials in extreme environments, including radiation effects in novel nuclear materials, H/He synergistic effects in fusion reactors, and material modification using ion beams and high pressure.

Radionuclide Tracing Based in situ Corrosion and Mass Transport Monitoring of 316L Stainless Steel in a Molten Salt Closed Loop

Yafei Wang
Shanghai Jiao Tong University

In the study, we report an in situ corrosion and mass transport monitoring method developed using a radionuclide tracing technique for the corrosion study of 316L stainless steel (316L SS) in a NaCl-$MgCl_2$ eutectic molten salt natural circulation loop. This method involves cyclotron irradiation of a small tube section with 16 MeV protons, later welds at the hot leg of the molten salt flow loop, generating radionuclides ^{51}Cr, ^{52}Mn, and ^{56}Co at the salt-alloy interface. By measuring the activity variations of these radionuclides at different sections along the loop, both the in situ monitoring of the corrosion attack depth of 316L SS and corrosion product transport and its precipitation in flowing NaCl-$MgCl_2$ molten salt are achieved. While 316L SS is the focus of this study, the technique reported herein can be extended to other structural materials being used in a wide range of industrial applications.

Yafei Wang is an Associate Professor at the Shanghai Jiao Tong University. Before his appointment at the university, he worked as an assistant scientist and postdoctoral researcher at University of Wisconsin-Madison (USA). His research was focused on the molten salt chemistry and molten salt corrosion for the deployments of molten salt reactor and pyroprocessing of spent nuclear fuel. He got his PhD degree and Master degree in Nuclear Engineering from Virginia Tech and The Ohio State University, respectively.

Concurrent effect of hydrogen and irradiation in Zr-based alloy

Huilong Yang[1], Dongyue Chen[2], Sho Kano[3], Hiroaki Abe[3]
[1] Shanghai Jiao Tong University;
[2] Tsinghua University;
[3] The University of Tokyo

Zr-based alloys have been extensively utilized as fuel cladding and structural components in almost all water-cooled reactors. Various degradation behaviors readily occur during the long-term operation of these components, such as irradiation and hydrogenation induced embrittlement. To date, great efforts have been attempted to understand the single effect of either irradiation or hydrogenation on the microstructural and mechanical property change. However, the synergetic effects from both irradiation and hydrogenation have not been fully understood, which is of great importance to understand and to predict the in-pile performance of the key Zr-based components, as these components are suffered from the harsh environment including both intense irradiation and hydrogen adsorption. Therefore, the purpose of this study is to elucidate the concurrent effect of hydrogen and irradiation on the microstructural and mechanical property change in Zr-based alloys.

Zircaloy specimens were employed for this study. Zircaloy-2 specimens with and without H-charging were irradiated in BR2 reactor at Belgium, the irradiation temperature was maintained ~563 K and the dose was achieved up to ~0.2 dpa. Tensile testing at RT-563 K was conducted to evaluate the mechanical property change upon irradiation, followed by the fracture morphology observation. Results show that a significant hardening/strengthening occurs at all the testing temperatures, strength increases quickly at low dose regions up to ~0.01 dpa, then gradually saturated at higher doses. For the H-charged specimen, high temperature (563 K) testing results show that the elongation is greater compared to the specimen without irradiation, meanwhile, the elongation of H-charged specimen (0.025 dpa) is even greater than that of the specimen (0.01 dpa, without irradiation), indicating that the proper adsorption of hydrogen could assist to mitigate the irradiation-induced embrittlement. TEM observation further shows that the presence of H results in a greater size and a lower density of dislocation loops, indicating that the irradiation defect formation and accumulation process are influenced by hydrogenation.

To further elucidate the concurrent effect of hydrogen and irradiation in Zr-based alloy, in-situ TEM observation under irradiation was conducted at the High Fluence Irradiation Facility, the University of Tokyo (HIT). 1 MeV Fe^{+} ion was used as ion source, and the irradiation temperature was 633 K. In-situ TEM observation results show that the density of irradiation loops increases quickly at the dose < 0.1dpa, and the size increases at the dose >0.4 dpa due to the agglomeration and the coalescence of the point defects. The defect density in H-charges specimen is slightly higher than that of the as-received specimen, probably because the formation of H-vacancy complex lowers the migration rate and enhances the nucleation of defects. Evidence that solute H atoms facilitate the formation of irradiation loops was also clear observed. In addition, the hydrogen in solid solution from the dissolved hydrides during heating process could be reprecipitated during the cooling process. Irradiation suppresses the precipitation of hydrides, probably because irradiation-induced defects increase the solid solubility of hydrogen in Zr by trapping hydrogen atoms. These insights can be supplemented to provide a better understanding of the degradation behaviors of the Zr-based key components in water-cooled reactors.

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2016. 04 - 2017. 09: Project Researcher, The University of Tokyo
2014.10 - 2016.03: Postdoctoral fellow, Tohoku University

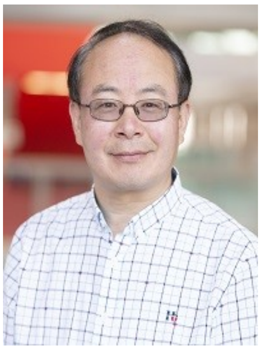
Unveiling the Microstructural Evolution in Zirconium Alloys Through In-Situ Transmission Electron Microscopy under Irradiation

Zhongwen Yao
Smith Engineering, Queen's University, Canada

Understanding the impact of irradiation on the microstructure of zirconium alloys is critical for ensuring the safety and reliability of nuclear reactors. In this presentation, we applied in-situ/ex-situ transmission electron microscopy (TEM) as a powerful tool to unravel the dynamic evolution of zirconium alloy microstructures during irradiation or after irradiation.

Through real-time TEM/STEM imaging, we explore the intricate processes taking place within zirconium alloys, such as Excel alloy, Zr-2.5Nb and Zircaloy as they are subjected to irradiation. We observe and analyze the complicated changes in microstructure, including a-type and c-type dislocation formation, defect accumulation, and phase transformations. These microstructural changes have a direct bearing on the material's mechanical properties and dimensional stability, making this research crucial for nuclear industry.

This presentation highlights the invaluable contribution of in-situ TEM techniques in providing real-time, atomic-scale observations of zirconium alloy behavior during irradiation, paving the way for enhanced material design and reactor safety.



Dr. Zhongwen Yao is a full professor at Queen's University since 2009. He obtained his PhD at the Swiss Federal Institute of Technology, then he joined the University of Oxford as a postdoctoral research fellow in 2005. He is an expert in the field of electron microscopy, nuclear materials, and radiation damages. He has published more than 150 refereed journal articles and contributed critical work in understanding of irradiation effects of nuclear materials.

Influence of helium on the self-healing of irradiation defects in tungsten

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Department of Physics, Beihang University, Beijing 100191, China

Helium (He), as a typical impurity element in nuclear materials, plays a crucial role on the microstructure and mechanical properties. Generally, the presence of He is expected to aggravate the irradiation damage in materials. However, recent experiments found that the He addition may also reduce the number of surviving defects in materials under high-energy particles irradiation, indicating the enhancing effect of He on self-healing efficiency, while its underlying mechanism is still unclear.

We have systematically investigated the interactions between He and self-interstitial atoms (SIAs) as well as their influences on the kinetic behaviors of SIAs in tungsten (W), using both first-principles and Object Kinetic Monte Carlo methods. It is found that there are attractive interactions between He and SIAs, which become stronger with the increasing of SIA numbers. Specifically, the He-SIA1 and He-SIA2 complexes adopt a three-dimensional (3D) migration pattern with an effective energy barrier of 0.38 and 0.61 eV, respectively, which is completely different from the 1D migration of SIAs in W ($\leq$ 0.033 eV) without He. Such an unexpected collaborative 3D motion of He-SIA complexes increases the probability of Frenkel pairs recombination and reduces the number of surviving defects. Consequently, our calculations reveal the enhanced effect of He on the self-healing efficiency in W, which is originated from the collaborative 3D motion of He-SIA complexes. The current results can improve our fundamental understanding of the influence of He on the evolution of irradiation defects and have great implications to estimate the performance of W-PFMs in fusion environment.

Hong-Bo Zhou is a Professor in the Department of Physics in Beihang University. His research interests lie in the irradiation damage of nuclear materials. He has conducted systematic research on the mechanism and controlling of irradiation damage for fusion materials: developed a series of high-efficiency programs (i.e., kinetic Monte Carlo and cluster dynamics) for irradiation damage simulation, revealed the interstitial-mediated mechanism of transmutation precipitation and identified a phase diagram for the co-evolution of transmutation elements and irradiation defects; revealed the synergistic effect of small helium clusters on irradiation hardening. Up to date, he has published 90 journal papers, granted 7 software copyrights. Besides, he has been nominated as the member of the International Scientific Committee for the International Workshop on Models and Data for Plasma-Material Interaction in Fusion Devices (MoD-PMI) and the council member of Nuclear Materials/Fusion and Plasma Physics branches in Chinese Nuclear Society. He has been selected as Young Yangtze River Scholar, honored with Early Career Award in Computational Materials Science.

Deuterium-helium mixed plasma irradiation on surface modification of ZrC dispersion-strengthed tungsten

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Tungsten (W) is the primary candidate for plasma-facing materials in future fusion device, among which ZrC particle dispersion-strengthened W (WZrC) exhibits excellent performance in strength/ductility, ductile-brittle transition temperature (DBTT), and thermal shock resistance. However, the influence of reactor-relevant plasma irradiation on WZrC has not been well understood. In this presentation, we reported the surface modification of WZrC induced by pure helium (He) plasma, pure deuterium (D) plasma, and D-He mixed plasma irradiation conducted in CLIPS (Compacted Linear Plasma-Surface interaction device, located in USTC), under both high temperature (~900°C) and high fluence (from ~10^{24} m^{-2} to ~10^{26} m^{-2}) under the energy < 100 eV. For pure He irradiation, typical fuzz structures were observed on the W matrix of WZrC, and the ZrC particles were eroded with increasing He fluence. For pure D irradiation, the W matrix became rough. And the ZrC particles was found swelling, which should be ascribed to the formation of Zr hydrides. When under D-He mixed plasma irradiation, fuzz structures were also observed on W matrix, with comparable thickness to pure He irradiation. However, ZrC particles entirely disappeared, left smooth-edge craters. EDX analysis indicated that there was no Zr content near these craters. The fuzz nearby has distinctive structures. Moreover, further investigations have been done on the cross-section of the craters under D-He mixed plasma irradiation.

Dr. Ze Chen was born in 1994 in China. He received the Doctor of Engineering degree from University of Science and Technology of China (USTC) in 2022 and became a post-doc at USTC since then. He used to work in Oak Ridge National Lab (ORNL) in USA from 2018 to 2020 as a joint-PhD student. He is interested in hydrogen isotopes permeation and retention in fusion reactor materials, plasma-wall interaction, and relevant molecular dynamics simulation.

Surface modification of ZrC dispersion-strengthened W under low energy He plasma irradiation

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Tungsten (W) is considered as a candidate for plasma-facing materials (PFMs) in fusion devices. However, the brittleness, poor machinability, and low strength at high temperature limit its performance. To resolve these problems, various W with enhanced mechanical properties have been developed over recent decades. ZrC dispersion-strengthened W exhibits high strength/ductility, low ductile-to-brittle transition temperature, and excellent thermal shock resistance, making it a promising candidate PFM for future fusion devices [1].

In this study, surface modification of 0.5 wt.% ZrC dispersion-strengthened W (WZrC) under low energy and high fluence He plasma irradiation at high temperature was presented. Under the energy of 90 eV and fluence ranging from 6 × 10^{24} He·m^{-2} × 10^{26} He·m^{-2} He irradiation at about1200 K in CLIPS (Compact LInear Plasma-Surface interaction device), a typical fuzz nanostructure appeared on the W matrix of WZrC. The fuzz showed comparable thickness and structural features to pure W, indicating limited effects of the particle's addition on resistance to high fluence He irradiation at high temperatures. Besides, the erosion behavior of particles under He plasma irradiation has been investigated. As show in Fig. 1, under the He influence of 6 × 10^{24} He·m^{-2}, only nanopores were observed. With fluence increasing to 5 × 10^{25} He·m^{-2}, the surface became relatively uneven with larger holes. When fluence further increased to 2 × 10^{26} He·m^{-2}, the particles were eroded completely and covered by the extended fuzz, forming cavities. In addition, distinctive layered nanotendrils were observed above the cavities (Fig. 2d-f), which were characterized to consist of inner W-riched skeletons and outer Zr-riched layers. It indicates that the layered nanotendrils should be the result of fuzz extension combined with particle sputtering/deposition.

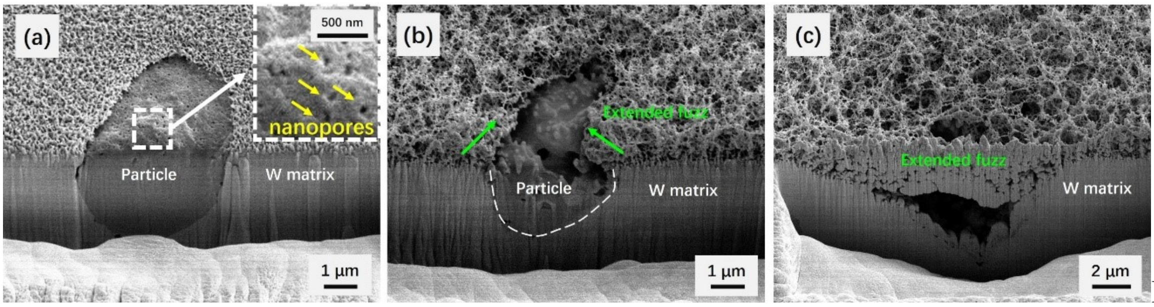

Figure 1. (a) Cross-section images of the particles in WZrC under different He fluences. (a) Low He fluence (6 × 10^{24} He·m-2). (b) Middle He fluence (5 × 10^{25} He·m-2). (c) High He fluence (2 × 10^{26} He·m-2).

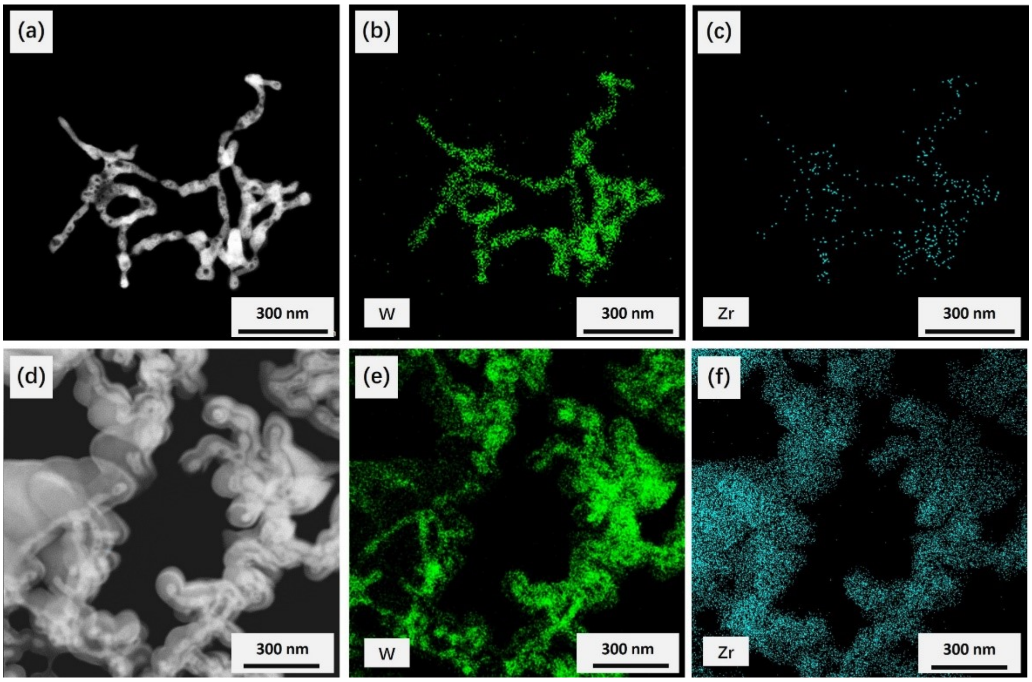

Figure 2. (a) HAADF-STEM of the uniform fuzz on W matrix. (b) and (c) W and Zr mappings of the uniform fuzz. (d) HAADF-STEM of layered nanotendrils above cavities. (e) and (f) W and Zr mappings of layered nanotendrils.

Long Li, PhD candidate, School of Nuclear Science and Technology, University of Science and Technology of China.

Effect of low-energy He/D plasma irradiation on W-Ta-V-Cr multi-component alloys

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Plasma-facing materials (PFMs) are exposed to severe conditions, including high thermal loads, intense neutron irradiation, and high-flux plasma, resulting in the formation of irradiation defects that degrade mechanical performance and induce surface damage. Multi-component alloys (MCAs), also known as high-entropy alloys, have emerged as promising candidates in the field of nuclear materials due to their remarkable radiation resistance. However, to be considered for use in PFMs, MCAs require a thorough investigation into the surface modification processes following low-energy plasma irradiation, in addition to enhancing their thermal conductivity.

In this presentation, we present a series of low-energy, high-flux He/D plasma irradiation experiments conducted on W-Ta-V-Cr MCAs using the Compact LInear Plasma-Surface interaction device (CLIPS) at the University of Science and Technology of China. The MCAs were fabricated via the Arc-melting technique. The irradiated specimens underwent characterization via thermal desorption analysis (TDA), transmission electron microscopy (TEM), and scanning electron microscopy (SEM) coupled with the focused ion beam (FIB) technique and energy-dispersive X-ray spectroscopy (EDS) to assess He and D retention, surface morphology, and chemical composition alterations. Our findings advance the understanding of surface modifications in MCAs under low-energy He/D plasma irradiation, elucidating the relationship between surface defect properties and hydrogen isotope retention.

Dr. Chao Yin is an associate researcher at the University of Science and Technology of China (USTC). He pursued his doctoral degree in Belgium from 2017 to 2021, conducting four years of research on the high-temperature neutron irradiation effects of tungsten materials for fusion reactors at the Belgian Nuclear Research Centre (SCK CEN) and participating in EU nuclear fusion/nuclear fission-related projects. He continued his research as a postdoctoral fellow at USTC from 2021 to 2024 under Professor Minyou Ye's supervision. Currently, his research focuses on simulating neutron irradiation-induced annealing phenomena and irradiation effects on high-entropy alloys/multi-component alloys.

Dr. Yin has led a National Natural Science Foundation of China (NSFC) youth fund project, an Anhui provincial natural science foundation youth project, and has participated to a NSFC joint fund project. He has received recognition, including third place in the 2020 Belgian Nuclear Society Young Generation Scientific Contest, being shortlisted for the 2020 European Nuclear Education Network PhD Prize, and grants such as the Taiwan Young Talents Program of the Chinese Academy of Sciences, the Mozi Outstanding Young Scholar Special Allowance, and the Xinao Postdoctoral Award. He has published 9 first-author/corresponding-author SCI research articles and more than 15 co-authored SCI/EI research articles.

In-situ Irradiation Facility at Xiamen University in China

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This report primarily focuses on the in-situ irradiation facility at Xiamen University in China, detailing its structure and the advancements in irradiation research that have been made using this facility. The facility, known as the Xiamen Multiple Ion Beams In-situ TEM Analysis Facility (XIAMEN Facility), consists of a TECNAI G20 F30 TWIN TEM coupled to a 400kV implanter (30-400kV) and a 50 kV hydrogen/helium coaxial ion source (10-50kV) to provide a wide range of ion species with different energies. The facility's design includes two ion beam lines intersecting at a 45º angle, which meet the electron beam at a 68º angle. Presently, the facility offers seven distinct ion beam combinations: heavy ions only, heavy ions+He^+, heavy ions+$H_2^+(H^+)$, heavy ions+He^++$H_2^+(H^+)$, He^++$H_2^+(H^+)$, He^+ alone, and $H_2^+(H^+)$. The construction of the facility was completed in January 2020. With the support of the ITER key research project of the Ministry of Science and Technology, we have carried out corresponding transformation of the equipment, and realized a wide range of ion beam generation technology by adding a scanning system and a beam outlet baffle, thus increased the output limit of the hydrogen/helium coaxial ion source; Then, the weak beam measurement technology was improved to accurately measure the weak beam intensity, spot size and uniformity. Finally, the high-stability control technology of irradiation beam was perfected to achieve accurate dose control, uniform and stable irradiation in the whole process of long-term in-situ irradiation. The modified equipment realizes that the dose rate and hydrogen/helium injection rate when three ions are irradiated at the same time can be adjusted in two orders of magnitude.

Leveraging this facility, in-situ studies on the irradiation behavior of various nuclear energy materials have been conducted. These studies have led to a number of groundbreaking findings. The materials examined include Mo-Re alloys for space nuclear reactors, W and W-ZrC alloys for fusion reactors, FeCrAl steels for fuel claddings, Pd for tritium storage, and simulated fuel CeO2, among others. The research conducted at this facility has been featured in prestigious publications such as Acta Materialia, Scripa Materialia, Journal of Materials Science & Technology, Materials Today, ACS Applied Energy Materials, and Journal of Nuclear Materials. Figure 1 shows schematics and images of the Xiamen Multiple Ion Beam In-Situ TEM Analysis Facility.

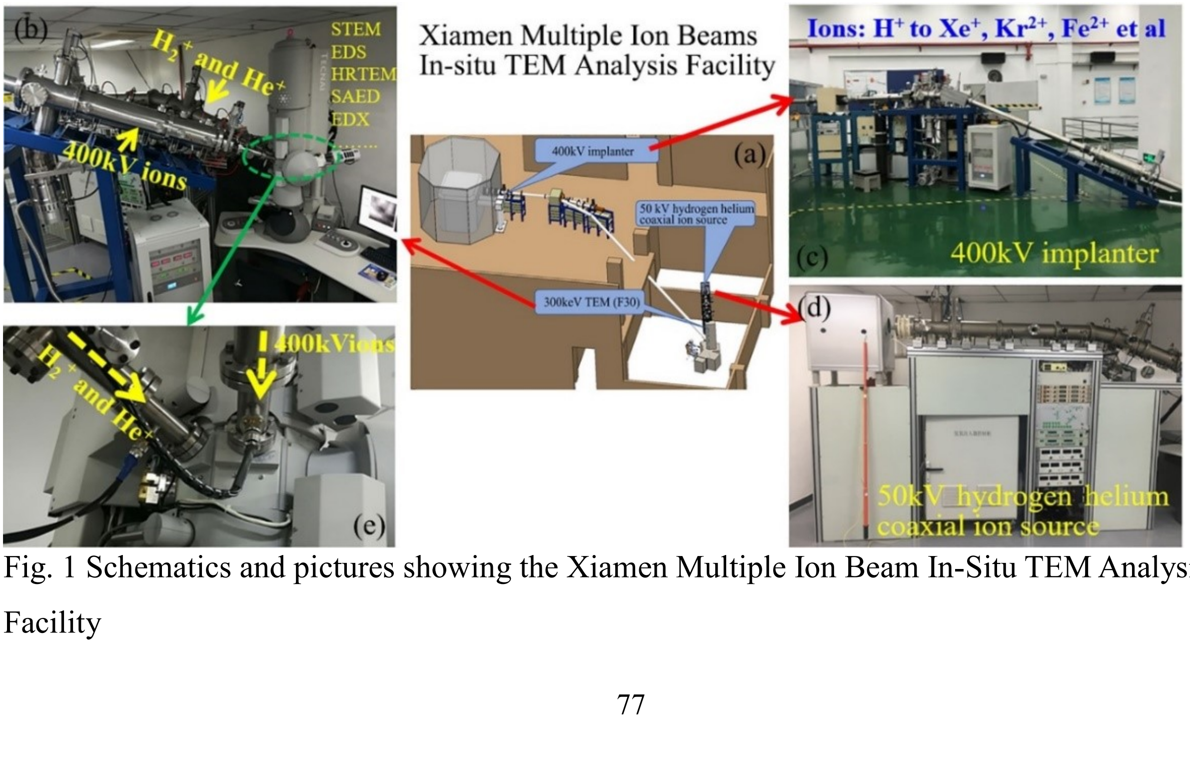

Fig. 1 Schematics and pictures showing the Xiamen Multiple Ion Beam In-Situ TEM Analysis Facility

Jinchi Huang is the engineer of colleage of energy, Xiamen University. And is currently responsible for the operation and maintenance of the Xiamen Facility. The research direction is the material irradiation effect and the ion implantation behavior. He operates for both external and internal projects.

In-situ TEM observation of microstructure evolution in Mo-Re alloy under Fe+ irradiation

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Molybdenum-rhenium (Mo-Re) alloys are considered as the promising core materials of new high temperature reactor, owing to their excellent comprehensive performance such as high temperature strength, good low temperature processing performance, and outstanding compatibilities with nuclear fuels. In this study, in-situ irradiation with 400 keV Fe ion were performed on molybdenum rhenium alloys on investigation of the influences of irradiation temperature, irradiation dose, grain structure, sample thickness, and Re content on dislocation loop evolution including nucleation, growth, reaction, coalescence, and changes in habit planes. The average size of dislocation loops increases with the increase of irradiation dose. At the same time, the density will begin to decrease after reaching a peak at the beginning of irradiation. There are three main reasons for the disappearance of dislocation loops. One is that the reaction between dislocation loops leads to the disappearance of the initial dislocation loop, the other is that the dislocation loop is absorbed by a strong defect sink, and the third kind of surrounding dislocation loop leads to the absorption and disappearance of the dislocation loop. The smaller the grain size and the higher the pre-existing dislocation line density, the less likely to generate irradiation induced dislocation loops. The thicker the TEM sample, the higher the density of dislocation loops will be generated. At the same sample thickness, the size of the dislocation loop increases significantly with increasing irradiation temperature, while the density decreases. With the increase of Re content, the size of dislocation loops decreases and the density increases, resulting in a higher degree of radiation hardening caused by irradiation induced dislocation loops.

Xi Qiu, PhD and Engineer at Nuclear Power Institute of China (NPIC). His main research area focus on the design and analysis of nuclear fuels and materials. He has published over 20 academic papers in the journal such as Acta materialia, Journal of Nuclear Materials, Journal of Materials& Technology and Nuclear Power Engineering.

Surface modification of Si by low-energy He ion irradiation

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The phenomenon of self-organized nanopatterning on material surfaces, particularly Si, through ion beam irradiation (IBI) has garnered considerable attention in recent decades. Significantly noteworthy progress has been made in elucidating the underlying mechanisms governing ion-material interactions, coupled with notable advancements in experimental results. These outcomes affirm the versatility, expeditiousness, and cost-effectiveness of the IBI method as a bottom-up approach for achieving surface nanopatterning. This technique exhibits promising potential for applications across various domains [1,2].

However, a legitimate constraint emerges concerning the ion-substrate pairing: scenarios arise where no structures manifest when the ion possesses a lower mass than the substrate [3]. This phenomenon substantiates the inadequacy of He and Ne ion beams for Si nanopatterning, contributing to the limited extent of related investigations. Nevertheless, a well-established realization within the past decade reveals that low-energy He plasma irradiation can induce various nanostructure formations on diverse metals and semiconductors [4]. This finding contradicts preceding knowledge, presenting a paradox.

In this study, we experimentally investigate Si substrates exposed to low-energy (< 100 eV) and high-flux (~ 10^{22} $m^{-2}s^{-1}$) He plasma irradiation under normal incidence. By rigorously maintaining contamination control measures, the irradiated Si surface reveals distinctive mountain-like nanostructures with well-defined facets and grooves extending along mutually perpendicular <110> directions. In cases involving pre-deposited metal impurities, notably Ta, which previously exhibited no impact on nanostructure growth during Ar ion beam irradiation trials, a densely packed arrangement of nanocones emerges on the irradiated Si surface. SEM cross-sectional images distinctly illustrate the presence of He bubbles, oriented nearly perpendicular to the irradiated surface. These subsurface He bubbles introduce a novel instability, fostering nanostructure formation on the Si surface, even under normal incidence conditions. This finding necessitates a reassessment of the nanopatterning effects of low-energy He ion irradiation on materials. Furthermore, it calls for a comprehensive investigation into the influence exerted by He bubble formation on the scaling laws in surface evolution.

Zhe Liu is a Ph.D. candidate at the University of Science and Technology of China, under the supervision of Prof. Minyou Ye. His research focuses on the surface modification of semiconductors by ion beam and plasma irradiation.

Modelling of the fission gas swelling for UO_2 fuel in PWR

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The irradiation swelling caused by fission gas can promote UO_2fuel cladding contact and reduce fuel thermal conductivity, which is a key behavior affecting fuel element performance. This article builds a fission gas irradiation swelling model for different burnup rangesbased on rate theory. The model first proposes control equations for intragranular gas, intergranular gas, and point defects under low burnup. Then, the control equations for intragranular gas considering grain subdivision and non-equilibrium growth of grain boundary pores under medium burnup are given. Finally, a model for coalescence and coarsening of grain boundary pores under high burnup is established. On this basis, COMSOL software is used to solve the control equations. The model was preliminarily validated using experimental data, and the predicted bubble size and porosity of the model were in good agreement with the experimental results.

Keywords: Fission gas; Irradiation swelling; Model; High burnup

Yi Zhou is a leading scientist and a senior engineer at the Nuclear Power Institute of China. He has long been engaged in the design of nuclear fuel components and the evaluation of their irradiation performance. He has published more than 60 papers and co-authored 2 academic monographs. He holds 20 authorized patents and registered software copyrights, and has received 7 provincial and ministerial-level and association-level achievements.

The synergetic effect of He and Kr irradiation on helium bubble evolution in SiC/SiC composite: Combining in-situ TEM observation with MD simulation

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SiC fiber reinforced SiC matrix composites (SiC/SiC composites) are considered as one of the promising structural materials for liquid type breeding blanket in the fusion energy system. The transmutation gas production rate in fusion reactor is significantly higher compared with fission reactors. In order to evaluate the synergetic effects of helium and irradiation on SiC/SiC composites, an innovative in-situ ion irradiation and TEM observation combined with MD simulation were utilized. A comprehensive experiment matrix including Kr irradiation, He pre-implantation with sequential Kr irradiation as well as simultaneous dual beam at 1073 K to 16 dpa and 2400 appm He (150 appm/dpa) was performed using 800 keV Kr and 50 keV He ions.

Combining in-situ TEM observation and MD simulation, the impact of relative concentration ratio of vacancies and helium atoms on the bubble evolution was elucidated. Bubble nucleation does not occur without helium atoms. When only helium atoms are introduced, limited bubbles can nucleate and will quickly reach saturation. When vacancies are introduced after helium implantation, a number of helium bubbles immediately nucleate at first and then the bubble density reduce with increasing irradiation dose. When introducing helium atoms and vacancies simultaneously, the He_nV_m cluster develop into helium bubbles continuously.

In addition, the direct evidence for the competition of irradiation-assisted bubble growth and re-dissolution was in-situ observed in SiC matrix for the first time. Vacancies introduced by irradiation are dispensable to the nucleation and growth of bubbles, while the attendant collision cascades and thermal spikes can result in the re-dissolution of helium atoms in bubbles. The competition between these two processes leads to the growth and shrinkage of bubbles.

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Using in-situ TEM to study the irradiation-induced defects in high-purity palladium (Pd)

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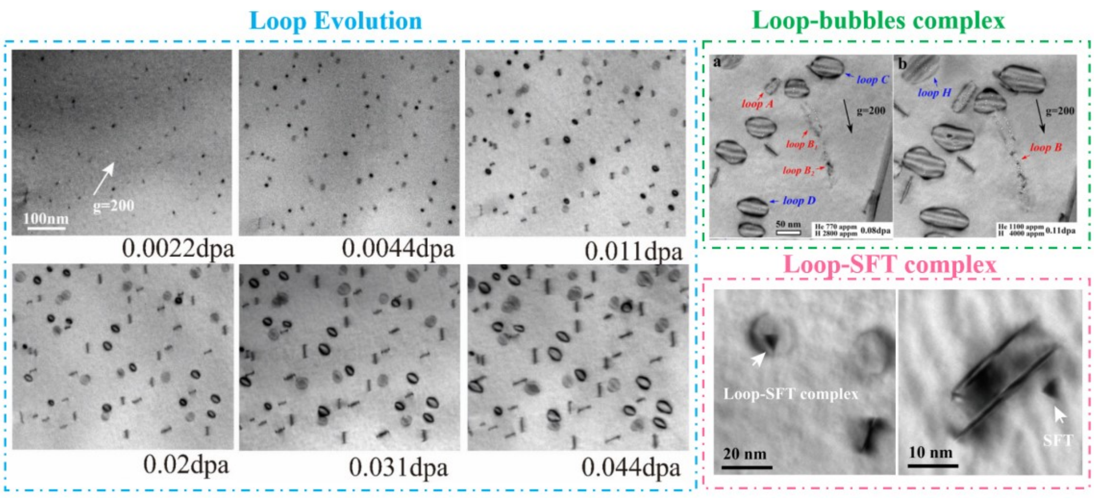
Ion irradiation performed within the advanced in-situ TEM greatly improves the ability to study the irradiation damage of nuclear materials. It not only enables better control of irradiation parameters (including irradiation dose, dose rate, and temperature), but also identifies dynamic processes within the regions of interest, continuously tracks the real-time response of microstructures to external stimuli (such as irradiation, heat, stress, etc.), and eliminates or quantifies the impact of research variables such as microstructural differences. Here, we present some novel research work on the evolution of irradiation defects (e.g., dislocation loops, helium bubbles, SFT, etc.) in high-purity Pd using the Xiamen Multiple Ion Beam In-situ TEM Analysis Facility (XIAMEN Facility). In-situ experiments of irradiated high-purity Pd with single H_2^+ beam, single He^+ beam and He^+-H_2^+ double beams were carried out. The evolution and interaction mechanisms of defects were investigated under a series of irradiation parameters. Our work tracked the in-situ transformation process of Frank dislocation loops (FDLs) to perfect dislocation loops (PDLs) and revealed that the unfaulting of FDLs depends on loop interactions, while loop energy is a necessary but insufficient condition for loop unfaulting. On the basis of exploring the interaction between dislocation loops and bubbles, the order of hydrogen and helium trapping ability was quantified as follows: PDLs > FDLs > the region outside the loops. For the first time, the irradiation-induced formation of novel Frank loop-SFT complexes in Pd with extremely high SFE was discovered, and their formation mechanism was revealed to be that vacancy clusters rearrange directionally to form SFTs due to the ambient stress deviations and compressive stress fields induced by interstitial Frank loops. This research was supported by the National Natural Science Foundation of China (Grant No. 12305308).



Yipeng Li is currently a special associate researcher/post-doctoral fellow at Xiamen University. His research focuses on the irradiation behavior and damage mechanism of nuclear materials, and he is also interested in micro- and nano-scale mechanical testing and microstructure-property relationships. Dr. Li has published 19 papers as first author, co-first author and co-corresponding author in journals such as *Acta*, *Scripta*, *JMST*, *MSEA* and *JNM*.
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