22nd INTERNATIONAL CONFERENCE ON FUSION REACTOR MATERIALS

ICFRM-22
Shizuoka

Tentative_20250915

Conference Program and Abstracts

https://www.icfrm-22.com

Organized by ICFRM-22 Organizing Committee

Co-organized by Shizuoka University

ICFRM-22
Shizuoka

Welcome to ICFRM-22

The conference general chairs, the IAC chair and the local organizing committee are pleased to welcome all of you to the 22nd International Conference on Fusion Reactor Materials (ICFRM-22) held in Shizuoka, Japan on 28th September to 3rd October, 2025.

ICFRM is the major international conference and the leading platform for experts studying, developing and characterizing materials for fusion energy technologies. This biennial conference focuses on structural, functional, plasma facing and high heat flux materials including irradiation effects, materials engineering and technologies, and cutting-edge research. The ICFRM-22 will include plenary lectures, invited and contributed oral presentations, as well as poster sessions and tutorials.

Dr. Ronald Klueh, a retired materials scientist at Oak Ridge National Laboratory, passed away on March 23, 2024. He was a pioneer in the field of RAFM steels for fusion applications and made significant contributions to ferritic/martensitic steel research. A memorial session will be held to honor his achievements and emphasize the progress made through his works.

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Wuxi Paike New Materials Technology Co., Ltd. (hereinafter referred to as "Paike New Materials") was founded in 2006, with a registered capital of 120 million yuan and total assets of 7.2 billion yuan. It currently has more than 900 in-service employees. Listed on the Shanghai Main Board in 2020 (stock code: 605123), Paike New Materials is a high-tech enterprise specializing in the R&D, production and sales of high-end metal materials. It has been honored with titles such as National Enterprise Technology Center, National Manufacturing Single Champion Demonstration Enterprise, and National Specialized, Refined, Differential and Innovative "Little Giant" Enterprise. Its products include various metal material products such as superalloys, titanium alloys, aluminum alloys, magnesium alloys and special stainless steels, which are widely used in aerospace, energy and power, marine engineering and ships, petrochemical machinery and other industries. It has particularly rich experience in independent R&D and manufacturing capabilities in the field of new materials.

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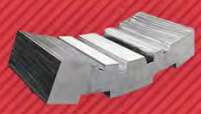
Paike New Materials is equipped with multiple forging equipment such as 7000T press, 6000T press, 3150T press and 3600T press, as well as multiple ring rolling equipment including 5m, 8m and 10m ring rolling machines. It is also furnished with corresponding heat treatment and machining equipment, as well as an independent laboratory. With a high degree of equipment automation, it can meet the product requirements of low cost, high reliability and high stability, representing the first-class level of precision forming of China's special alloys. Paike New Materials has 57 invention patents of various types, and has obtained the Civil Nuclear Manufacturing License (Nuclear Grade 1 Forgings Qualification), Special Equipment Manufacturing License (Pressure Piping Components). It has passed certifications such as AS9100 Aerospace Quality Management System, ISO14001 Environmental Management System, ISO45001 Occupational Health and Safety Management System, EU TUV, NADCAP Forging, NADCAP Heat Treatment, and NADCAP - NDT. Its laboratory has obtained CNAS qualification and NADCAP qualification.

Paike New Materials has always kept up with the trend of scientific and technological development and actively embraced the development of the fusion industry. It has successively carried out material research and development and product supply for key components such as vacuum chambers, shielding blankets, divertors, first walls and coil boxes for the fusion project.


Shielding Cladding Module

Vacuum Chamber Window

Vacuum Chamber Plate

High-temperature Creep Rupture Testing Machine

Heat Treatment Furnace

Ultrasonic Flaw Detection

Coordinate Measuring Machine

WUXI PAIKE NEW MATERIALS TECHNOLOGY CO., LTD.
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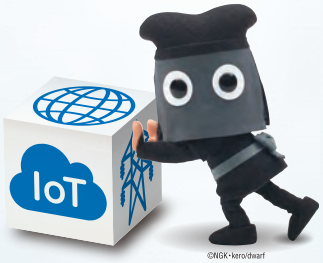

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A.L.M.T. Corp.

HQ: 8-21-1, Ginza, Chuo-ku Tokyo, JAPAN
TEL +81-3-6733-3616 / FAX +81-3-6733-3621
https://www.allied-material.co.jp/en.html

- A.L.M.T. Corp. contributes to various industries by creating products using refractory metals such as tungsten and molybdenum, and diamond.
- For fusion, A.L.M.T. developed tungsten plates for the ITER divertor, and as they showed thermal shock resistance superior to conventional tungsten materials, they were applied to the actual machine as "Tungsten Monoblocks" bonded with copper.
- For future fusion systems such as DEMO, A.L.M.T. is developing advanced tungsten materials as "FUSION GRADE" that achieve longer life and higher toughness.

  - *Tungsten and molybdenum based materials from basic grades to "Fusion Grades"*
  - *Plasma facing components for fusion, e.g., Tungsten Monoblocks for ITER divertors*
  - *Composites with wide range of thermal properties, e.g., CuW*
  - *Precise machining technologies for components production*
  - *Advanced technologies for future fusion, e.g., dissimilar material joining, materials for radiation-shielding, advanced spherical powders*

SUMITOMO ELECTRIC GROUP

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Scientific Topics

A. Development and qualification of structural materials for DEMO and beyond

B. Materials for high heat flux plasmas facing components: divertor, limiter, first wall

C. Development and characterization of functional materials applied in blankets and other n-irradiation affected systems

D. Fusion-specific applications of materials, including environmental effects

E. Materials engineering and application including joining of similar or dissimilar materials

F. Qualification of irradiation effects in neutron sources, accelerators and other test systems including advanced characterization methods

G. Materials-design interface and interactions codes, standards, and standardization including SSTT

H. Fundamental studies of radiation effects: modeling and experimental validation

I. National fusion projects and private fusion activities

J. Cross-cutting issues and synergism with materials applications in other large-scale projects or highly loaded systems

K. Ron Klueh Memorial

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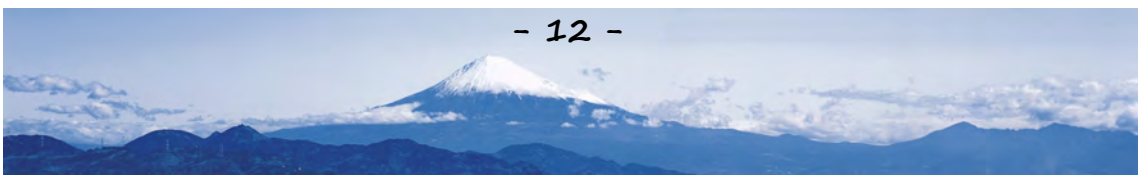

General Information

The main conference is organized over five days from Monday to Friday at MARINART. Tutorial Lectures and Welcome Reception in the afternoon on Sunday (28th September) and Banquet (Gala dinner) in the evening on Thursday (2nd October) are held at Hotel Associa Shizuoka. Monday starts at 9:30 am with the Opening session and Keynote speeches given by representatives of JP and US government. The conference venue of MARINART opens at 9:00 am. Thereafter, plenary talks, parallel sessions and poster sessions are followed.

Conference Venue, MARINART (Google map)

Welcome Reception and Conference Dinner, Hotel Associa Shizuoka (Google map)

Registration desk opens at 13:00-19:00 on Sunday (28th September) in Hotel Associa Shizuoka. From Monday to Friday, the registration desk opens at 9:00-17:00 (Fri: 12:00) at the entrance of MARINART.

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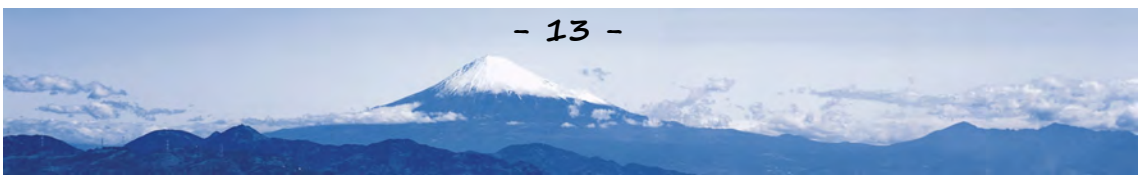

Official Conference Website : https://www.icfrm-22.com

WiFi (MARINART):　SSID：ICFRM_Shizuoka　Pass：icfrm2025

(Please note that TV conference (Zoom, WebEx, MS Teams, etc) is not available on this WiFi to reduce the traffic.)

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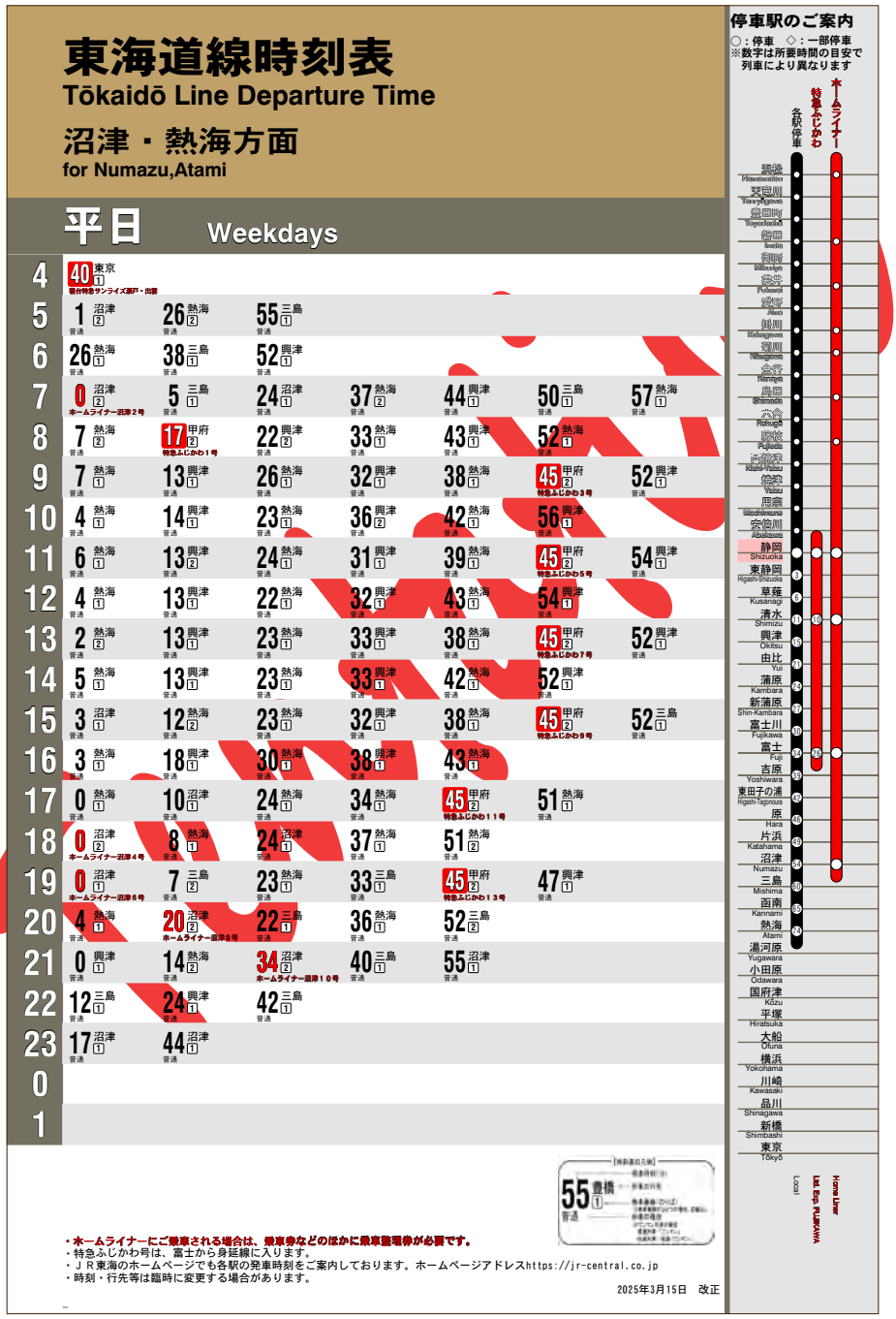
Train Time Table (JR Tokaido Line)

Shizuoka station (near Hotels)



東海道線時刻表
Tōkaidō Line Departure Time
沼津・熱海方面
for Numazu,Atami
平日 Weekdays

4	40東京						
5	1沼津	26熱海	55三島				
6	26熱海	38三島	52興津				
7	0沼津	5三島	24沼津	37熱海	44興津	50三島	57熱海
8	7熱海	17甲府	22興津	33熱海	43興津	52熱海	
9	7熱海	13興津	26熱海	32興津	38熱海	45甲府	52興津
10	4熱海	14興津	23熱海	36興津	42熱海	56興津	
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15	3沼津	12熱海	23熱海	32興津	38熱海	45甲府	52三島
16	3熱海	18興津	30熱海	38興津	43熱海		
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19	0沼津	7三島	23熱海	33三島	45甲府	47興津	
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23	17沼津	44沼津					
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55豊橋 普通

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静岡 Shizuoka
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清水 Shimizu
興津 Okitsu
由比 Yui
蒲原 Kambara
新蒲原 Shin-Kambara
富士川 Fujikawa
富士 Fuji
吉原 Yoshiwara
東田子の浦 Higashi-Tagonura
原 Hara
片浜 Katahama
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Local
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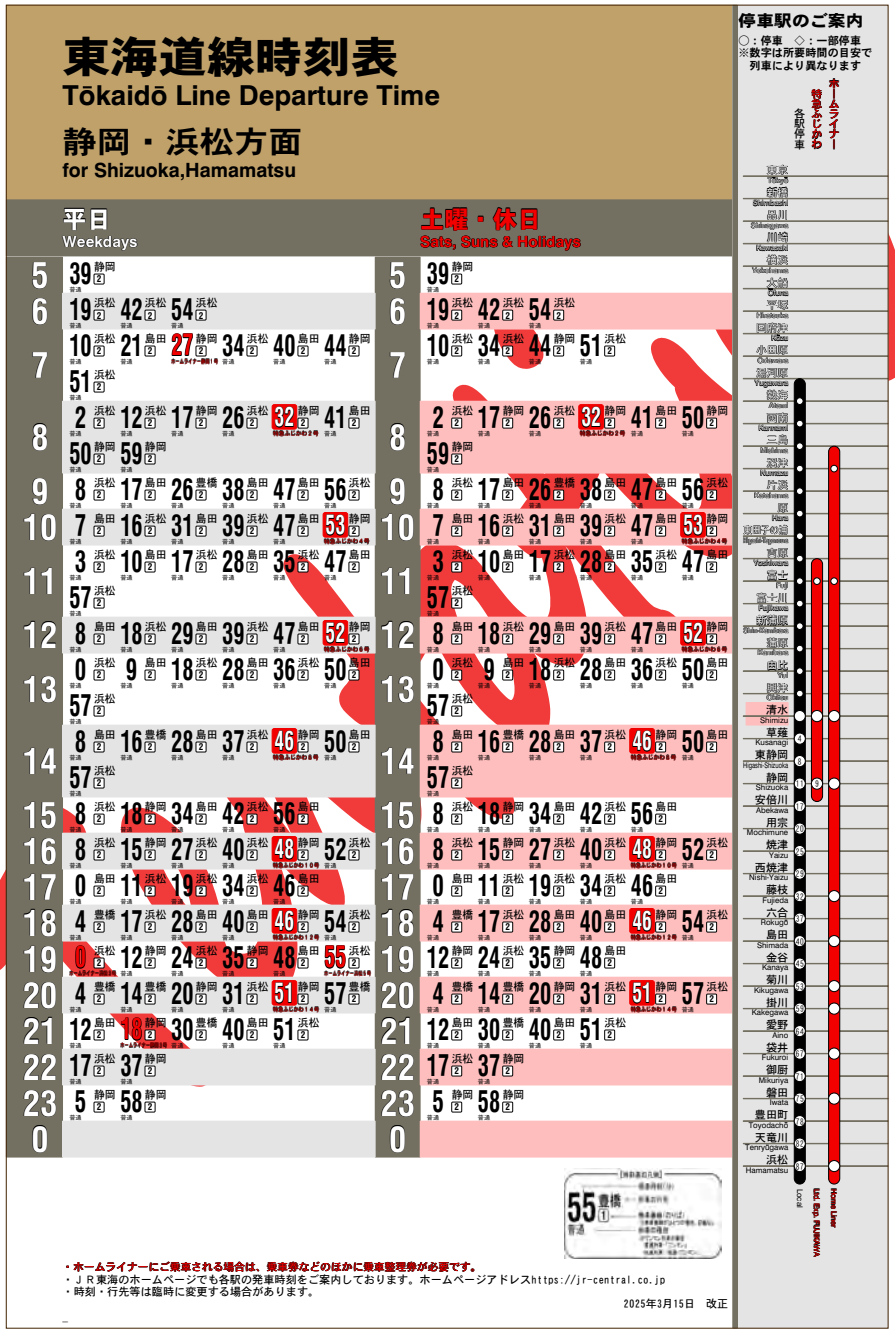
Train Time Table (JR Tokaido Line)

Shimizu station (near Conference Venue)



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Tōkaidō Line Departure Time
静岡・浜松方面
for Shizuoka,Hamamatsu

平日 Weekdays

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土曜・休日 Sats, Suns & Holidays

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8	2 浜	17 静	26 浜	32 静	41 島	50 静	59 静												
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辻堂 Tsujido
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金谷 Kanaya
菊川 Kikugawa
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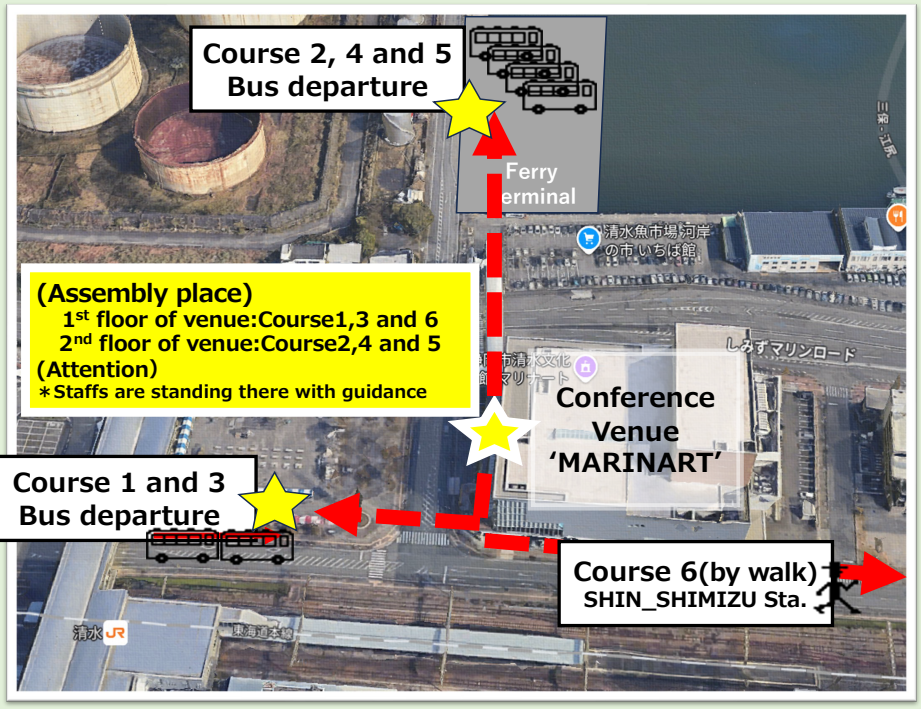
愛野 Aino
袋井 Fukuroi
御厨 Mikuriya
磐田 Iwata
豊田町 Toyodachō
天竜川 Tenryūgawa
浜松 Hamamatsu

Excursions

(Thursday afternoon)

Please check your excursion ticket in your name tag and meet at following assembly place on Thursday afternoon. Do not forget to bring your water bottle with green tea / water.

Assembly Place for Excursions



Course 1 From 1st floor of venue to assembly place
Training Center of JR Central
Assembly at 13:30
Departing at 13:40

Course 2 From 2nd floor of venue to assembly place
Yamanashi Prefectural Maglev Exhibition Center
Assembly at 13:35
Departing at 13:40

Course 3 From 1st floor of venue to assembly place
Hamaoka Nuclear power plant
Assembly at 13:40
Departing at 13:45

Course 4 From 2nd floor of venue to assembly place
Representative tourist destinations of Shizuoka city
Assembly at 13:40
Departing at 13:45

Course 5 From 2nd floor of venue to assembly place
MISHIMA Skywalk
Assembly at 13:45
Departing at 13:50

Course 6 From 1st floor of venue to assembly place
Walking at Central Shizuoka
Assembly Meeting up at 13:45
Departing at 13:50

Course 1. Visit the Training center of Central Japan Railway Company and Ohito Station

Course 2. Visit for Yamanashi Prefectural Maglev Exhibition Center and GOTEMBA PREMIUM OUTLETS

Course 3. Visit Hamaoka Nuclear Power Plant

Course 4. Visit the representative tourist destinations of Shizuoka city

Course 5. Visit Mishima Skywalk and GOTEMBA PREMIUM OUTLETS

Course 6. Walk at Shizuoka city and Visit Shizuoka SENGEN Shrine

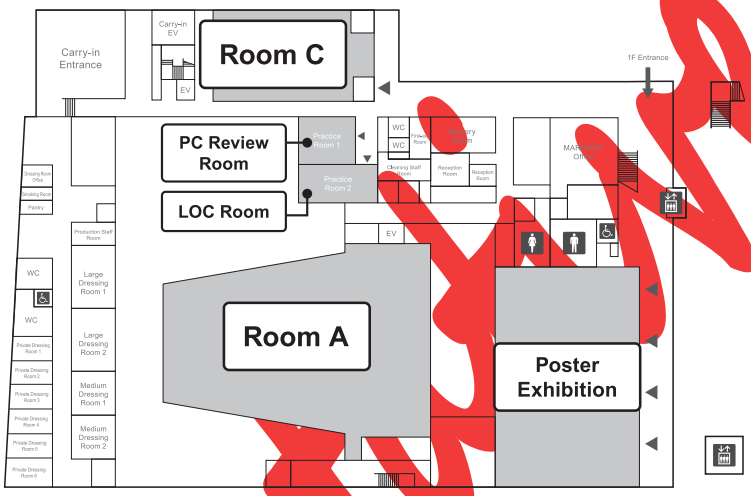
Tem

Conference Venue

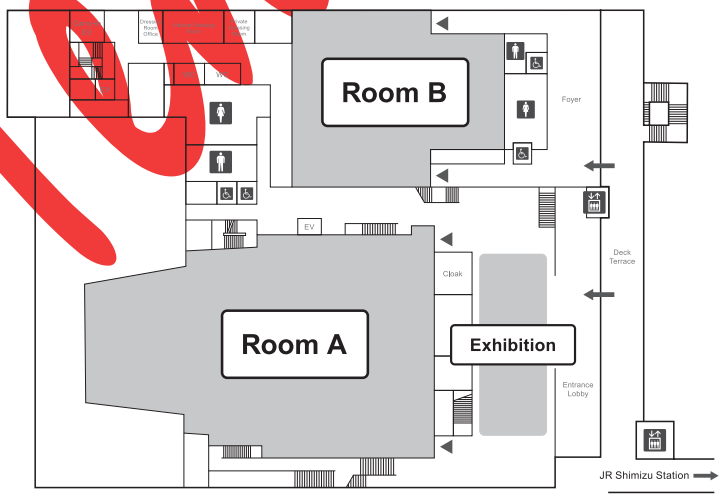
ICFRM-22 Floor Map

MARINART

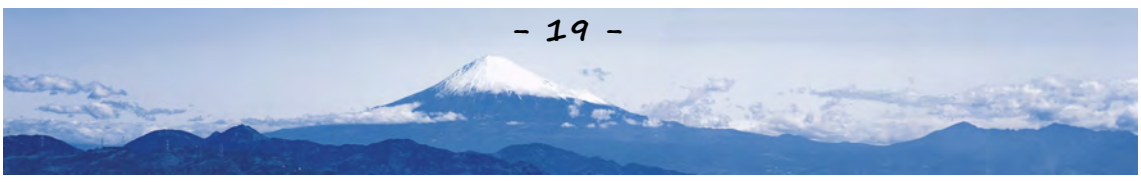
1F



2F



Tentative



Presentation Guideline

Guideline for Oral Presentations

1. The presentation times are as follows:

Keynote Lectures: 40 min + 5 min Q&A

Plenary presentations: 30 min + 5 min Q&A

Invited presentations: 25 min + 5 min Q&A

Oral presentations: 15 min + 5 min Q&A

2. To ensure the smooth progress of the conference sessions, please submit your presentation file **one day before the presentation**. The presentation files should be uploaded through USB drive at the “PC Review Desk” or “PC Review Room” as outlined below. Please inform us in advance if your presentation requires specific technical arrangements (e.g., audio playback, embedded video).

3. The accepted formats are PowerPoint (pptx) and PDF (pdf) and screen ratio is 16:9.

4. Previews of presentation data will be accepted as follows.

September 28 (Sun): 12:00 p.m. to 8:00 p.m.

September 29 (Mon): 9:15 a.m. to 4:30 p.m.

September 30 (Tue): 9:15 a.m. to 4:30 p.m.

October 1 (Wed): 9:15 a.m. to 12:30 p.m.

October 2 (Thu): 9:15 a.m. to 4:30 p.m.

October 3 (Fri): 9:15 a.m. to 10:00 a.m.

Location

September 28 (Sun): Hotel Associa Shizuoka 3rd Floor “PC Review Desk”

September 29 (Mon) to October 3 (Fri): Marinart(conference venue) 1st Floor “PC Review Room”

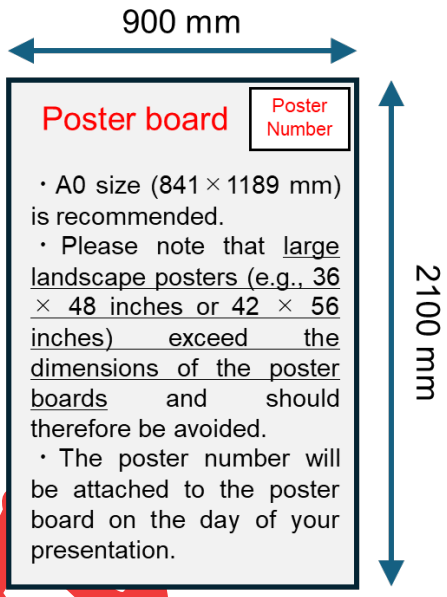
Tentative

Guideline for Poster Presentations

The presenter must be an author of the poster and a registered attendee.

Poster Format:

The allocated space per poster is approximately 900 mm × 2100 mm. There are no specific format restrictions, while A0 size is recommended. Please ensure that your poster does not exceed the allocated space, especially large landscape posters, which may extend beyond the board.



The 'title' and 'authors' should be clearly displayed in large print across the top of the poster.

Poster Display and Removal:

All posters must be displayed before lunchtime on the day of the presentation and removed at the end of the same day. Please make sure to check your assigned poster number on the board. Any posters remaining after the end of the session day will be removed and temporarily stored by the organizers. The posters not collected will be discarded. Pushpins will be available in the poster hall. Please return the pins after removing your poster.

Any posters not verified at check-in prior to display will be removed from the session.

Tentative

Program (Overview)

Tentative

	Sep. 28 (Sun)	
9:00		
9:30		
10:00		
10:30		
11:00		
11:30		
12:00		
12:30		
13:00	(13:00) Tutorial (Associa Shizuoka) SiC: L.L. Snead W: A. Hasegawa (15:00)	Carrying in Exhibits
13:30		
14:00		
14:30		
15:00	Break	
15:30	(15:30) Tutorial (Associa Shizuoka) RAFM & ODS: A. Bhattacharya (17:30)	
16:00		
16:30		
17:00		
17:30	(17:30) Reception (Associa Shizuoka) (19:30)	
18:00		
18:30		
19:00		

<table>
<tr><td></td><td colspan="3">Sep. 29 (Mon)</td><td rowspan="19">Exhibition</td></tr>
<tr><td>9:00</td><td colspan="3">Registration</td></tr>
<tr><td>9:30
10:00
10:30</td><td colspan="3">Room A
(9:30) Opening 29S
Welcome address Y. Hatano (Tohoku U, JPN)
Keynote 1 K. Sawada (MEXT, JPN)
Keynote 2 J.P. Allain (DOE, USA)
(11:20)</td></tr>
<tr><td>11:00</td><td colspan="3">Coffee break</td></tr>
<tr><td>11:30
12:00
12:30
13:00</td><td colspan="3">Room A
(11:50) 29PL
29PL1 M. Rieth (KIT, DEU)
29PL2 S. Nogami (A.L.M.T. Corp., JPN)
29PL3 M. Boleininger (UKAEA, GBR)
(13:35)</td></tr>
<tr><td>13:30
14:00</td><td colspan="3">(13:35)
Group Photo (Take 1)
Lunch
(14:40)</td></tr>
<tr><td>14:30
15:00
15:30
16:00
16:30</td><td>Room A
(14:40) 29A
29A1I A. Perrin
29A2I D. Hamaguchi
29A3C H. Yu
29A4C F. Canillas
29A5C D. Zhu
29A6C J. Riesch
(17:00)
Topic B</td><td>Room B
(14:40) 29B
29B1I J. Marian
29B2I C. Wang
29B3C L. Clowers
29B4C F. Luo
29B5C L. Gao
29B6C Y. Ma
(17:00)
Topic H</td><td>Room C
(14:40) 29C
29C1I W. Zhong
29C2I Y. Yuan
29C3C M. Zibrov
29C4C Y. Hoshino
29C5C E. Prestat
29C6C Y.Oya
(17:00)
Topic D</td></tr>
<tr><td>17:00
17:30
18:00
18:30</td><td colspan="3">Gallery hall
(17:00) P1
Poster (1) + coffee
(19:00)</td></tr>
<tr><td>19:00</td><td colspan="3"></td></tr>
</table>

Tentative

	Sep. 30 (Tue)			
9:00	Registration			
9:30 10:00 10:30	Room A (9:15) 30PL 30PL1 S.J. Zinkle (UTK, USA) 30PL2 F. Sun (HUT, CHN) 30PL3 N. Hashimoto (Hokkaido U, JPN) (11:00)			Exhibition
11:00	Coffee break			
11:30 12:00 12:30 13:00	Room A (11:20) 30A 30A1I K. Field 30A2I H. Tanigawa 30A3C A.Bhattacharya 30A4C Y. Wang 30A5C Z. Jiao 30A6C Y. Hatano (13:40) Topics K&I	Room B (11:20) 30B 30B1I G-H. Lu 30B2I F. Granberg 30B3C Y. Osetsky 30B4C B. Curtis 30B5C H. Xu 30B6C Z. Hu (13:40) Topic H	Room C (11:20) 30C 30C1I H. Gietl 30C2I Y. Lian 30C3C T. Oda 30C4C P. Dwivedi 30C5C J. You 30C6C M. Li (13:40) Topic B	
13:30 14:00	(13:40) Lunch (14:40)			
14:30 15:00 15:30 16:00 16:30	Room A (14:40) 30D 30D1I Y. Katoh 30D2I M. Dias 30D3C X. Wang 30D4C M. Zhao 30D5C X. Hu 30D6C M. Lloyd (17:00) Topic B	Room B (14:40) 30E 30E1I L. Hewitt 30E2I Y. Watanabe 30E3C S. Barwick 30E4C A. Albarodi 30E5C J. Geringer 30E6C P. Song (17:00) Topic H	Room C (14:40) 30F 30F1I M. Klimenkov 30F2I S. Kano 30F3C D. Sprouster 30F4C M. Ando 30F5C N. Wieber 30F6C Y. Li (17:00) Topics D&F	
17:00 17:30 18:00 18:30	Gallery hall (17:00) P2 Poster (2) + coffee (19:00)			
19:00				

tentative

Oct 1 (Wed)

9:00 Registration

9:30 – 10:30
Room A
(9:15) 01PL
01PL1 M. Serrano (CIEMAT, ESP)
01PL2 D. Terentyev (SCK CEN, BEL)
01PL3 C. Taylor (INL, USA)
(11:00)

11:00 Coffee break

11:30 – 12:00

Room A	Room B	Room C
(11:20) 01A	(11:20) 01B	(11:20) 01C
01A1I X. He	01B1I K. Mukai	01C1I J. Aktaa
01A2I G. Aiello	01B2I P. Charlesworth	01C2I N. Oono-Hori
01A3C E. Gaganidze	01B3C R. Zhang	01C3C L. Reali
(12:40)	(12:40)	(12:40)
Topic G	Topic C	Topic J

Exhibition

12:30 – 13:00 (12:40) Lunch (13:40)

13:30 – 19:00 (13:40) Excursion

	Oct. 2 (Thrs)			
9:00	Registration			
9:30 10:00 10:30	Room A (9:15) 02PL 02PL1 Y. Wang (Peking U, CHN) 02PL2 T. Koyanagi (ORNL, USA) 02PL3 T. Nozawa (QST, JPN) (11:00)			Exhibition
11:00	Coffee break			
11:30 12:00 12:30	Room A (11:20) 02A 02A1I A. Theodorou 02A2C D. Nguyen-Manh 02A3C X. Shen 02A4C H. Zhou 02A5C L.L. Snead (13:10) Topics C&D	Room B (11:20) 02B 02B1I J. Wang 02B2C C. Ruestes Vegas 02B3C J. Hou 02B4C A. Karcher 02B5C Z. Chen (13:10) Topic H	Room C (11:20) 02C 02C1I P-C. Simon 02C2C Y-R. Lin 02C3C A. Durif 02C4C M. Miyamoto 02C5C G. Zuo (13:10) Topic B	
13:00 13:30	(13:10) ***Group Photo (Take 2)*** Lunch (14:10)			
14:00 14:30 15:00 15:30 16:00	Room A (14:10) 02D 02D1I H. Oka 02D2I X. Mao 02D3C Y. Lan 02D4C J. Chen 02D5C J. Suh 02D6C G. Pintsuk (16:30) Topic A	Room B (14:10) 02E 02E1I A. Elliott 02E2I S. Kondo 02E3C B. Zhang 02E4C K. Samarkhanov 02E5C A. Fellman 02E6C M. Zheng (16:30) Topic H	Room C (14:10) 02F 02F1I M. Zmitko 02F2I W. Wang 02F3C Y. Yamamoto 02F4C P. Barron 02F5C E. Unterberg 02F6C D. Wilkinson (16:30) Topic E	
16:30 17:00 17:30 18:00	Gallery hall (16:30) P3 Poster (3) + coffee (18:30)			
18:30				
19:00	(19:30) Conference Dinner (Associa Shizuoka)			

Tentative

	Oct. 3 (Fri)			
9:00	Registration			
9:30 10:00 10:30	Room A (9:15) 03A 03A1I A. Zinovev 03A2C V. Chakin 03A3C M. Vrellou 03A4C E. Polselli 03A5C S. Lee (11:05) Topic A	Room B (9:15) 03B 03B1I T. Chikada 03B2C Q. Zhou 03B3C E. Curtis 03B4C J. Kim 03B5C R. Gaisin (11:05) Topic C	Room C (9:15) 03C 03C1I A. Schrell 03C2C P. Cara 03C3C A. Harte 03C4C D. Lee 03B5C Y. Soma (11:05) Topic F	Carrying out Exhibits
11:00	Coffee break			
11:30 12:00 12:30 13:00	Room A (11:25) 03D 03D1C J. Haley 03D2C A. Leide 03D3C M. Tunes 03D4C H. Tipping 03D5C Y. Yang (13:05) Topic A	Room B (11:25) 03E 03E1I Z. Bergstrom 03E2C A. Litnovsky 03E3C N. Garrivier 03E4C T. Hinoki 03E5C H. Zhu (13:15) Topics E&I	Room C (11:25) 03F 03F1I B. Gong 03F2C K. Collins 03F3C I. Palermo 03F4C A. Morrison 03F5C C. Zheng (13:15) Topic C	
13:30 14:00	(13:15) Summary and Closing (13:45)			
14:30 15:00 15:30 16:00 16:30 17:00 17:30 18:00 18:30 19:00				

Tentative

Program (Tutorial Lectures)

Neutron irradiation effects in RAFM & ODS steels for fusion reactor first-wall/blanket structures.

Arunodaya Bhattacharya

Fusion Energy University of Birmingham

2009 B.Sc. (honours) Physics, University of Delhi, Hans Raj College, India

2010 Master of Technology, University of Delhi, India

2011 Master of Nuclear Energy, University of Paris XI, Orsay, France

2015 PhD in Physics, University of Paris XI, France

Host laboratory: CEA-Saclay & CSNSM-Orsay

Tentative

Structural Ceramics and Composites for Nuclear Applications

Lance L Snead

Massachusetts Institute of Technology,

Irradiation Materials Sciences Group NRL

State University of New York Stony Brook

1986 Bachelor of Science in Physics, State University of New York

1988 Master of Science in Nuclear Engineering, Rensselaer Polytechnic Institute

1992 Doctor of Philosophy in Nuclear Engineering, Rensselaer Polytechnic Institute

History and current status of irradiation studies of W for fusion applications.

Akira Hasegawa

Institute for Materials Research, Tohoku University

1984 Graduate School of Engineering, Tohoku University: Ph.D

1992 National Research Institute for Metals, Tsukuba-Laboratories, Japan

2022 Faculty of Engineering, Tohoku University, Japan

2022- Institute for Material Research, Tohoku University, Japan

Tentative

Program (Oral presentations)

Tentative

29th September 2025 (Monday)

Time 9:30-11:20 Room A Chairs: Yuji Hatano (Tohoku University), T. Nagasaka (National Institute for Fusion Science)			
9:30-9:50	**Welcome address**	**Yuji Hatano** Tohoku University	
9:50-10:35	**Keynote1**	**Kazuhiro Sawada** MEXT	Fusion Energy Innovation Strategy
10:35-11:20	**Keynote2**	**Jean Paul Allain** DOE	Building Bridges – U.S. Fusion Energy Sciences Vision, Strategy, and Roadmap

Time 11:50-13:35 Room A Chairs: G. Pintsuk (Forschungszentrum Julich GmbH), S. Zinkle (University of Tennessee)			
11:50-12:25	**29PL1**	**Michael Rieth** Karlsruhe Institute of Technology (KIT)	Status and perspective of the European neutron irradiation program
12:25-13:00	**29PL2**	**Shuhei Nogami** A.L.M.T. Corp.	Development of Fusion-Grade Tungsten Materials at Industrial Scale
13:00-13:35	**29PL3**	**Max Boleininger** UK Atomic Energy Authority	Developing predictive materials models for the virtual fusion reactor from atomistic simulations - proof of concept application to irradiation creep

Tentative

Time 14:40-17:00 Room A Chairs: D. Sprouster (Stony Brook University), E. Tejado (Universidad Politecnica de Madrid)			
14:40-15:10	**29A1**	**Alice Perrin** Oak Ridge National Laboratory	Neutron-irradiation effects on the mechanical and electrical properties of bimodally precipitate strengthened Cu alloys
15:10-15:40	**29A2**	**Dai Hamaguchi** National Institutes for Quantum Science and Technology	Advanced In-situ SEM/EBSD Analysis of CuCrZr alloys: Impact of Neutron Irradiation on Deformation Mechanisms
15:40-16:00	**29A3**	**Hao Yu** Tohoku University	Development of low-energy mechanical alloying technology to enable mass production of ODS-Cu alloys for fusion divertors
16:00-16:20	**29A4**	**Francisco Canillas** CIEMAT - National Fusion Laboratory	A multi-scale microstructural characterization of additively manufactured ITER-Grade CuCrZr with superior properties
16:20-16:40	**29A5**	**Da huan Zhu** Institute of Plasma Physics, Academy of Sciences	Application of ITER-like W/Cu PFCs for high heat flux components in EAST
16:40-17:00	**29A6**	**Johann Riesch** Max Planck Institute for Plasma Physics	Tungsten fibre-reinforced tungsten composites - materials qualification at the high heat flux test facility GLADIS

Tentative

Time 14:40-17:00 Room B Chairs: Y. Watanabe (National Institutes for Quantum Science and Technology), W. Setyawan (Pacific Northwest National Laboratory)			
14:40-15:10	**29B1**	**Jaime Marian** UCLA	Computational materials science evaluation of the D/Li-stripping neutron source as a prototypical fusion materials testing facility
15:10-15:40	**29B2**	**Chenxu Wang** Peking University	Helium-Hydrogen Synergistic Effects in Structural Materials in Fusion Reactors
15:40-16:00	**29B3**	**Logan N. Clowers** University of Michigan	Understanding the Role of Hydrogen on Cavity Evolution in Triple Ion Irradiated RAFM Steels
16:00-16:20	**29B4**	**Fengping Luo** Peking University	Multiscale studies on the synergistic effects of helium and hydrogen on cavity evolution in ferritic steels in the fusion environment
16:20-16:40	**29B5**	**Liang Gao** Institute of Fusion energy and Nuclear Waste Management: Plasmaphysik, Forschungszentrum Julich GmbH	Unraveling damage accumulation due to pre-existing irradiation defects interaction with newly generated vacancies upon low-energy H plasma exposure
16:40-17:00	**29B6**	**Yiyi Ma** University of Science and Technology of China	Molecular dynamics study on hydrogen-helium induced hardening effect in RAFM steels

Tentative

Time 14:40-17:00 Room C Chairs: E. Prestat (UK Atomic Energy Authority), S. Masuzaki (National Institute for Fusion Science)			
14:40-15:10	**29C1**	**Weicheng Zhong** Oak Ridge National Laboratory	Simultaneous Neutron Irradiation and Hydrogen exposure on Tungsten: Hydrogen Retention
15:10-15:40	**29C2**	**Yue Yuan** Beihang University	Low Hydrogen Isotope Retention and High Irradiation Resistance in Columnar-Grained Tungsten Prepared via Chemical Vapor Deposition for Fusion Applications
15:40-16:00	**29C3**	**Mikhail Zibrov** Max Planck Institute for Plasma Physics	Investigation of deuterium trapping by helium bubbles in tungsten
16:00-16:20	**29C4**	**Yuzuka Hoshino** Shizuoka University	Effect of He irradiation on hydrogen isotope retention in W-Re
16:20-16:40	**29C5**	**Eric Prestat** UK Atomic Energy Authority	Oxidation of tungsten and its implications to tritium retention in decommissioning, maintenance and accident scenarios in future devices
16:40-17:00	**29C6**	**Yasuhisa Oya** Shizuoka University	Impact on He seeding for D and He retention in W alloys by DPE D+He plasma exposure

Tentative

30th September 2025 (Tuesday)

Time 9:15-11:00 Room A Chairs: M. Caturla (Universidad de Alicante), H. Tanigawa (National Institutes for Quantum Science and Technology)			
9:15-9:50	**30PL1**	**Steven J. Zinkle** University of Tennessee	Mechanical Properties and High Temperature Helium Embrittlement Mechanisms in Irradiated Alloys
9:50-10:25	**30PL2**	**Fei Sun** Hefei University of Technology	Hydrogen isotope retention and removal in plasma-facing materials
10:25-11:00	**30PL3**	**Naoyuki Hashimoto** Hokkaido University	Role of grain boundaries on irradiation damage evolution in a reduced activation ferritic/martensitic steel F82H

Time 11:20-13:40 Room A Chairs: Y. Katoh (Oak Ridge National Laboratory), N. Hashimoto (Hokkaido University)			
11:20-11:50	**30A1**	**Kevin G Field** University of Michigan	Towards a Comprehensive Understanding of MX-Type Precipitate Behavior Under Irradiation in Castable Nanostructured Alloys
11:50-12:20	**30A2**	**Hiroyasu Tanigawa** National Institutes for Quantum Science and Technology	F82H Development - Past and Future Issues

Tentative

12:20-12:40	**30A3**	**ARUNODAYA BHATTACHARYA** University of Birmingham	Effect of helium on low-temperature hardening/embrittlement (LTHE) in neutron irradiated isotopically tailored RAFM steels
12:40-13:00	**30A4**	**Yanli Wang** Oak Ridge National Laboratory	Improving the balance between strength and DBTT in industrial-scale production of Castable Nanostructure Steels
13:00-13:20	**30A5**	**Zhijie Jiao** University of Michigan	Ion emulation of neutron irradiation effects on ferritic-martensitic steels
13:20-13:40	**30A6**	**Yuji Hatano** Tohoku University	Accomplishments of the Japan-US FRONTIER Project -Fusion Research Oriented to Neutron Irradiation Effects and Tritium Behavior at Material Interfaces-

Time 11:20-13:40 Room B Chairs: M. Boleininger (UK Atomic Energy Authority), Y. Matsukawa (Kumamoto University)			
11:20-11:50	**30B1**	**Guang-Hong Lu** Beihang University	Development of a Multiscale Modeling Platform for Fusion Neutron Irradiation Effects of Materials
11:50-12:20	**30B2**	**Fredric Granberg** University of Helsinki	Multiscale modelling of defect evolution in fusion relevant materials
12:20-12:40	**30B3**	**Yuri Osetsky** ORNL	Computational framework for modeling TiB_2 under fusion conditions
12:40-13:00	**30B4**	**Beatrix India Curtis** University of Oxford	Crystal plasticity finite element modelling of thermal and irradiation creep in fusion structural materials

Tentative

13:00-13:20	**30B5**	**Haixuan Xu** University of Tennessee Knoxville	The Influence of Stacking Fault Tetrahedra on Void Swelling in Irradiated Copper
13:20-13:40	**30B6**	**Zhiwei Hu** CNRS	New Insights into Quantifying Vacancy Distributions in Self-Ion Irradiated Tungsten: A Combined Experimental and Computational Study

Time 11:20-13:40 Room C Chairs: X. Hu (Sichuan University), M. Yajima (National Institute for Fusion Science)			
11:20-11:50	**30C1**	**Hanns Gietl** IPP Garching	Understanding the limitations and potential of micro tensile testing of tungsten and needs for crystal plasticity modelling
11:50-12:20	**30C2**	**Youyun Lian** Southwestern Institute of Physics	Comparison of the mechanical properties and recrystallization temperature of W-K materials prepared by two plastic-deformation technologies
12:20-12:40	**30C3**	**Takuji Oda** Seoul National University	Atomistic simulations of hydrogen dynamics on tungsten surfaces using machine learning potential
12:40-13:00	**30C4**	**Prashant Dwivedi** Czech Technical University in Prague	High-Velocity Dust Impact on Tungsten Fuzz as a Plasma-Facing Component in Fusion Reactors
13:00-13:20	**30C5**	**Jeong-Ha You** Max Planck Institute for Plasma Physics	Neutron-based techniques for stress measurement and defect inspection: round robin tests for high-heat-flux components

Tentative

13:20-13:40	**30C6**	**Mingshen Li** Tsinghua University	Crack elimination and mechanical property of 3D printing pure tungsten and tungsten alloys

Time 14:40-17:00 Room A Chairs: P. Jenus Belec (Jozef Stefan Institute), M. Fukuda (National Institutes for Quantum Science and Technology)			
14:40-15:10	**30D1**	**Yutai Katoh** Oak Ridge National Laboratory	EFFECTS OF HFIR RB-19J THERMAL NEUTRON-SHIELDED IRRADIATION IN TUNGSTEN METALS AND ALLOYS
15:10-15:40	**30D2**	**Marta Silva Dias** Instituto de Plasmas e Fusao Nuclear (IPFN), Instituto Superior Tecnico, Universidade de Lisboa	Study of W- based high entropy alloys for nuclear fusion applications
15:40-16:00	**30D3**	**Xing Li Wang** College of Physics and Optoelectronic Engineering, Shenzhen University	Recent progress on coating-based W/steel first wall structure through detonation spraying in Shenzhen University
16:00-16:20	**30D4**	**Mingzhong Zhao** National Institute for Fusion Science (NIFS)	LHD divertor plasma induced surface modifications on unirradiated tungsten and pre-irradiated tungsten
16:20-16:40	**30D5**	**Xunxiang Hu** Sichuan University	Nuclear Transmutation and its Impact on In-Service Performance of Tungsten
16:40-17:00	**30D6**	**Matthew James Lloyd** University of Birmingham	Ductlisation of Tungsten using Rolling and Rhenium Alloying

Tentative

Time 14:40-17:00 Room B Chairs: Z. Zaurbekova (Institute of Nuclear Physics), S. Dudarev (UK Atomic Energy Authority)			
14:40-15:10	**30E1**	**Luke Hewitt** United Kingdom Atomic Energy Authority	Irradiation temperature dependence of hardening and microstructure in Eurofer97/3 using thermal gradients
15:10-15:40	**30E2**	**Yoshiyuki Watanabe** National Institutes for Quantum Science and Technology	Multiscale computational modeling of the helium effect on cavity swelling in F82H steel under different irradiation fields
15:40-16:00	**30E3**	**Sophie R Barwick** The University of Sheffield	Effects of Mn content and normalisation temperature on radiation induced clustering in ferrous alloys for fusion.
16:00-16:20	**30E4**	**Abdulrahman Albarodi** Max Planck Institute for Plasma Physics	Effect of high displacement damage at different dose rates and temperatures on Deuterium retention in Eurofer97
16:20-16:40	**30E5**	**Josina Wilhelmina Geringer** Oak Ridge National Laboratory	The role of temperature in radiation hardening of neutron-irradiated Fe-(8-10%)Cr model alloys
16:40-17:00	**30E6**	**Peng Song** Institute of Material Research, Tohoku University	Characterization of the radiation hardenings of Fe9Cr and Eurofer97 by means of ball-indentation and micropillar compression

Tentative

Time 14:40-17:00 Room C Chairs: R. Gonzalez-Arrabal (Universidad Politecnica de Madrid), T-S. Byun (Oak Ridge National Laboratory)			
14:40-15:10	**30F1**	**Michael Klimenkov** Karlsruhe Institute of Technology	TEM characterization and statistical analysis of radiation-induced defect microstructure in tungsten and tungsten alloys
15:10-15:40	**30F2**	**Sho Kano** National Institutes for Quantum and Radiological Science and Technology (QST)	Feasibility study of a TEM/STEM analysis for the characterization of irradiation-induced heterogeneous microstructures
15:40-16:00	**30F3**	**David Sprouster** Stony Brook University	Synchrotron-based characterization of radiation damage in Ultra high temperature ceramics
16:00-16:20	**30F4**	**Masami Ando** National Institutes for Quantum Science and Technology	Micro Fracture Strength of Neutron Irradiated F82H by using μ-Tensile Testing
16:20-16:40	**30F5**	**Natalie Wieber** Savannah River National Laboratory	Tensile and Permeation Properties of Austenitic Stainless Steel Tubing with Internal Hydrogen/Tritium
16:40-17:00	**30F6**	**Yu Li** Institute of Plasma Physics, Chinese Academy of Sciences	Anisotropic irradiation hardening in MeV-ion damaged tungsten

Tentative

1st October 2025 (Wednesday)

Time 9:15-11:00 Room A Chairs: Y-H. Park (Korea Institute of Fusion Energy), A. Perrin (Oak Ridge National Laboratory)			
9:15-9:50	**01PL1**	**Marta Serrano** CIEMAT	Cross-cutting material issues for fusion and fission nuclear systems: connection with the unified European strategic research agenda of CONNECT-NM
9:50-10:25	**01PL2**	**Dmitry Terentyev** SCK CEN	Qualification of EUROFER97 base metal and weldments for ITER TBM: EUROfusion contribution
10:25-11:00	**01PL3**	**Chase N Taylor** Idaho National Laboratory	Tritium production from thermal neutron irradiated solid breeder materials

Time 11:20-12:40 Room A Chairs: M. Seranno (CIEMAT), T. Nozawa (National Institutes for Quantum Science and Technology)			
11:20-11:50	**01A1**	**Xinfu He** China Institute of Atomic Energy	Standardization of Small-Scale Mechanical Testing for Fusion Reactor Materials: Progress and Results
11:50-12:20	**01A2**	**Giacomo Aiello** EUROfusion	Development of C&S for fusion in-vessel components: from ITER to DEMO
12:20-12:40	**01A3**	**Ermile Gaganidze** Karlsruhe Institute of Technology (KIT), Institute for Applied Materials	Development and validation of small specimen testing technique for ductile fracture toughness

Time 11:20-12:40 Room B Chairs: J. Leys (Karlsruhe Institute of Technology), L.L. Snead (Stony Brook University)			
11:20-11:50	**01B1**	**Keisuke Mukai** National Institute for Fusion Science	Corrosion kinetics of parent and welded F82H by LTZO ceramic breeder pebbles
11:50-12:20	**01B2**	**Pedr William Charlesworth** University of Oxford	Phase Stability and Irradiation Resistance of the Octalithium Ceramics (Li8MO6, M = Zr, Sn, Pb and Ce) for Tritium Breeding
12:20-12:40	**01B3**	**Ruijie Zhang** Department of Nuclear Engineering, Seoul National University,	Molecular dynamics simulations of tritium diffusivity in lithium orthosilicate crystals using a machine learning potential

Time 11:20-12:40 Room C Chairs: Y. Yamamoto (Oak Ridge National Laboratory), M. Rieth (Karlsruhe Institute of Technology)			
11:20-11:50	**01C1**	**Jarir Aktaa** Karlsruhe Institute of Technology (KIT)	Progress and mid-term achievements of the INNUMAT project
11:50-12:20	**01C2**	**Naoko OONO-HORI** Faculty of Engineering, Yokohama National University	Development of FeCrAl-based Oxide Dispersion Strengthened (FeCrAl-ODS) Alloy Free from Ageing Embrittlement
12:20-12:40	**01C3**	**Luca Reali** UK Atomic Energy Authority	Irradiation effects in the mechanical model of a full fusion tokamak

Tentative

2nd October 2025 (Thursday)

Time 9:15-11:00 Room A Chairs: R. Kasada (Tohoku University), Q. Huang (Hefei Institutes of Physical Science, Chinese Academy of Sciences)			
9:15-9:50	**02PL1**	**Yugang Wang** School of Physics, Peking University	China's Vision for R&D of Fusion Reactor Materials at Different Stages
9:50-10:25	**02PL2**	**Takaaki Koyanagi** Oak Ridge National Laboratory	Advancements and Perspectives in the Development of SiC-Based Materials for Fusion Applications
10:25-11:00	**02PL3**	**Takashi Nozawa** National Institutes for Quantum Science and Technology	Development and Irradiation Effects of Welds and Joints in Reduced-Activation Ferritic/Martensitic Steel F82H

Time 11:20-13:10 Room A Chairs: F. Sun (Hefei University of Technology), K. Katayama (Kyushu University)			
11:20-11:50	**02A1**	**Andreas Theodorou** Max Planck Institute for Plasma Physics	Retention and release of deuterium in displacement-damaged and helium-implanted EUROFER97 at reactor-operating temperatures
11:50-12:10	**02A2**	**Duc Nguyen-Manh** United Kingdom Atomic Energy Authority	Detritiation in tungsten and its oxides: Prediction from multi-scale materials modelling and engineering implication

Tentative

12:10-12:30	**02A3**	**Xin Shen** Institute of Plasma Physics, Hefei Institutes of Physical Science, Chinese Academy of Sciences	Influence of reproducible trapping consideration and W-RAFM bonding region on hydrogen isotope transport behavior in the first wall
12:30-12:50	**02A4**	**HaiShan Zhou** Institute of plasma physics, Chinese Academy of Sciences	Mass production of boron-containing steels for fusion shielding application
12:50-13:10	**02A5**	**Lance Snead** Stony Brook University	Development of Grey Cryo-Steels-Combining Structural and Shield Function to Enable Compact Fusion Design

Time 11:20-13:10 Room B Chairs: D. Mason (UK Atomic Energy Authority), J .Geringer (Oak Ridge National Laboratory)			
11:20-11:50	**02B1**	**Jing Wang** Hefei University of Technology	Evolution of Vacancy Cluster, Precipitates, and Hardening in Binary Tungsten Alloys after Irradiation at Elevated Temperatures
11:50-12:10	**02B2**	**Carlos Javier Ruestes Vegas** Instituto de Fusion Nuclear "Guillermo Velarde" - Universidad Politecnica de Madrid	Mechanical properties of nanoporous tungsten under irradiation - modelling and experiments
12:10-12:30	**02B3**	**Jie Hou** Hunan University	Hydrogen-Stabilized Dislocation Interactions Drive Blister Formation in Tungsten

12:30-12:50	**02B4**	**Annemarie Karcher** Technical University of Munich, Max Planck Institute for Plasma Physics	Experimental investigation of the flux-dependence of He clustering in polycrystalline and monocrystalline W
12:50-13:10	**02B5**	**Ze Chen** University of Science and Technology of China	The effect of grain boundary on hydrogen isotopes diffusion in tungsten

Time 11:20-13:10 Room C Chairs: J-H. You (Max Planck Institute for Plasma Physics), D. Hamaguchi (National Institutes for Quantum Science and Technology)			
11:20-11:50	**02C1**	**Pierre-Clement Simon** Idaho National Laboratory	Leveraging the Fusion ENergy Integrated multiphys-X (FENIX) Computational Tool for Fusion Materials Research
11:50-12:10	**02C2**	**Yan-Ru Lin** Oak Ridge National Laboratory	Perspectives and challenges of ultra-high temperature ceramics for fusion plasma-facing applications
12:10-12:30	**02C3**	**Alan DURIF** CEA	Thermal ageing study of liquid metal based plasma facing material: tungsten capillary porous system wetted by tin
12:30-12:50	**02C4**	**Mitsutaka Miyamoto** Shimane University	Detailed analysis of gas retention behavior of bubbles in fusion reactor materials using in-situ TEM and STEM-EELS
12:50-13:10	**02C5**	**Guizhong Zuo** Institute of Plasma Physics, Hefei Institutes of Physical Science, Chinese Academy of Sciences	Low-Z materials coating to enhance long pulse H mode plasma performance in EAST

Tentative

Time 14:10-16:30 Room A Chairs: S. Noh (Pukyong National University), N. Oono-Hori (Tohoku University)			
14:10-14:40	**02D1**	**Hiroshi Oka** Hokkaido University	Swelling resistance in reduced activation ODS medium-entropy alloys under triple ion irradiation
14:40-15:10	**02D2**	**Xiaodong Mao** Shanghai University	11B isotope enhanced strength-ductility synergy in ODS steel
15:10-15:30	**02D3**	**Yubin Lan** Tsinghua University	A novel RAFM steel with superior high-temperature creep lifetime prepared by Additive manufacturing
15:30-15:50	**02D4**	**Jiming Chen** Southwestern Institute of Physics	Fabrication of ODS steel via a Z type Homogenized Blending (ZHB) route
15:50-16:10	**02D5**	**Joowon Suh** Korea Atomic Energy Research Institute, Seoul National University	Investigation of in-situ nano oxide formation in ODS F/M steels produced by L-DED
16:10-16:30	**02D6**	**Gerald Pintsuk** Forschungszentrum Julich GmbH	Structural material development and down-selection process for in-vessel components within the European fusion materials program

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Time 14:10-16:30 Room B Chairs: J. Trelewicz (Stony Brook University), C. Zhang (Institute of Modern Physics, Chinese Academy of Sciences)			
14:10-14:40	**02E1**	**Alicia M. Elliott** Massachusetts Institute of Technology	Toward emulation of neutron damage with ions: effects of helium on microstructural evolution in ion-irradiated V-4Cr-4Ti alloy (Heat US832864)
14:40-15:10	**02E2**	**Sosuke Kondo** Institute for Materials Research, Tohoku University	High-Dose Irradiation Response of SiC Fibers
15:10-15:30	**02E3**	**Baoshan Baoshan Zhang Zhang** Peking University	Mechanistic exploration of insulation materials low-temperature irradiation damage through multiscale simulation coupled with experimental validation
15:30-15:50	**02E4**	**Kuanysh Samarkhanov** Institute of Atomic Energy Branch of the National Nuclear Center of the Republic of Kazakhstan	Impact of High Neutron Fluence on Mechanical Properties of FBG Sensors for Nuclear Applications
15:50-16:10	**02E5**	**Aslak Fellman** University of Helsinki	Radiation damage simulations of FCC high-entropy alloys using machine-learning potentials
16:10-16:30	**02E6**	**Mingjie Zheng** Hefei Institutes of Physical Science, Chinese Academy of Sciences	Accelerated design of novel RAFM steels and reduced-activation high entropy alloys for fusion applications by machine learning

Tentative

Time 14:10-16:30 Room C Chairs: H. Gwon (Korea Institute of Fusion Energy), B. Garcia-Diaz (Savannah River National Laboratory)			
14:10-14:40	**02F1**	**Milan Zmitko** Fusion for Energy (F4E)	Progress and Future Perspectives in the European Test Blanket Modules (TBM) Manufacturing
14:40-15:10	**02F2**	**Wanjing Wang** Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP)	Research Progress in Manufacturing and Assembly of Tritium Breeding Blanket Component at ASIPP
15:10-15:30	**02F3**	**Yukinori Yamamoto** Oak Ridge National Laboratory	Newly proposed PWHT-free design for reduced-activation bainitic steels
15:30-15:50	**02F4**	**Paul J Barron** Kyoto Fusioneering	Progress towards a mock-up vanadium alloy blanket
15:50-16:10	**02F5**	**Ezekial Unterberg** Oak Ridge National Laboratory	Tungsten Coatings on an Actively-cooled, Additively-manufactured, Reciprocating Probe in the Long-pulse WEST Device
16:10-16:30	**02F6**	**Daniel Patrick Wilkinson** Department of Materials University of Oxford	Development and Testing of a Novel FAST-Diffusion Bonding Process for Joining Eurofer97 Steel to Tungsten for Plasma Facing Components

Tentative

3rd October 2025 (Friday)

Time 9:15-11:05 Room A Chairs: J. Stubbins (University of Illinois at Urbana-Champaign), M. Vrellou (Karlsruhe Institute of Technology)			
9:15-9:45	**03A1**	**Aleksandr Zinovev** Belgian Nuclear Research Centre (SCK CEN)	EUROfusion characterisation programme of CuCrZr alloy for the DEMO divertor heat sink: mechanical and thermophysical properties
9:45-10:05	**03A2**	**Vladimir Chakin** Karlsruhe Institute of Technology, Institute for Applied Materials	Tensile and Charpy properties of EUROFER97-3 steel after neutron irradiation at 325 °C and 550 °C to damage doses of 19-24 dpa
10:05-10:25	**03A3**	**Maria Vrellou** KIT	Multiscale Analysis of Hydrogen Isotope Effects on EUROFER97
10:25-10:45	**03A4**	**Ethan M Polselli** University of Michigan	Impact of Manufacturing Processing Control on Irradiation Precipitate Stability and Swelling of Advanced Steels
10:45-11:05	**03A5**	**Sumin Lee** Tohoku University	Local deformation behavior in proton-irradiated reduced-activation ferritic/martensitic steels under nanoindentation

Tentative

Time 9:15-11:05 Room B Chairs: C. Taylor (Idaho National Laboratory), I. Kenzhina (Satbayev University)			
9:15-9:45	**03B1**	**Takumi Chikada** Shizuoka University	Development of repair technology for functional ceramic coating

9:45-10:05	**03B2**	**Qilai Zhou** Wuhan University of Technology	Tritium release and defect annihilation of neutron-irradiated lithium ceramics with Pb addition
10:05-10:25	**03B3**	**Emily Curtis** UKAEA	Thermal Stability of Thin-Film Coatings for Breeder Blankets
10:25-10:45	**03B4**	**JAEHWAN KIM** National Institutes for Quantum Science and Technology	Current R&Ds on Advanced Breeding Functional Materials for JA DEMO activities
10:45-11:05	**03B5**	**Ramil Gaisin** Karlsruhe Institute of Technology	Progress and Challenges in the Development of Neutron Multiplier Materials for Breeding Blankets

Time 9:15-11:05 Room C Chairs: T. Hinoki (Kyoto University), X. Chen (Oak Ridge National Laboratory)			
9:15-9:45	**03C1**	**Adrian Schrell** Oak Ridge National Laboratory	Materials Irradiations in HFIR for Fusion Relevant Materials
9:45-10:05	**03C2**	**Philippe CARA** Fusion for Energy	Overview of the DONES Programme
10:05-10:25	**03C3**	**Allan Harte** UK Atomic Energy Authority	Synergistic irradiation-thermal-mechanical loading in fusion steels: First results from a new facility
10:25-10:45	**03C4**	**Dong Won LEE** Korea Atomic Energy Research Institute	Current Status and Plan of Fusion Material Research with Neutron Sources and Heavy Ion Irradiation Facility

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10:45-11:05	**03C5**	**Yasutaka Soma** Japan Atomic Energy Agency	Development of an Irradiation Test System for Basic Material Studies

Time 11:25-13:05 Room A Chairs: Y. Yang (Oak Ridge National Laboratory), D. Terentyev (SCKCEN)			
11:25-11:45	**03D1**	**Jack Haley** UKAEA	Development of Boron-strengthened B-RAFM Fusion Steels
11:45-12:05	**03D2**	**Alex Leide** UK Atomic Energy Authority	Developing and testing silicon carbide composites for fusion-relevant conditions
12:05-12:25	**03D3**	**Matheus A. Tunes** Department of Metallurgy, Chair of Nonferrous Metallurgy, Montanuniversitat Leoben	The saga of nonferrous metallurgy in the quest of new radiation-resistant fusion materials beyond pure tungsten
12:25-12:45	**03D4**	**Hannah Tipping** University of Bristol	Effect of disruption heat loads on tungsten-steel fusion composites for tokamak front wall
12:45-13:05	**03D5**	**Ying Yang** Oak Ridge National lab	Nanoprecipitate strengthened Vanadium alloys for Fusion First Wall and Blanket Application

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Time 11:25-13:15 Room B Chairs: E. Unterberg (Oak Ridge National Laboratory), W. Wang (Institute of Plasma Physics, Chinese Academy of Sciences)			
11:25-11:55	**03E1**	**Zachary Bergstrom** General Atomics	Developing Innovative Material Solutions for Fusion Pilot Plants at General Atomics
11:55-12:15	**03E2**	**Andrey Litnovsky** Forschungszentrum Juelich	Advanced plasma-facing components for future fusion power plants: an overview and first results of the “a-SMART-FIT” project
12:15-12:35	**03E3**	**Natan Garrivier** Paul Scherrer Institute	Multimodal Synchrotron and Neutron Characterization of PFC Tungsten-Steel Interfaces in Laser Additive Manufacturing
12:35-12:55	**03E4**	**Tatsuya Hinoki** Kyoto University	Irradiation Effect on Tungsten/Silicon Carbide Interface Joined by Hot-press
12:55-13:15	**03E5**	**Huiping Zhu** North China Electric Power University	Numerical Simulations of the Effects of Residual Surface Stress Distribution on Micro-Hardness Data in Fusion Reactor Structural Materials

Time 11:25-13:15 Room C Chairs: P. Charlesworth (University of Oxford), T. Tanaka (National Institute for Fusion Science)			
11:25-11:55	**03F1**	**Baoping Gong** Southwestern Institute of Physics	R&D of advanced functional materials and engineering qualification for the fusion solid blanket

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11:55-12:15	**03F2**	**Kailee Jewelle Collins** University of Tennessee, Knoxville	Investigating Deuterium Retention and Transport Behavior in Neutron-Irradiated Tungsten by Gas-Driven Permeation, Laser Ablation Mass Spectrometry, and Thermal Desorption Spectroscopy
12:15-12:35	**03F3**	**IOLE PALERMO** CIEMAT, Fusion Technology Division	Water Lead Ceramic Breeder (WLCB) Breeding Blanket (BB) for the EU DEMO: neutronic campaigns for the optimization of its performances
12:35-12:55	**03F4**	**Alasdair Morrison** Oxford Sigma	Experimental validation for the qualification of lithium ceramic breeders.
12:55-13:15	**03F5**	**Ce Zheng** Northwestern Polytechnical University	Innovative FCI Design with Al_2O_3-SiC Fiber-Reinforced CMC Composites: Enhanced Thermal Insulation, Maintained Electrical Insulation, and High-Temperature Mechanical Stability

Tentative

Program (Poster presentations)

Tentative

29th September 2025 (Monday)

Time 17:00-19:00 Gallery Room Chairs: A. Perrin (ORNL), M. Wirtz (FZJ), D. Hamaguchi (QST), K. G. Field (U Michigan), L. Peng (USTC)		
P1-001	**Chansun Shin** Myongji University	Irradiation Hardening and Microstructural Evolution of K-RAFM Steel Under Fe-Ion Irradiation up to 20 dpa
P1-002	**Kotaro Seki** QST	Effect of Proton Irradiation under Tensile Stress on Irradiation Hardening Behavior of F82H
P1-003	**Yukito Taoka** Laboratory of Advanced Materials	On irradiation damage microstructure around grain boundaries in the reduced activation ferritic/martensitic steel: F82H
P1-004	**Xingjian Zhao** Institute of Plasma Physics, Hefei Institutes of Physical Science, Chinese Academy of Sciences	Machine Learning-Based Jiles-Atherton Modeling of Ferrite Materials Under Strong Stray Magnetic Fields in Fusion Applications
P1-005	**Leyla Franz** Karlsruhe Institute of Technology	Corrosive Effect of Stagnant Molten Lead on EUROFER97 in WLCB Breeding Blanket
P1-006	**Bing Bai** China Institute of Atomic Energy	Analysis of Radiation Hardening effect of Ferritic/Martensitic Steel Based on Bayesian Optimization
P1-007	**Xiang (Frank) Chen** Oak Ridge National Laboratory	Effect of Irradiation Temperature on EUROFER97 Fracture Toughness after 20 dpa Neutron Irradiation

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P1-008	**James Gibson** United Kingdom Atomic Energy Authority	Irradiation Hardening in Advanced Reduced Activation Ferritic-Martensitic Steels for Future Fusion Applications
P1-009	**Igor Penalva** University of the Basque Country (UPV/EHU)	Direct experimental measurement of the solubility of hydrogen isotopes in the Eurofer-97 RAFM
P1-010	**Yutao Zhai** Institute of nuclear safety technology, Hefei Institute of Physical Sciences, CAS	Progress of Engineering Qualification on CLAM and Welding Wire for ITER-TBM in China
P1-011	**Weihua Wang** Anhui University	Corrosion behavior and mechanical properties of low activation ferritic martensitic steel and its welded joints in liquid lithium lead
P1-013	**Carsten Bonnekoh** Karlsruhe Institute of Technology, Institute for Applied Materials	Reduced activation 9Cr-ODS steels with EUROFER97-like Charpy properties and industrialization efforts
P1-014	**Benjamin R. S. Evans** UKAEA (United Kingdom Atomic Energy Authority)	High Throughput Analysis of Thermally Aged Reduced-Activation Ferritic-Martensitic Steels for Future Fusion Applications
P1-015	**Marta Serrano** CIEMAT	Experimental challenges for the determination of ductile fracture toughness of F/M steels
P1-016	**Diancheng Geng** IMR, Tohoku University	Corrosion resistance of High-Mn Oxide-Dispersion Strengthened (ODS) Alumina-Forming Austenitic (AFA) Steels in Liquid Metal Sn
P1-017	**Kazuma Kikkawa** Hokkaido university, Laboratory of Advanced Materials	Possibility of fabricating ODS high entropy alloys by using cold spray method

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P1-018	**Sanghoon Noh** Pukyong National University	Fabrication and Characterization of Oxide Dispersion Strengthened Alloys with directly introduced Y2Ti2O7 Pyrochlore Oxides for Fusion Applications
P1-019	**Kazuki Ogawa** Department of Quantum Science and Energy Engineering, Graduate School of Engineering, Tohoku Univ.	Impact of Mesoscale Heterogeneous Microstructure on Tensile Properties and Deformation Behavior of ODS Ferritic Alloys
P1-020	**Zhangjian Zhou** University of Science and Technology Beijing	Understanding of the high temperature creep resistance of ODS low activation F/M steels with carbon free design
P1-021	**Michael Thomas Duerrschnabel** Karlsruhe Institute of Technology	Nanoscale insights into 9Cr-ODS steels
P1-022	**Teng Li** Graduate School of Engineering Science, Yokohama National University	Retardation of Phase Transformation by Oxide Particles in Oxide Dispersion Strengthened (ODS) Steels
P1-023	**Luke Howard** University of Birmingham	Design of novel low activation intermetallic reinforced nano-structured steels for fusion
P1-024	**Alice Perrin** Oak Ridge National Laboratory	Effect of Secondary Phases and Solute Additions on Bonding and Thermal Transport at W/Cu Interfaces
P1-025	**Shinsuke Okubo** Shimane University	Effect of helium bubbles on hydrogen retention in tungsten studied by STEM-EELS

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P1-026	**Jannik Tweer** Forschungszentrum Juelich GmbH, Institute of Fusion Energy and Nuclear Waste Management (IFN-1)	High heat flux testing of wire-based laser metal deposition coated plasma-facing components
P1-027	**Shingo Okumura** Shizuoka University	Trapping and Diffusion Profile of Hydrogen Isotopes for Damaged Tungsten-Tantalum alloy
P1-028	**Naoya Matsuta** A.L.M.T. Corp., Japan	Development of potassium doped tungsten thick plates with optimized mechanical properties by microstructure control.
P1-029	**Alexander Lau** Institute of Fusion Energy & Nuclear Waste Management (IFN-1), Forschungszentrum Julich GmbH	From Lab-Scale to Industry: Advanced CVI Strategies for Wf/W Composite Fabrication
P1-030	**Zimo Gao** Tohoku University	Development of Novel WO3-doped ODS-Cu Alloys Towards Industrial-Scale Production
P1-031	**Koray Iroc** SCKCEN (Belgium Nuclear Research Center)	Comparative Study on Microstructural Response of ITER-Grade Tungsten Under Fusion-Relevant Irradiation Conditions
P1-032	**Harsh P Dudhatra** University of Miami	Mesoscale Modeling of First Wall and Blanket Materials Under Fusion Reactor Conditions
P1-033	**Xiaoqiang Zhang** School of Materials Science and Engineering, Hefei University of Technology	Wet chemical surface pre-activation of tungsten powders for additive manufacturing
P1-034	**Sungmin Lee** Seoul National University	Enhancing Fracture Toughness of Tungsten via W-Cr Composite with Tungsten Heavy Alloy Structure for Plasma-Facing Components

Tentative

P1-035	**Menghan Ma** Tsinghua University	A new crack self-healing mechanism in additive manufactured tungsten alloy for nuclear fusion
P1-036	**Stanley Chang** Massachusetts Institute of Technology	Designing irradiation-tolerant TaTiVW refractory high-entropy alloys with defects simulation and heavy ion irradiation
P1-037	**Kentaro Masuta** Interdisciplinary Graduate School of Engineering Sciences, Kyushu University	Hydrogen Permeation Behavior during the Formation of Tungsten Deposition Layers under Simultaneous Irradiation with Hydrogen and Helium
P1-038	**Yiming Niu** School of Materials Science and Engineering, Tsinghua University	Crack Suppression and Microstructural Evolution in Binary Tungsten Alloy fabricated by Laser Powder Bed Fusion
P1-039	**Toni Dunatov** Rudjer Boskovic Institute	Examination of a WTaCrV high entropy alloy under single and dual-beam ion irradiation
P1-040	**Ayumu Hayakawa** Shizuoka University	Effect of irradiation fluence on deuterium retention for damaged W-Mo alloy
P1-041	**Toshiki Saito** Tohoku University	Neutron irradiation effect on the material properties and the evaluation of bonding strength with W of Cu alloys
P1-042	**Kenshiro Miura** Graduate School of Integrated Science & Technology, Shizuoka University	Evaluation of deuterium retention behavior in potassium-doped tungsten with irradiation defects under high-temperature conditions
P1-044	**DEBAMOY PEGU** Institute of Advanced Study in Science and Technology	Unveiling the Impact of Dopants on the Metallic Cu-W Interface for Fusion Reactor Applications

Tentative

P1-045	**Lukas Klos** Karlsruher Institut fur Technologie, Karlsruhe	Overaging of CuCrZr: Degradation and Limits to the Restoration of the Initial Properties
P1-046	**Tianren Yang** Beihang University	Exploring the inhibitory effect of WTaVCr high-entropy alloys on hydrogen retention: From dissolution, diffusion to desorption
P1-047	**Ryan David Kerr** United Kingdom Atomic Energy Authority	Development of an interatomic potential for the study of tungsten-based SMART alloys under the atomic cluster expansion formalism
P1-048	**Minha Park** Institute for Materials Research, Tohoku university	Effects of neutron irradiation on hardness and electrical resistivity of tungsten and tungsten alloys after neutron irradiation in BR2
P1-049	**Miyuki Yajima** National Institute for Fusion Science	Vacancy Cluster Formation and Hydrogen Isotope Exposure Effects on Neutron Irradiated Tungsten
P1-050	**Elena Tejado** Universidad Politecnica de Madrid	Effects of ageing on the service thermal and mechanical properties of additive manufactured CuCrZr alloys
P1-051	**Juan Du** Southwestern Institute of Physics	Performance study of tungsten fiber-reinforced tungsten composites for nuclear fusion reactor
P1-052	**Kouichi Tougou** University of Fukui	Interaction between the dislocations and cavities in tungsten with in-situ TEM observation during tensile test at 973 K
P1-054	**Daniel Dorow-Gerspach** Forschungszentrum Julich GmbH	High heat flux test components for repaired plasma-facing materials
P1-055	**Marius Wirtz** Forschungszentrum Juelich	High pulse number tests of pre-damaged plasma facing surfaces

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P1-056	**Edgar Leon-Gutierrez** National Fusion Laboratory - CIEMAT	Optimization of additive manufacturing process parameters in ITER-grade CuCrZr using rapid screening methods
P1-057	**Arkadi Kreter** Forschungszentrum Juelich	Overview of deuterium retention measurements in tungsten with impurity seeding in the linear plasma device PSI-2
P1-058	**Armin Manhard** Max Planck Institute for Plasma Physics	Direct comparison of gas- and ion-driven deuterium permeation through fusion-relevant materials
P1-059	**Begona Savoini** Universidad Carlos III de Madrid	Evaluation of residual thermal stress and temperature distributions in W-HEA-CuCrZr multilayer systems for fusion nuclear divertors.
P1-061	**Kuanysh Samarkhanov** Institute of Atomic Energy Branch of the National Nuclear Center of the Republic of Kazakhstan	Non-isothermal high-temperature corrosion behavior of CrBe12 in vapor-gas mediums with different isotopic compositions
P1-062	**Yutaka Sugimoto** QST	The evaluation of deuterium retention properties of Be12Ti after irradiation with D ions at high temperature
P1-063	**Yiming Pu** School of Materials Science and Engineering, Wuhan University of Technology	Analysis of hydrogen isotope release from lithium ceramics with different pore sizes and distributions
P1-064	**Charles Victor Thorogood** University of Sheffield	The impact of lithium deficiency and presence of helium on the performance of lithium metatitanate (Li2TiO3).
P1-065	**Harsh Patel** Institute for Plasma Research	CTE Measurement of Pebble Beds with Radial Expansion Correction of the Sample Holder

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P1-066	**Julia Leys** Karlsruhe Institute if Technology (KIT)	Thermal Cycling Test of EU Reference Tritium Breeder Pebbles
P1-067	**Wei Lu** Anhui University	Study on the kilogram-scale preparation lithium lead titanate ceramic pebbles as a new neutron-tritium complex breeder for fusion reactor blanket
P1-068	**Saulet Askerbekov** Institute of Nuclear Physics, Almaty	TGA study of high-temperature corrosion behavior of neutron-irradiated Be12Ti in the presence of water vapor
P1-069	**Fernando Mota** Laboratorio Nacional de Fusion por Confinamiento Magnetico, CIEMAT	Liquid Breeder Materials qualification in IFMIF-DONES Facility
P1-070	**Etsuo Ishitsuka** Ibaraki University, RECAS	What is the Optimal Size of Beryllium Pebbles as a Neutron Multiplier?
P1-071	**Young ah Park** Korea Institute of Fusion Energy	Study on Powder Synthesis with Core-shell Structure for Tritium Breeder
P1-072	**German D Samolyuk** Oak Ridge National Laboratory	Novel hybrid Li-Be ceramics, combining both functions of tritium (T) breeder (Li) and neutron multiplier (Be)
P1-073	**Inesh Kenzhina** Satbayev University	Investigation of the possibility of obtaining beryllides of rare-earth metals
P1-074	**Kosuke Okubo** Shizuoka University	Li-Pb corrosion effects on deuterium permeation in functional coatings using yttrium oxide and zirconium oxide
P1-075	**Hnin Lai Lai Wai** Shizuoka University	Corrosion mechanisms of ceramic-iron joint coatings in lithium-lead under thermal cycling and gamma-ray irradiation

Tentative

P1-076	**Ren Tsukamoto** Shizuoka University	Ion irradiation and implantation effects on electrochemical properties of functional ceramic coatings
P1-077	**Kieran Paul Ralston** University of Sheffield	Development of Tritium Permeation Barriers for Non-plasma-facing Applications for Nuclear Fusion
P1-078	**Koichiro Doi** Kyoto University	Measurement of nitrogen diffusivity in Fe?Ti alloy films by glow discharge optical emission spectroscopy
P1-079	**Yuiki Ogura** Nagoya University	Study on irradiation damage and temperature monitoring applying photoluminescence properties of Er2O3 under Gamma-ray environment
P1-080	**Guillermo de la Cuerda Velazquez** Universidad Politecnica de Madrid	Permeation and corrosion barrier capabilities of amorphous SiC coatings under PbLi exposure
P1-081	**Shane Miller Wylie** Shizuoka University	Deuterium permeation and Li-Pb corrosion behaviors of single- and multi-layer coatings using Y-Al-O system
P1-082	**Heping Li** Huazhong University of Science and Technology	One-step fabrication of AlPO4 nanosheet reinforced α-Al2O3 coatings with improved fracture toughness and hydrogen resistance
P1-083	**Pei-song Du** The Institute of Plasma Physics, Chinese Academy of Sciences	HIP diffusion bonding of FeCrAl and RAFM steel for preparation of tritium permeation barrier
P1-084	**Tommaso Sgrilli** Gencoa Ltd	Development of ceramic coatings as protective layers for structural vanadium-based alloys
P1-085	**Alexandre Thomas Bernard Lhoest** Imperial College London	A mesoscale phase-field model of intergranular liquid lithium corrosion of ferritic/martensitic steels

Tentative

P1-086	**Azusa Matsumoto** Tohoku University	Oxidation behavior of neutron-irradiated tungsten and its dependence on irradiation temperature
P1-087	**Yuki Setoguchi** Kyushu University	Reaction products at the interface between supercritical carbon dioxide and stainless steel
P1-088	**Takahiro Aoki** Kindai University	Mobilization behavior of Re oxide sublimated from W-Re alloys during high-temperature oxidation
P1-089	**Kota Okada** Institute for Materials Research, Graduate School of Engineering, Tohoku University	Evaluation of the integrity of CVD-SiC coatings after exposure to high temperature and high pressure water.
P1-090	**Claude MY De LamaterBrotherton** ORNL UTK	Compatibility of composite insulators with high temperature liquid Lithium
P1-091	**Marie Romedenne** Oak Ridge National Laboratory	Impact of high temperature liquid metal exposure on design of structural first wall and blanket materials
P1-092	**Xingwen Feng** Institute of Materials, China Academy of Engineering Physics	Ultra-efficient hydrogen isotope oxidation by Pt nanoclusters confined in Silicalite-1 zeolite
P1-093	**Joven Lim** UK Atomic Energy Authority	Degradation behaviour of Vanadium-Lithium Tritium Breeder Blanket System under Fusion-relevant Neutrons
P1-094	**Takaaki Koyanagi** Oak Ridge National Laboratory, Materials Science and Technology Division	Corrosion of monolithic and composite SiC by molten Be-bearing salt

Tentative

P1-096	**Xiaolong Fu** China Academy of Engineering Physics, Institute of Nuclear Physics and Chemistry	Tritiated water treatment of fusion reactor enhanced by metal-organic frameworks
P1-097	**Suneui Lee** Kyoto Fusioneering Ltd.	Hydrogen Capacity Evaluation of the ZrCo Tritium Storage Bed in Fusion Power Plant and Verification of Its Repeated Recovery Effect
P1-100	**Luke Christopher Olson** Savannah River National Laboratory	Development and De-risking of Tritium Extraction Processes by Flow-Loop Integration
P1-101	**Motoki Nakajima** QST	Effect of hydrogen peroxide on corrosion properties of fusion reactor materials
P1-102	**Liberato Volpe** UKAEA	Microstructural analysis of oxidised Al-base coated EUROFER97 for corrosion protection
P1-103	**Julia Navas Gomez** CIEMAT (Centro de Investigaciones Energeticas Medioambientales y Tecnologicas)	Experimental Design for Assessing the Performance of Structural Materials and Components in Liquid Lithium Environments
P1-105	**Julian Patino** CIEMAT (Centro de Investigaciones Energeticas, Medioambientales y Tecnologicas)	The use of Yttria-stabilized zirconia for the manufacturing of an electrochemical hydrogen sensor for liquid metals
P1-106	**Makoto Kobayashi** National Institute for Fusion Science	Thermogravimetry analysis for the reaction of tungsten and tungsten-rhenium alloy under oxygen atmosphere in 873-1173 K

Tentative

P1-107	**Hongfei Liu** Institute of Materials Research and Engineering, Agency for Science, Technology and Research	Corrosion Behavior and Characterization of Ni- and Ti-Based Alloys in Molten Salt and Liquid Metal Environments
P1-108	**Maria Sanchez Arenillas** CIEMAT (Centro de Investigaciones Energeticas, Medioambientales y Tecnologicas)	Reuse of yttrium as a getter material to retain hydrogen isotopes in an impurity trap present in the ICS of IFMIF-DONES
P1-109	**Anastacia Wright** University of Oxford	Microstructural analysis of oxidised Al-base coated EUROFER97 for corrosion protection
P1-110	**Brandon Jay Ralphe Stratton** Bangor University	Investigating the decomposition of lithium hydride isotopologues for fusion fuel breeder blanket development
P1-111	**Bradley Young** Oxford Sigma	Building a high temperature liquid lithium flow loop: Lessons for liquid metal engineering
P1-112	**Jingchuan Wang** Institute of Materials, China Academy of Engineering Physics	Investigation of Hydrogen Isotope Effect at Different Atomic Sites in Nano-Crystalline ZrCo
P1-113	**Yuriy N. Gordienko** Institute of Atomic Energy Branch of the National Nuclear Center of the Republic of Kazakhstan	Studies of high-temperature interaction of Pb16Li eutectic with chemically active gases and water vapor
P1-116	**Nikolai Zimber** Karlsruhe Institute of Technology	Characterizing the mechanical behaviour of different heat treated EUROFER97 materials in non-irradiated and irradiated state

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P1-117	**Gaurav Verma** Karlsruhe Institute of Technology	Advancements in WEBAM for Fabrication of HCPB Breeding Blanket Components
P1-118	**Charlotte Brown** The University of Manchester	Impact of the use of a dual laser system on the microstructure and mechanical properties of PBF-LB/M tungsten
P1-119	**JianKe Tian** Advanced Energy Science and Technology Guangdong Laboratory	Phase Evolution and Electrochemical Performance of CoCrFeNiM (M = Ta, Nb, V) High-Entropy Alloys Synthesized by Powder Metallurgy
P1-120	**Gunnar Schmidtmann** Forschungszentrum Julich GmbH	Performance of Plasma-Sprayed Tungsten Coatings Under Plasma Exposure
P1-121	**Samuel F. Yeates** The University of Sheffield	Multi-Length Scale Characterisation of the role of Ti(CON) precipitates on radiation damage in V-based, reduced activation, metallic systems
P1-122	**Wen Yin** Institute of High Energy Physics, CAS	The Irradiation Effect Research of the Spallation Target
P1-123	**Agostina Orefice** ENEA	New concrete mix design for Nuclear Application
P1-124	**Fei Wang** School of Aerospace Engineering, Xi'an Jiaotong University	Enhancing Hydrogen Isotope Permeation Resistance: The Significance of Plasma - Induced - Oxide - Interlayer
P1-125	**Hyoseong Gwon** Korea Institute of Fusion Energy	Development of Fabrication Technologies for the Breeder Unit Cooling Plate in the HCCP Breeding Blanket

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P1-127	**Zhanar Bugybay** Institute of Nuclear Physics, Almaty	Development of irradiation capsule design and plan for neutron irradiation of chromium beryllide samples in the WWR-K reactor
P1-128	**Joven Lim** UK Atomic Energy Authority	Scale-able and Tune-able Compact Accelerator Neutron Source for Materials Testing under Fusion Environments
P1-129	**Lauryn Kortman** University of Michigan, Ann Arbor	Advancements in Multi-Ion Beam Simulations for Fusion Material Testing at Elevated Temperatures
P1-130	**Zachary Seth Hartwig** MIT Plasma Science and Fusion Center	Introducing the MIT PSFC LMNT Facility: Implementing high throughput, near prototypic irradiation of bulk fusion materials with 10-30 MeV protons
P1-131	**Kihyun Lee** Korea Atomic Energy Research Institute	ECR Ion Source Upgrade for Fe Ion Beam Supply at KAHIF for Nuclear Materials Irradiation Tests
P1-132	**Masaya YAMAMOTO** Japan Atomic Energy Agency	The experimental sodium cooled fast reactor Joyo as powerful neutron irradiation facility for developing fusion reactor materials
P1-133	**Rahul Rayaprolu** Forschungszentrum Julich	A unified approach to active sample handling and plasma exposure at the forthcoming nuclear linear plasma device JULE-PSI
P1-134	**Sangbeen Lee** Korea Atomic Energy Research Institute	Operation and upgrade progress of KAHIF for fusion material irradiation testing
P1-135	**Sara Elizabeth Ferry** MIT Plasma Science and Fusion Center	The MIT PSFC LMNT facility: near-term research plans to rapidly advance fusion materials science and support fusion commercialization

Tentative

P1-136	**Lou P. Wainwright** Massachusetts Institute of Technology	The MIT PSFC LMNT facility: Target design, testing methods, and experimental techniques for 10-30 MeV proton irradiation
P1-137	**Angus P C Wylie** Department of Nuclear Science and Engineering, Massachusetts Institute of Technology	Facility for the Evaluation of Thermal and Elastic Material Properties During Plasma or Ion Irradiation
P1-138	**Seunghyun Lee** Korea Atomic Energy Research Institute	Overview of a KAERI Heavy-ion Irradiation Facility (KAHIF) for nuclear fusion/fission material research
P1-139	**DaeSik Chang** Korea Atomic Energy Research Institute	Application of EPICS in KAHIF and KoHLT-EB System for Fusion Material Irradiation and Plasma Facing Component Testing
P1-140	**Shuka Tsuda** The Graduate University for Advanced Studies (SOKENDAI)	Evaluation of Deuterium Retention Properties on Pure Tungsten with Various Cold-Working Ratio
P1-141	**Anthony Francis Tom** University of Tennessee Nuclear Engineering	Modeling the Effect of He/dpa Ratio on Prismatic Loop Evolution in Irradiated Ferritic-Martensitic Alloys
P1-142	**Yuki Noguchi** Kyushu University	Machine-learning force fields for hydrogen and vacancy complex in tungsten
P1-143	**Mario Matic** Ruder Boskovic Institute	SIMS and TEM characterization of Eurofer97 irradiated with helium ions using energy degrader and dose gradient

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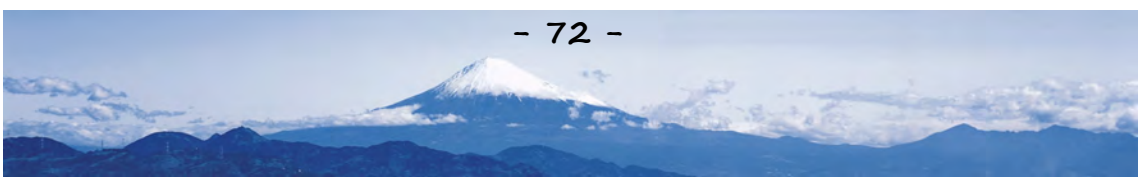
P1-144	**Zhenyu Jiangzhenyu123 Jiang** School of Physics, Beihang University	The effect of microstructure and defect concentration of tungsten on surface blistering and deuterium retention
P1-145	**Guanying Wei** University of Helsinki	Diffusion of helium in tungsten-based refractory alloys by molecular dynamics
P1-146	**Yiheng Chen** Institute of Heavy Ion Physics, Peking University	Investigation hydrogen-helium synergistic damage mechanisms using multibeam ion irradiation
P1-147	**Yuxin Liu** Peking University	The role of H and He in synergistic effects on cavity evolution in pure Ni at multiple temperatures
P1-148	**Zhanna ZAURBEKOVA** Institute of Nuclear Physics	Modeling experimental dependencies of deuterium penetration through SS316IG stainless steel under reactor irradiation conditions
P1-149	**Mary Alice Cusentino** Sandia National Laboratories	Atomistic Modeling of Hydrogen Permeation in Vanadium Structural Alloys for FLiBe Blankets
P1-150	**Prashanth Srinivasan** United Kingdom Atomic Energy Authority (UKAEA)	Atomistic simulations of tritium across a vanadium-lithium interface using machine-learning interatomic potentials
P1-151	**Koichi Sato** Kagoshima University	Hardening of ion-irradiated and electron-irradiated single-crystal tungsten induced by hydrogen charging
P1-152	**Peter Hosemann** UC Berkeley	Helium induced swelling and blister formation in fusion candidate materials
P1-153	**Sanjeet Kaur** United Kingdom Atomic Energy Authority	Multi-scale modelling of hydrogen isotopes retention and diffusion in highly irradiated materials

Tentative

P1-155	**Shijun Zhao** City University of Hong Kong	Atomistic Response of Tungsten-Containing Refractory High-Entropy Alloys to Helium Ion Irradiation
P1-156	**Jianrong Sun** Institute of Modern Physics, Chinese Academy of Sciences	Helium behavior of FeCrVTix medium-entropy alloys for advanced nuclear applications
P1-157	**Yu-Hao Li** Beihang University	Suppressing the nucleation and growth of helium bubbles in WTaVCr high-entropy alloys
P1-158	**Flyura Djurabekova** University of Helsinki	Effect of chemical complexity and presence of vanadium on He dynamics in concentrated refractory alloys
P1-159	**Daniel Robert Mason** UKAEA	Multiscale simulation of hydrogen isotope diffusion in irradiated tungsten
P1-160	**Chuanxin Liu** Northwestern Polytechnical University	Helium behaviors on the microstructure and mechanical properties of the Cansas 3303 silicon carbide composites
P1-161	**Youping Lu** Southwestern Institute of Physics	The Comparison of Hardness between MAX Phase Materials Irradiated with Protons and Neutrons
P1-162	**Katie Karl** University of Tennessee	The Effect of Alloy Composition on Mechanisms of Helium Diffusion in MoNbTaTi High Entropy Alloys
P1-163	**Regis Bisson** Aix-Marseille University	Simulating plasma-surface interactions with deuterium beams experiments: effect of oxygen and carbon impurities on fuel retention and recycling
P1-164	**Xiaoou Yi** University of Science and Technology Beijing, China	Damage microstructure, hardening and deuterium retention in neutron irradiated tungsten and tungsten-rhenium alloy

Tentative



P1-165	**Kaigui Zhu** School of Physics, Beihang University	The effect of annealing temperatures on plasma induced surface blistering and deuterium retention in tungsten
P1-166	**Yan Ge Zhang** Hefei Institutes of Physical Science, Chinese Academy of Sciences	Hydrogen trapping, desorption and clustering in heterophase interfaces of W-ZrC alloy
P1-167	**Cuncai FAN** Zhejiang University	Using Helium Ion Microscopy to understand the mechanisms of helium irradiation blistering in tungsten
P1-168	**Amon Ikubo** Graduate school of science and engineering, Kagoshima University	Effect of multiple trapping of hydrogen atoms on formation energy of vacancy-single helium complexes in α-iron
P1-169	**Martin Balden** Max-Planck Institut for Plasma Physics	Disentangling Hydrogen- and Defect-driven Segregation in the Iron-Tungsten-System
P1-170	**Hong-Bo Zhou** Beihang University	Influence of hydrogen on the displacement defect evolution and mechanical properties of tungsten
P1-171	**Byeongchan Lee** Kyung Hee University	Changes in the neutron-irradiation damages to W due to the presence of H
P1-172	**Xiaochun Li** Institute of Plasma Physics, Chinese Academy of Sciences	Helium Segregation and Diffusion Behavior at Grain Boundaries in Tungsten: A Molecular Dynamics Study
P1-173	**Pavel V. Vladimirov** Karlsruhe Institute of Technology	A DFT- and Machine Learning-based Study of Tritium Retention in Titanium Beryllide

Tentative

P1-174	**Dale Hitchcock** Savannah River National Laboratory	Tritium Material Interactions Research at the Savannah River National Laboratory
P1-175	**Liberato Volpe** UKAEA	Oxidation and environmentally-assisted cracking behaviour of Eurofer-97 under simulated water-cooled breeder blanket conditions

Tentative

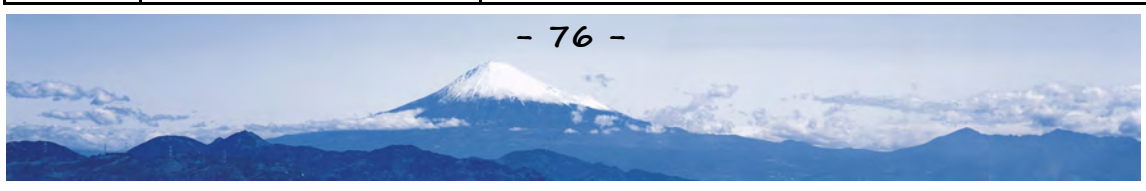
30th September 2025 (Tuesday)

Time 17:00-19:00 Gallery Room Chairs: J. Aktaa (KIT), F. Sun (HFUT), X. Chen (ORNL), F. Luo (PKU), Y. Watanabe (QST)		
P2-001	**Marcelo Roldan Blanco** CIEMAT	Stability of Lab-Scale 14%Cr Oxide dispersion-strengthened steel (ODS) produced by STARSc route
P2-002	**Viacheslav Kuksenko** United Kingdom Atomic Energy Authority	Performance of high-Cr oxide dispersed strengthened steels under high-dose neutron irradiation
P2-003	**Tingjun Huang** Institute of Nuclear Energy Safety Technology	Preparation and properties of high-density oxide strengthened ODS steel by liquid metal method
P2-004	**Xuan MENG** Lanzhou University	The S/TEM Characterization of MA956 ODS Steel after SHI Irradiation
P2-005	**Cheng Cheh Dennis Tan** Institute of Materials Research and Engineering	Effect of milling atmosphere on microstructure and mechanical properties of ODS-Steel fabricated by spark plasma sintering
P2-006	**Tatsuya Kawakami** Hokkaido University, Laboratory of Advanced Materials	Evaluation of oxide dispersion strengthened high entropy alloys fabricated by laser additive manufacturing
P2-007	**Tiago Lopes** University of Helsinki	Development of Machine Learning Interatomic Potentials for HfNbTaTiZr Alloys
P2-008	**Lakshay Chauhan** Karlsruhe Institute of technology	Disorder-order phase transformation in ion-irradiated Co-free high entropy alloy

Tentative

P2-009	**Deasik Kim** Pukyong National University	Effect of Re, Ta Additions on the Mechanical Properties and Microstructure of Spark Plasma Sintered ODS Tungsten Alloys
P2-010	**Roman Afanasenko** Karlsruhe Institute of Technology	Influence of shielding materials on SDDR of the DEMO tokamak for ex-vessel areas at initial operation
P2-011	**Shunji Koide** Graduate school of Engineering	Microstructural change in oxide dispersion strengthened medium entropy alloys under electron irradiation
P2-013	**Mahmud Hasan Ovi** University of Illinois Urbana Champaign	Laser Powder Bed Fusion Printing of Tungsten on G91 Steel for Fusion Reactor Applications
P2-014	**Sean Mills** University of California Berkeley	SINGLE ION REIRRADIATION OF DUAL ION IRRADIATED TEMPERED MARTENSITIC ALLOYS TO MEASURE POST INCUBATION VOID SWELLING RATES
P2-015	**Jie Chen** Forschungszentrum Julich GmbH, Institute of Fusion Energy and Nuclear Waste Management (IFN-1)	Effects of different microstructures of WCrY SMART material on the oxidation resistance and mechanical properties
P2-016	**Alexander Knowles** University of Birmingham	Design of novel nanostructured tungsten and ferritic superalloys with high sink strength for fusion
P2-017	**Jesper Byggmastar** University of Helsinki	Understanding refractory metals and alloys with machine-learned interatomic potentials
P2-018	**Chenyang Lu** Xi'an Jiaotong University	Study on Microstructures and Helium Ion Irradiation Effects in WTaV Multi-Principal Element Alloys
P2-019	**Thomas Davis** Oxford Sigma	Advancements in pressure codes & standards for fusion power plants

Tentative



P2-020	**Jingjie Shen** National Institute for Fusion Science	Microstructure changing and irradiation hardening behavior of high-purity vanadium alloys
P2-021	**Bo-Shiuan Li** National Sun Yat-sen University	Optimizing Laves Phase in Refractory HEAs for Fusion Applications
P2-022	**Chun-Liang Chen** Department of Materials Science & Engineering, National Dong Hwa University	Effect of V/Ti Ratio, Reinforcement Phases/Oxide Particles on the Microstructure and Mechanical Properties of WMoVTi Alloys Synthesized by Mechanical Alloying
P2-023	**Wei-Bing Liao** Shenzhen University	A preliminary study on low-activation refractory high-entropy alloys with improved mechanical properties
P2-024	**Zhe Sun** Institute of plasma physics, Chinese Academy of Sciences	Enhancing Passive Transport Performance in Microchannel-based Capillary Porous Systems for Liquid Metal Divertors: A Comprehensive Analysis
P2-025	**Pablo Yelo** CIEMAT	Parametrical study of the thermal response of liquid metal-W capillary porous systems exposed to high thermal loads
P2-026	**Yuya Ando** Kindai University	Deuterium permeation behavior through liquid LiPb supported by V-based alloy for hydrogen isotope extraction
P2-027	**Yuto Murata** Kindai University	Characterization of alloy layers formed by reaction between a liquid LiPb and supporting metal substrates
P2-028	**Zongbiao Ye** Sichuan University	The design of 3D-printing solid tungsten-liquid lithium combined divertor target plate and its performance investigation

Tentative

P2-030	**Juergen Rapp** Oak Ridge National Laboratory	Plasma-Material Interaction Science enabled by the new capabilities of MPEX
P2-031	**Jiaqin LIU** Beijing University of Chemical Technology	The effect of W precursors on in-situ induced Y2O3/W dispersive particles synergistic strengthened copper alloys
P2-032	**Chad Parish** Oak Ridge National Laboratory	Plasma-Focused Ion Beam Analysis of WEST Monoblocks
P2-033	**Hyoung Chan KIM** Korea Institute of Fusion Energy	Investigation of the Thermal Stability of Potassium-doped Tungsten fabricated by the Spark Plasma Sintering method
P2-034	**Long Li** University of Science and Technology of China	Deuterium permeation and retention behaviors in dispersion-strengthened W
P2-035	**Eduardo Alves** Instituto de Plasmas e Fusao Nuclear (IPFN), Instituto Superior Tecnico, Universidade de Lisboa	$(CrVW)_{1-x}$ $(TiTa)_x$ as thermal barrier for nuclear fusion
P2-036	**Mathias Jetter** Karlsruhe Institute of Technology	Brittle Fracture Assessment for Tungsten
P2-037	**Rui Liu** Institute of Solid State Physics, HFIPS, Chinese Academy of Sciences	High performance W-ZrC alloy and nanostructured Cu-W for plasma-facing components

Tentative

P2-038	**Kunru Yang** Dalian University of Technology	Simulation of net erosion effects on tungsten nano-tendril bundles growth under He/Ne plasma irradiation
P2-039	**Tshepo Mahafa** Tritium Fuel Cycle, UK Atomic Energy Authority	The retention and release of hydrogen isotopes from boron coated tungsten
P2-040	**Tim Graening** Oak Ridge National Laboratory	Geometric properties relationships for electron beam melted tungsten alloys
P2-041	**Yue Xu** Hefei University of Technology	Effects of helium irradiation damage on deuterium plasma-driven permeation through tungsten coated RAFM steel
P2-042	**Wei Jin** Science and Technology on Surface Physics and Chemistry Laboratory	Surface damage and deuterium retention in tungsten under high-flux detached recombining linear plasmas
P2-043	**Gayatri Dhamale** Oak Ridge National Laboratory	Low-ion energy sputtering yields of high-Z refractory coatings for helicon plasma source window material
P2-044	**Jan Coenen** Forschungszentrum Julich GmbH, Institute of Fusion Energy and Nuclear Waste Management, IFN-1	Industrial Scale Up for Tungsten Fiber-Reinforced Tungsten Composites for Plasma-Facing Applications
P2-045	**Binyou Yan** Xiamen Tungsten Co., Ltd.	Fabrication and Characterization of Potassium-Doped Bulk Tungsten with Superior High-Temperature Microstructural Stability and Low-Temperature Toughness for Future Fusion Devices

Tentative

P2-047	**Marcin Rasinski** Forschungszentrum Julich GmbH	The influence of Ne admixture on erosion and fuzz formation of W exposed to D+He+Ne plasma in a PSI-2 linear plasma device
P2-048	**Heung Nam Han** Seoul National University	A Study on Ductile-to-Brittle Transition Temperature of Pure and Potassium-Doped Tungsten
P2-049	**Marco Minissale** PIIM laboratory, CNRS	Microscopic and macroscopic investigation of He bubbles formation on tungsten via in situ spectroscopic ellipsometry
P2-050	**Hiroyuki Noto** National Institute for Fusion Science	A Study of Ultra-strong Tungsten-based Multiphase High Entropy Alloys
P2-051	**Camilo Jaramillo-Correa** Postdoctoral Research Associate	Deuterium retention and thermal stability on boron surfaces
P2-052	**Lauren Nuckols** Oak Ridge National Lab	Deuterium Erosion and Retention Behavior of Ultra-high Temperature Ceramic TiB_2
P2-054	**Svitlana Rudchenko** Technical University of Denmark	Thermal stability of tungsten fiber-reinforced tungsten composites fabricated by powder metallurgy
P2-055	**Kota Muroi** Tokyo Metropolitan University	Development of high-entropy-type REBCO superconductor with multiple sites substitution (Y, Gd, Dy, Yb)$_{0.25}$Ba$_{2-x}$Sr$_x$Cu$_3$O$_7$-δ
P2-056	**Hiroto Yamada** Yokohama National University	Evaluation of radiation resistance by in-situ observation of electron irradiation of REBa$_2$Cu$_3$O$_7$-δ with multiple element substitution
P2-057	**Rui Gao** Xi'an Jiaotong University	The Influence of Particle Irradiation on the Electrical Properties of REBCO High-temperature Superconducting Strips for Compact Fusion Reactors

Tentative

P2-058	**Yang Wang** Hokkaido University LOAM Laboratory	Fabrication of high-purity Zr_3AlC_2 MAX phase by spark plasma
P2-059	**Takumi Sato** Institute of Laser Engineering , Osaka University	Evaluation of Helium Irradiation Damage to Dielectric Mirrors in Laser Fusion Reactors
P2-060	**Victoria Haak** Max-Planck-Institut fur Plasmaphysik	Effects of neutron irradiation on LaB6-emitters used in neutral gas pressure gauges
P2-061	**Augusto Pereira** CIEMAT	Generative AI-expert system for the characterization of optical materials in nuclear fusion engineering
P2-062	**Rafael VILA** Lab. Nacional de Fusion - CIEMAT	Neutron Irradiation of DEMO Functional Materials: Bridging Current Research and Future Requirements for Diagnostic, Control and Heating Systems
P2-063	**Martin Stiehler** Cranfield University	Developing Metallic Glasses for Nuclear Fusion applications
P2-064	**Andrey Ushakov** TNO	In-vessel radiofrequency power transmission circuits development for first mirror plasma cleaning in 55.C2.ETS and 55.E6.VSRS ITER optical diagnostic systems
P2-065	**Kaiji Yamamura** Tohoku University	Optimization of Sintering Conditions for Tungsten Boride Cermets for Application as Radiation Shielding Materials
P2-066	**Mashu Harada** Imperial College London	Volumetric Swelling in Helium Ion-Irradiated WB2-Based Neutron Shielding
P2-067	**Christopher Moore** Tokamak Energy Ltd,	Establishing and testing of alternative low activation tungsten carbide shielding materials for fusion energy systems

Tentative

P2-068	**Guangnan Luo** Institute of plasma physics, Chinese Academy of Sciences	Preparation of tungsten-based shielding material with large size
P2-069	**Seiji Nakabayashi** A.L.M.T. Corp.	A new concept of radiation shielding material - W-B-C system for improved mechanical properties.
P2-070	**Sandeep Irukuvarghula** Tokamak Energy Ltd,	Advanced microstructural characterisation of tungsten borides as high-performance shields for fusion
P2-071	**Yuanyuan Li** Nuclear Power Institute of China	Preparation and Properties of Tungsten Monoboride Shielding Material
P2-072	**Max Rigby-Bell** UK Atomic Energy Authority	Ion Irradiations of tungsten carbide and borides for fusion neutron shielding
P2-073	**Max Emmanuel** UK Atomic Energy Authority	Study of WC based cemented carbides as shielding candidates for compact spherical tokomaks
P2-074	**Daichi Miura** Department of Quantum Science and Energy Engineering, Graduate School of Engineering, Tohoku Univ.	Development of additive manufacturing method for high-purity SiC and its material properties
P2-075	**Xuxiao Han** Institute of Modern Physics, Chinese Academy of Sciences	Effect of Zr/Hf addition on ODS steel
P2-076	**Robert Kolasinski** Sandia National Laboratories	Mechanisms of hydrogen and oxygen adsorption on vanadium super-permeable membrane surfaces
P2-077	**Kensuke Hofuku** Kureha Corporation Ltd.	New manufacturing strategy of SiC fiber required for fusion reactor application

Tentative

P2-078	**Shijie Wang** Institute of Materials Research and Engineering, Agency for Science, Technology and Research	High entropy oxides with tunable thermal-mechanical properties for fusion applications
P2-079	**Keshav Vasudeva** Massachusetts Institute of Technology, Department of Nuclear Science and Engineering	Evaluation of SiC and SiC-based ceramics for use in advanced vacuum-vessel designs
P2-080	**YUKA MORI** Kyushu University	Measurement of low concentration deuterium in lithium using gas chromatography method with water dissolution - isotope exchange reaction
P2-081	**Weicheng Zhong** Oak Ridge National Laboratory	Mechanistic Evaluation of Deuterium Trapping in Strengthening Microstructure Features of Ferritic-Martensitic Steels
P2-082	**Ao Zhang** Kyushu University	Purge gas effect on tritium release behavior from neutron-irradiated FLiNaBe mixed with Ti powder
P2-083	**Kota Saito** Plasma Research Center, University of Tsukuba	Comparison of deuterium desorption from tungsten fuzz between multiple plasma devices
P2-084	**Maria Sakamoto** Plasma Research Center, University of Tsukuba	Deuterium retention properties of He-W co-deposited layers formed at different sample temperature and flux ratio of W to He
P2-086	**Hiroki Isogawa** Kyushu University	Tritium Production and Confinement in High-Temperature Gas-Cooled Reactors for Fusion Fuel Supply
P2-087	**Ibuki Sato** Hokkaido University	Deuterium enrichment by AEM water electrolysis with varying water utilization

Tentative

P2-088	**Yui Ishimori** Hokkaido University	Tritium Separation using Water Electrolysis and Fuel Cell with Polymer Electrolyte Membrane
P2-089	**Jonathan Coburn** Sandia National Laboratories	Retention and Surface Morphology Evaluation of Fine-grain Dispersion-Strengthened Tungsten for Plasma-Facing Component Applications
P2-090	**Arthur Adriaens** Ugent & KMS/ERM	Using TOMAS to study ITER and W7-X wall conditioning schemes
P2-091	**Kazunari Katayama** Kyushu University	Hydrogen transfer from pressurized water to gas phase through Inconel
P2-093	**Kenta Akashi** Kyushu University	Tritium permeation behavior from pressurized tritiated water through Inconel
P2-094	**Masahiro Tanaka** National Institute for Fusion Science	Thermal desorption of tritium from stainless steel as the first wall used in the LHD deuterium plasma experiment
P2-095	**Suguru Masuzaki** National Institute for Fusion Science	Study on depth profiles of tritium in the LHD plasma-facing wall using a sputtering treatment
P2-096	**Luke Olson** Savannah River National Laboratory	Evaluating Electrode Material Properties in an Electrolysis-Based Tritium Extraction Process
P2-097	**Alper Pahsa** HAVELSAN Inc.	Statistical Shannon Entropy of Potential Energy Computations in Molecular Dynamics of Retention of Tritium in Graphene Material
P2-098	**Gianluca D'Ovidio** CIEMAT	The spontaneous ignition of liquid lithium pools in controlled gaseous atmospheres
P2-099	**Zongxiao Guo** Shenzhen University	Optimization and Thermal Performance Evaluation of Chamfering Structures for the ITER-like W/Cu monoblocks Divertor in EAST

Tentative

P2-100	**Yueyang Gu** Institute of Metal Research, Chinese Academy of Sciences	Evolution of microstructure and changes in hydrogen embrittlement resistance of CLAM steel after long-term aging
P2-101	**Shane Evans** University of New Mexico	Structure and Composition Effects on Advanced Multicomponent-W for First Wall Applications
P2-102	**Da Chen** Southeast University	An Comprehensive Evaluation of Displacement Damage and Radioactivity of High Entropy Alloys Under Typical Fission and Fusion Neutron Spectrum
P2-103	**Saleem Lubbad** Materials Department, The University of Oxford	Data-Driven Approach for Evaluating Small Punch Test Results of Irradiated Materials Using Finite Element and Gaussian Process Regression Surrogate Modelling
P2-104	**Weidong Ding** China Institute of Atomic Energy	Simulation and Analysis of High-Efficiency Operating Modes for Helium-Hydrogen Separation Bed
P2-106	**Yuta Tobata** National Institutes for Quantum Science and Technology (QST)	Fracture behavior of laser weld joints with reduced activation ferritic/martensitic steel F82H
P2-107	**Ibrahim Karaman** Department of Materials Science and Engineering, Texas A&M University	Transitioning from Tungsten to low-activation Steels using Laser-Directed Energy Deposition for Fusion Applications
P2-108	**Samara Levine** Tokamak Energy	Effect of Vacuum Environment on the Electron Beam Welding of Vanadium Alloys
P2-109	**Masayuki Tokitani** National Institute for Fusion Science	Development of W/stainless steel first-wall component with ODS-Cu intermediate layer for high-temperature wall in QUEST

Tentative

P2-110	**Ryui Kanai** Yokohama National University	Development of Joining Technology and Analysis of Sustainability of FeCrAl Alloy as Inner Lining of Liquid Metal Blanket Piping
P2-111	**Nerea Garcia** National Fusion Laboratory (CIEMAT)	Mechanical characterization of copper-stainless steels joints in IFMIF-DONES oriented SSTT specimens
P2-112	**Xuebing Peng** Institute of Plasma Physics, Chinese Academy of Sciences	Qualification tests for tube-to-tube welding of CuCrZr to Inconel 625
P2-113	**Arata Nishimura** National Institute for Fusion Science	Conceptual design of welding between vacuum vessel sectors for a torus fusion device
P2-114	**Qingyan Xue** Hokkaido University	Effect of cold rolling and annealing temperature on microstructure of diffusion bonded Cr0.8FeMn1.3Ni1.3 / F82H joint
P2-115	**Ramu Ambati** Renaissance Fusion	Corrosion of austenitic steels (316L, 316Ti, 310S and 317) and their welding in molten tin at 900oC
P2-116	**Haotian Sun** Hokkaido University	Dislocation - mediated rapid atomic diffusion in diffusion bonded reduced-activation $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$ high entropy alloy/316 stainless steel joints
P2-117	**Hua Xu** Institute of Plasma Physics, Chinese Academy of Sciences	The novel methods for achieving high-strength W/steel bonding via regulating interface diffusion behavior or induced interface alloying

Tentative

P2-118	**Kaito Mizuno** Department of Quantum Science and Energy Engineering, Tohoku University	Evaluation of grain boundary strength of tungsten materials using micro-cantilever bending test
P2-120	**Eun-chae Jeon** University of Ulsan	A Nanoindentation Study on Characterization of Irradiation Damage Depth in RAFM steel
P2-121	**Hanqing Wang** School of Physics, Beihang University	Temperature-dependent defect evolution of displacement damage under heavy ion irradiation in CVD tungsten.
P2-122	**Minghuan Cui** Institute of Modern Physics, Chinese Academy of Sciences	Microstrcture and nano-hardness evolution in tungsten under strain field
P2-123	**Jarmila Degmova** INPE FEI STU	Probing Helium-Induced Defect Evolution in Ferrous Alloys for Fusion Application: Insights from Positron Annihilation Spectroscopy and Multi-Energy Ion Implantation
P2-124	**Ke Jin** Beijing Institute of Technology	Towards reliable evaluation of ion-irradiation induced mechanical property change based on micro/nano-mechanical testing
P2-125	**Christoph Hugenschmidt** Technische Universitat Munchen	Irradiation Induced Defects in W-Re Alloys Studied by Positron Annihilation Spectroscopy
P2-126	**Long Cheng** Beihang University	Retarded recrystallization of tungsten by helium irradiation: simulation and experiment

Tentative

P2-127	**Raquel Gonzalez-Arrabal** Universidad Politecnica de Madrid	Plasma facing materials in inertial and magnetic confinement: peculiarities of radiation environment and radiation-induced damage
P2-128	**Wentuo Han** University of Science and Technology Beijing	Anomalous Exothermic Behavior of Bubbles in Metals Induced by Electron Irradiation
P2-129	**Chao Li** Karlsruhe institute of technology	Constitutive Modelling of Ratcheting Behaviour of Eurofer97 at High Temperatures
P2-130	**Yasuyuki Ogino** Institute for Materials Research, Tohoku University	Optimization and feasibility study of radiation shielding materials for fusion reactors by radiation transport simulation
P2-131	**Chao Hai** Institute of Nuclear Energy Safety Technology (INEST), Hefei Institutes of Physical Science (HFIPS),	Advanced Design and Integrated Manufacturing of Low-Activation Steel/Tungsten Functionally Graded Materials for Plasma-Facing Component Applications
P2-132	**Alvaro Martinez-Pechero** University of Oxford	Modelling fracture properties of single-crystal Beryllium as a function of crystal orientation
P2-133	**Zihao Gao** Institute of Modern Physics, Chinese Academy of Sciences	Advanced deep learning framework for multi-scale prediction of mechanical properties from microstructural features in polycrystalline materials
P2-134	**Donggyu Lee** Seoul National University	Development of an open platform for hydrogen isotope properties data

Tentative

P2-135	**Axel Klix** Karlsruhe Institute of Technology	Measurement of water activation with collimated germanium detectors at the JSI KATANA facility
P2-137	**Philipp Lied** Fusion Catalyst, Karlsruhe, Germany	MatDB4Fusion: A new initiative to collect, merge, and leverage fusion material properties data
P2-138	**Andrew Warwick** United Kingdom Atomic Energy Authority	A comprehensive ab initio database of radiation-induced point defect for metals across the Periodic Table
P2-139	**Nils Holstein** Karlsruhe Institute of Technology	Correlation of material properties and monitoring of dissolved impurities via electrochemical methods: detecting hydrogen in liquid lithium loops of IFMIF-DONES
P2-140	**Ang Xu** Huazhong University of Science and Technology	Research on Low-Cost Preparation and Radiation Resistance of Textured MAX Phases
P2-141	**Yibin Dong** Shandong University	Enhanced resistance to radiation by periodic pattern of dislocation networks at a W/Fe interface
P2-142	**Xinqing Han** Institute of Frontier and Interdisciplinary Science and Key Laboratory of Particle Physics	Unraveling the Atomic Mechanism of the Crystalline Phase-Dependent Structural Features and Special Spectral Design of α-, β-, and ε-Ga_2O_3
P2-143	**Elena Botica-Artalejo** Massachusetts Institute of Technology	Advances in rapid radiation damage-resistant alloy selection
P2-146	**Kuan-Che Lan** National Tsing Hua University	Radiation Shielding Effectiveness of Aluminum Alloy 6061 with CoCrFeNi thin film for P-type Si(100) against 230 MeV Proton

Tentative

P2-147	**Gabriel Gorelick** UCLA	Raytracing Monte Carlo model of electronic thermal conductivity in single crystal W and W-Re alloys for Fusion Environments
P2-148	**Gabrielle Koknat** Sandia National Laboratories	Modeling Non-Equilibrium Ion Dynamics in Tungsten Alloys: A Machine Learning Approach to Radiation Damage
P2-149	**Tomi Vuoriheimo** University of Helsinki	Ion energy dependence on irradiation-induced defect stabilization effect in tungsten
P2-150	**Blanca Biel** Universidad de Granada	Beyond DFT: Development of a Fe-C fusion ready potential within the NEP approach
P2-151	**Alexandre Bergero** University of Helsinki	Sputtering of High Entropy Alloys
P2-152	**Maxwell Rae** Massachusetts Institute of Technology	Large-Area Measurement of Ion Irradiation Induced Voids Without TEM
P2-153	**Anirvinya Tirumala** United Kingdom Atomic Energy Authority (UKAEA)	An MD-Driven Bayesian Model for Void Evolution in Irradiated and Gas-Implanted Tungsten
P2-154	**Liang Xia** Peking University	Revealing the temperature-dependent hardening mechanisms of non-uniformly distributed defects for ion-irradiated metals
P2-155	**Sho Hayakawa** United Kingdom Atomic Energy Authority	Model for the mechanical response of a tensile loaded heavily irradiated material
P2-156	**Qianfei FENG** University of Science and Technology Beijing; Hokkaido University	Effect of oxide particles on swelling behaviour in Cu and Cu alloys under in-situ electron irradiations.

Tentative

P2-157	**Kaito Kamiyama** Graduate School of Engineering, Tohoku Univ.	Effect of low-temperature irradiation on hardness and microstructural evolution in Fe-Cr binary alloys
P2-158	**Bowen Zhang** Peking University	Object kinetic Monte Carlo simulations on the difference between fission neutron and heavy ion irradiation induced void evolution in Fe-Cr alloys
P2-159	**Abel Gutierrez-Camacho** Universidad de Alicante	Towards a standardised methodology of radiation damage defect distributions for microstructure evolution models
P2-160	**CHANDRA BHUSAN YADAV** University of Bristol, United Kingdom	Irradiation Defect Evolution in FeCr alloys and Tungsten for Fusion Application: The Role of Temperature and Dose Gradients
P2-161	**Zhiwei Hu** CNRS-CEMHTI	Lattice Strain and Damage Accumulation in Self-Ion Irradiated Tungsten: Insights from PAS and XRD Analysis
P2-162	**Jintong Wu** University of Helsinki	High dose irradiation of tungsten with realistic dose rate and atomistic resolution
P2-163	**Yuhan Sun** Beihang University	The influence of grain boundary migration and grain size on the irradiation damage behavior of tungsten
P2-164	**Adria Peterkin** Massachusetts Institute of Technology	Modeling Radiolysis Effects in FLiNaK and FLiBe Molten Fluoride Salt
P2-165	**Yafeng Shu** Institute of Modern Physics, Chinese Academy of Science	Data driven service life prediction of nuclear silicon carbide composite materials

Tentative

P2-167	**Salahudeen Mohamed Kunju** Karlsruhe Institute of Technology	Microstructural Evolution of Irradiation-Induced Defects in Tungsten: cluster dynamics study
P2-168	**Hanns Gietl** Idaho National Laboratory	Evaluating neutron activation to determine the value of RAFM steels in fusion, thermal, and fast reactors to accelerate collaborative development
P2-169	**Jonathan Hinks** University of Huddersfield	Microscope and Ion Accelerators for Materials Investigations (MIAMI-2): Status Update
P2-170	**Enrique Martinez** School of Mechanical and Automotive Engineering, Clemson University	Thermodynamic and Kinetic Properties of Defects in Concentrated Alloys
P2-171	**Yasuhiro Kamada** Iwate University	Fe+ irradiation effects on the microstructure and magnetic properties of Fe-10at%Cr films
P2-172	**Charles Lieou** University of Tennessee	Thermal and irradiation creep: A thermodynamic theory of coupling between point-defect diffusion and dislocation plasticity
P2-173	**Zhaonan Ding** Institute of Modern Physics, Chinese Academy of Sciences	Hardening and microstructure evolution of FeCrAl ODS steel under high-energy heavy ion irradiation
P2-174	**Karen Pacho Dominguez** University of Birmingham	In-situ irradiation effects and Ti(CON) stability in baseline and precipitation-strengthened V-4Cr-4Ti alloys
P2-175	**Paul Bagot** University of Oxford	Neutron irradiation and thermal annealing on W-alloys for fusion applications; recent Atom Probe Tomography, TEM and nanoindentation investigations

Tentative

2nd October 2025 (Thursday)

Time 16:30-18:30 Gallery Room Chairs: T. Nozawa (QST), C. N. Taylor (INL), Y. Mao (FZJ), M. Klimenkov (KIT), Y. Lin (ORNL)		
P3-001	**Christina Manoli** Karlsruhe Institute of Technology	Helium transmutation damage studies on EUROFER97 using ion implantation
P3-002	**Christina Hofer** University of Oxford	Advancing the understanding of radiation damage in reduced activation CCAs by atom probe tomography
P3-003	**Claudio Guarnaschelli** Rina Consulting - CSM S.p.A.	High temperature and fatigue behavior of RAFM microstructures optimized for DEMO Water Cooled Breeding Blanket
P3-004	**Carlo Cristalli** ENEA	Strategies to lower the DBTT of RAFM steels acting on chemical composition and thermomechanical treatments
P3-005	**Yuguang Chen** Institute of Modern Physics, Chinese Academy of Sciences	Investigation of the Nano-Mechanical Plastic Deformation Behavior of Heavy-Ion Irradiated RAFM Steel
P3-006	**Josina Geringer** Oak Ridge National Laboratory	Physical and mechanical properties of EUROFER97 after 20 dpa neutron irradiation
P3-007	**WENHAI GUAN** National Institutes for Quantum Science and Technology	Evaluation and prediction of Creep-fatigue lifetime of F82H for Water-Cooled Ceramic Breeder Blanket Application

Tentative

P3-008	**James Stubbins** University of Illinois at Urbana-Champaign	Investigation of Neutron Irradiation Effects on T91 Ferritic/Martensitic Steel for Fusion Reactor Applications
P3-009	**Sungyu Kim** Korea Institute of Fusion Energy	Optimization of Normalizing and Tempering Conditions for Microstructure and Mechanical Properties of ARAA
P3-010	**Lei Peng** University of Science and Technology of China	Post Irradiation Examination for Chinese Fusion Reactor Structural Steels Irradiated in STIP-VIII
P3-011	**Yuping Xu** Institute of Plasma Physics, Hefei Institutes of Physical Science, Chinese Academy of Sciences	Synergistic effects of displacement damage and helium on the deuterium permeation in the oxidized RAFM steel under dual-beam irradiation
P3-012	**Zhiqiang Fan** University of Science and Technology of China	Effect of thermal aging on microstructure, mechanical properties and corrosion resistance of reduced activated ferritic-martensitic and oxide dispersion strengthened steel
P3-014	**Jason Trelewicz** Stony Brook University	Mechanical properties of castable and sintered nanostructured low-activation ferritic-martensitic steels
P3-015	**Kazuyuki Furuya** Mechanical and Medical Engineering Course, Department of Industrial System Engineering	Mechanical Property and Irradiation Hardening in Low-activation Dual-phase Fe-Mn-Cr-Al-V-C Alloy
P3-016	**Zhao Meng** Institute of Modern Physics, Chinese Academy of Sciences	Screening and manipulation by segregation of dopants in grain boundary of Silicon carbide: First-principles calculations

Tentative

P3-017	**Taichi Miyagishi** Graduate School of Engineering, Tohoku University	Unveiling the Mechanism of Irradiation-Induced Flow in Mullite Ceramics Under Particle Irradiation
P3-018	**Shota Abe** Princeton Plasma Physics Laboratory	Silicon-carbon bond formation and deuterium retention on a silicon crystal surface in the DIII-D divertor
P3-019	**Roxane Marull** CEA Saclay - SRMA	Double walled heat exchanger tube design and fatigue cracking for Water Cooled Lithium Lead breeder module
P3-020	**Abdulmohssen Abaalkhail** Department of Mechanical and Aerospace Engineering, University of Miami	A Microstructure-based Viscoplastic Model of First Wall & Blanket Fusion Materials
P3-022	**Ziwei Li** SCK CEN	Investigation of neutron irradiated nuclear structural materials using Small Punch Test
P3-024	**Yifan Zhang** Hefei University of Technology	The effect of nanoparticle/matrix interface sink strength on the irradiation response of Cu-Y_2O_3 under He/D sequential irradiation
P3-025	**Marco Lamberti** ENEA	Low-cyclic fatigue behaviour of CuCrZr alloy for the DEMO divertor heat sink
P3-026	**Parth Kulkarni** The Open University, Milton Keynes	Multiscale Mechanistic Study of Creep Behaviour in Copper-Chromium-Zirconium Alloy Heat Sinks for Fusion Devices Using Advanced Experimental Methods
P3-027	**Yiran Mao** Forschungszentrum Juelich GmbH, IFN-1	Plasma facing components based on advanced high temperature heat pipes-concepts and design

Tentative

P3-028	**Akira Hasegawa** Institute for Materials Research, Tohoku University	Three-point bending behavior of W-Ta alloys after BR-2 irradiation at 290°C
P3-029	**Petra Jenus Belec** Jozef Stefan Institute	Development of W_2C-reinforced W as high heat flux material
P3-030	**Yann DE CARLAN** Universite Paris-Saclay, CEA, Service de Recherche en Materiaux et Procedes Avances	Development of reduced activation ODS steels by standard methods and additive manufacturing
P3-032	**Ian McCue** Northwestern University	Enhancing the performance of plasma-facing materials by laser powder-bed fusion
P3-033	**Zori Harutyunyan** Imperial College London	Radiation-Induced Effects in Tungsten-Based High-Entropy Alloys for Plasma-Facing Components in Fusion Devices
P3-034	**Zheng Wang** Beihang University	Deuterium behavior in damaged tungsten-rhenium alloy
P3-035	**Andrey Ushakov** TNO	Study of ion sputtering and degradation of Platinum- and Rhodium-coated first mirrors for optical diagnostics in fusion devices
P3-036	**Mehdi Firdaouss** CEA/IRFM	Ex-situ and in-situ laser cleaning of divertor tungsten surface for tokamak
P3-037	**Wuqingliang Peng** Institute of Plasma Physics Chinese Academy of Sciences	Low high-temperature hardness: A drawback of current ODS copper applied in monoblock-type components fabricated by HIP
P3-038	**John Cooper** University of Birmingham	Revealing deformation behaviour of W-Cu composites via in situ electron microscopy, optical microscopy, and neutron diffraction

Tentative

P3-039	**James Haag** Pacific Northwest National Laboratory	Microstructural evaluation of ductile phase toughened tungsten heavy alloys
P3-040	**Yifei Hu** Huazhong University of Science and Technology	Fabrication of High-Performance Tungsten-Based High-Entropy Alloy Complex Structures via Gel Casting and Sintering Additives
P3-041	**Xiaolei Ma** Peking University	Preparation and characterization of large-scale potassium-doped tungsten alloy
P3-042	**David Sprouster** Stony Brook University	Engineering Advanced Copper Alloys for Fusion Energy
P3-044	**Soon Han Bryan Teo** Australian National University	Sample temperature dependence of the post-plasma thermal evolution of He bubbles in tungsten and tungsten-based refractory alloys
P3-045	**Aljaz Ivekovic** Jozef Stefan Institute	Effect of baseplate preheating on the Laser Powder Bed Fusion of W-W_2C
P3-046	**Xiu-Li Zhu** North China Electric Power University	Comparison of deuterium retention in rolled and recrystallized tungsten with and without pre-damage
P3-047	**Yuriy Gordienko** Institute of Atomic Energy Branch of the National Nuclear Center of the Republic of Kazakhstan	Prospects for plasma-facing material research on the KTM tokamak for future fusion power plants
P3-048	**Arseniy Kuzmin** Kyoto University	Formation, Surface Characterization, and Stability of Boron-Containing Films Under Plasma Irradiation
P3-049	**Makoto Fukuda** National Institutes for Quantum Science and Technology	Recrystallization behavior of tungsten for the ITER divertor outer vertical target

Tentative

P3-050	**Xiaoyue Tan** Hefei University of Technology	Research summary on the self-passivating tungsten alloy for the prospective application in a fusion power plant
P3-051	**Gibum Kim** Seoul National University	Experimental and Computational Study of Hydrogen Permeation in Tungsten with Copper Coated Surface
P3-052	**Jan Horacek** Institute of Plasma Physics, Czech Academy of Sciences	Innovative STEAM PFC for steady Ultra-High Heat Flux Tokamak Divertors
P3-053	**Tong Zuo** Hefei University of Technology	Dislocation-controlled plastic deformation characteristics of W-2.0wt% Y_2O_3
P3-054	**Thomas Derra** Fraunhofer Institute for Production Technology IPT	Adaption of wire-based laser metal deposition for the regeneration of tungsten armored plasma-facing components
P3-055	**Jun Chen** Institute of Plasma Physics, Chinese Academy of Sciences	Thermal performance of W/Cu flat-type components as the lower divertor target in EAST
P3-056	**Simon Corah** School of Metallurgy and Materials, University of Birmingham	PLASMA MATERIAL INTERACTIONS IN ADVANCED TUNGSTEN BASED MATERIALS FOR FUTURE PLASMA FACING COMPONENTS
P3-057	**Pei Wang** FEAT SRTT & IMRE, A*STAR	Alloy development for metal additive manufacturing through a combinatory method using laser powder bed fusion
P3-058	**Yiwen Sun** Beihang University	Deuterium retention in recrystallized tungsten exposed to high-flux plasma with fluences up to 1×10^{29} m^{-2}
P3-059	**Mengxia Liang** Xihua University	Performance of doped tungsten with high potassium content under high-flux helium plasma irradiation

Tentative

P3-060	**Keonwook Kang** Yonsei University	Atomistic simulations of cumulative irradiation damages in single-crystalline tungsten and analysis of secondary defects
P3-061	**Xiang Zan** Hefei University of Technology	Heterogeneous recrystallization behavior of warm-rolled yttria dispersion strengthened tungsten plate along thickness direction
P3-062	**Yucheng Wu** Hefei University of Technology	Review on design strategy and scalable production route of tungsten-based materials for nuclear fusion application
P3-063	**Ladislav Vala** Research Centre Rez	Current Status of High Heat Flux Testing at the HELCZA facility
P3-064	**Jun Huang** School of Materials Science and Engineering, Hefei University of Technology	Effects of disordered interfacial transition layer of W-Y_2O_3 on irradiation and mechanical properties
P3-065	**Feng Ren** Wuhan University	Enhanced resistance to helium irradiations in nano-structural high-entropy-alloy plasma-facing materials
P3-066	**Daisuke Nishijima** University of California, San Diego	In-operando spectroscopic ellipsometry measurement of W surface during high-flux plasma exposure
P3-067	**Taehyun HWANG** QST	Overview of the mechanical and thermal properties of plasma sintered $Be_{12}Ti$
P3-068	**Arturs Zarins** University of Latvia	Influence of various types of ionising radiation on the formation of radiation-induced defect centres in advanced ceramic breeder pebbles
P3-069	**Y. Shi** University of Science and Technology of China	In situ characterization of 3D deformation and damage evolution of LiH ceramic under different thermal-mechanical load via DIC and DVC

Tentative

P3-070	**Xue Xie** Hefei University of Technology	Preparation and irradiation performance of advanced tritium breeder material: lithium lead titanate
P3-071	**Wenhao Wu** Institute of Nuclear Energy Safety Technology, Hefei Institutes of Physical Science, Chinese Academy	Study on Tritium Release Behaviors of Li_2TiO_3 Breeder Irradiated by Fusion Neutrons
P3-072	**Guangfan Tan** Tsinghua University	Study on tritium release behavior of ceramic pebbles fabricated by Centrifugal granulation method
P3-073	**Kenji Morita** Rokkasho Institute for Fusion Energy, National Institutes for Quantum Science and Technology	Status of Lithium-6 Enrichment using Electrodialysis with Ionic Conductor Membrane
P3-074	**Roman Petras** Research Centre Rez	Effect of Pb-16Li on the mechanical properties of ceramic materials after long-term exposure
P3-075	**Yi-Hyun PARK** Korea Institute of Fusion Energy	Preliminary Study on Tritium Release Properties of Li_2TiO_3 Pebbles Manufactured by PIM Process
P3-076	**QIN ZHAN** China Institute of Atomic Energy	Tritium production performances of Li_4SiO_4 solid tritium breeder prepared by wet method under reactor irradiation
P3-077	**Guangming Ran** Institute of Nuclear Physics and Chemistry, China Academy of Engineering Physics	Simulations of tritium release behavior of single-phase and two-phase ceramic breeder materials

Tentative

P3-078	**Marcelo Roldan Blanco** CIEMAT	Unveiling the in-depth microstructure of the Advanced Ceramic Breeder pebbles: the first comprehensive STEM exploration
P3-079	**Asset Shaimerdenov** The Institute of Nuclear Physics	Study of microstructural degradation of chromium beryllide under neutron irradiation
P3-080	**Maulik Panchal** Institute for plasma research	Experimental Quantification of Anisotropy in Effective Thermal Conductivity of Compressed Pebble Beds for Fusion Blankets
P3-081	**Tommaso Sgrilli** Gencoa Ltd	Functionalisation of fusion reactor grade structural materials via the physical vapour deposition of ceramic coatings
P3-082	**Hongtao Huang** Sun Yat-sen University	A high performance and dense Fe-Al/Al_2O_3 tritium permeation barrier prepared on 316LSS at low temperature
P3-083	**Hazel Gardner** UKAEA	Irradiation stability and tritium retention characteristics of novel ceramic coatings for fusion breeder blanket applications.
P3-084	**Teruya Tanaka** National Institute for Fusion Science	Examination of degradation in electrical insulating performance of Y_2O_3 coating layer under ion beam irradiation
P3-085	**Juro Yagi** Institute of Advanced Energy, Kyoto University	Development of anti-oxidation layer on fusion reactor material for the usage of supercritical CO_2
P3-086	**Timur Kulsartov** Satbayev University	Development of a Phenomenological Model of Beryllide Corrosion in Water Vapor
P3-087	**Magzhan Aitkulov** Institute of Nuclear Physics, Almaty, Kazakhstan	Investigation of chromium beryllide corrosion processes during high-temperature interaction with a gas mixture

Tentative

P3-088	**Nayoung Kim** MIT	Effects of Cationic Impurities on the Corrosion of Ni-based Alloys in Molten Fluoride Salts
P3-089	**Yichun Xu** Institute of Solid State Physics, Hefei Institutes of Physical Science, Chinese Academy of Sciences	The simulation study on the corrosion behavior of Fe-based alloys in liquid lead-lithium under nuclear-fusion devices
P3-090	**Zhanna ZAURBEKOVA** Institute of Nuclear Physics	Study of corrosion processes of chromium beryllides and titanium beryllide with water vapor at critical vapor parameters
P3-091	**Yi-Ming Lyu** Institute of Energy, Hefei Comprehensive National Science Center	Corrosion behavior of CLF-1 steel with different surface roughness in a magnetic field environment
P3-092	**Daniel Morrall** Savannah River National Lab	Microstructural effects on He bubble distribution from tritium decay.
P3-093	**Ashwin Persaud** University of Birmingham	UNDERSTANDING THE ROLE OF HELIUM-INDUCED CAVITIES ON TRITIUM TRAPPING IN FUSION MATERIALS
P3-094	**Guiomar Soria** CIEMAT	Identifying the He and D species trapped in selected fusion materials
P3-095	**Yuriy Ponkratov** Branch of the Institute of Atomic Energy of the National Nuclear Center	Study of tritium and helium generation and release processes from Ga-Li alloy under high-density neutron radiation
P3-096	**Marta Malo** CIEMAT	Effects of helium production and radiation in advanced permeation barriers

Tentative

P3-097	**Teresa Hernandez** CIEMAT	Stress corrosion cracking on Eurofer subjected to Li ceramics
P3-099	**Brenda Garcia-Diaz** Savannah River National Laboratory	FC FIRE: Advancing Fuel Cycle Solutions for Fusion Energy's Future
P3-100	**Yuriy Ponkratov** Branch of the Institute of Atomic Energy of the National Nuclear Center	Technology of synthesis of Ga?-Li? alloys with different lithium content and characterization of the obtained alloys by SEM, XRD and DSC methods
P3-101	**Frances Livera** United Kingdom Atomic Energy Authority	Manufacturing Optimisation for Joining of CuCrZr and Eurofer-97 for the DEMO Limiter Component
P3-102	**Mario Oliver** Kyoto Fusioneering	OpenMC-DONES source: Development and validation of IFMIF-DONES neutron source for OpenMC from McDeLicious-MCNP6 d-Li interaction results
P3-103	**Dilmurod TUYMURODOV** Lithuanian Energy Institute	Impact of Bolt Penetration Size on Radiation Streaming and Dose Rates in the IFMIF-DONES Test Cell Floor
P3-104	**Sergej Gordeev** Karlsruhe Institute of Technology, KIT-Campus-Nord	Thermal-Hydraulic Analysis on New DONES -HFTM Design
P3-105	**Santiago Becerril** Consorcio IFMIF-DONES Espana	Optimization of the electrical harnesses for the in-vessel systems at the IFMIF-DONES Test Cell
P3-106	**Yuefeng Qiu** Karlsruhe Institute of Technology	Correlation study of material damage and activation inventory for EUROFER, tungsten and CuCrZr alloy in the IFMIF-DONES irradiation environment

Tentative

P3-107	**Guangming Zhou** Karlsruhe Institute of Technology	A blanket functional materials irradiation module in IFMIF-DONES for Helium Cooled Pebble Bed breeding blanket
P3-108	**Arkady Serikov** Karlsruhe Institute of Technology (KIT)	Neutronics analysis of the structural and functional materials within the IFMIF-DONES Test Modules
P3-109	**Kentaro Ochiai** National Institutes for Quantum Science and Technology	Neutronics analysis for irradiation candidate materials with fusion neutron source
P3-110	**Evan Lambert** Massachusetts Institute of Technology	Low level dose characterization of 12 MeV proton irradiation on Inconel 718 alloy
P3-111	**Jae-Hoon Choi** School of Materials Science and Engineering, University of Ulsan	Defect formation behavior of ARAA steel under He ion irradiation
P3-112	**Marie Romedenne** Oak Ridge National Laboratory	Compatibility of pre-oxidized ODS FeCrAl under combined neutron irradiation and high temperature liquid metal effects
P3-113	**Thak Sang BYUN** Oak Ridge National Laboratory	Effect of Neutron Irradiation on the Mechanical Properties of Additively Manufactured 316L Stainless Steel and the Influence of Sampling Location
P3-114	**Roberto Coppola** ENEA	Neutron irradiation micro-structural effects in the low activation ferritic/martensitic steel EUROFER97
P3-115	**Charles Hirst** University of Wisconsin-Madison	Accelerated evaluation of swelling in F82H as a function of implanted H/He through dual gas-gradient triple-ion irradiation experiments

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P3-116	**Zhanfeng Yan** Institute of Nuclear Physics and Chemistry, China Academy of Engineering Physics	The effects of low temperature neutron irradiation on the mechanical property and microstructure evolution of Amosic-3 SiCf/SiC composites
P3-117	**Wahyu Setyawan** Pacific Northwest National Laboratory	Atomistic Study of Interaction and Diffusion of Solutes and Impurities in Fe-Cr
P3-118	**David Cereceda** Villanova University	Transmutation Effects on the Local Ordering Phenomena of Tungsten-Based Plasma-Facing Materials
P3-119	**Evgeniia Ponomareva** Department of Applied Physics, Aalto University	Electronic energy losses and sputtering mechanisms in Fe under light ion irradiation
P3-120	**Yuri Osetsky** ORNL	Radiation induced defects in TiB2: atomistic modeling and experimental validation
P3-121	**Timothy Lach** Oak Ridge National Laboratory	Post-irradiation examination of neutron-irradiated W/Cu interfaces
P3-122	**Ivan Tolkachev** University of Oxford	An investigation of irradiation induced grain growth in iron using x-ray diffraction experiments and molecular dynamics simulations
P3-123	**Sergei Dudarev** UKAEA	From microscopic defects to a full tokamak device: the European effort on modelling radiation effects in fusion reactor materials
P3-124	**Roshan Rajakrishnan** Karlsruhe Institute of Technology	Modeling the deformation and ductile damage of irradiated EUROFER97 for fusion reactor design optimization

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P3-125	**Blanca Biel** University of Granada	Application of DFT and machine-learning trained potentials to study radiation-induced defects in ferrite-cementite interfaces
P3-126	**Alexis Devitre** Massachusetts Institute of Technology	Cryogenic ion irradiation of high-temperature superconductors with in situ positron spectroscopy & AC susceptibility measurements of critical temperature
P3-127	**Li Yang** University of Tennessee	Defect formation, diffusion, and recombination in $YBa_2Cu_3O_7$-δ in a fusion environment: A first-principles study
P3-128	**JongHyeon Park** Seoul National University	Molecular dynamics simulations of radiation damage processes and effects near grain boundaries in tungsten
P3-129	**German Samolyuk** Oak Ridge National Laboratory	First principles-based approach to calculation of electrical and thermal conductivity in functional and structural materials
P3-130	**Victor Lindblad** University of Helsinki	Understanding deuterium's effect on defect production in irradiated tungsten
P3-131	**Hao Pan** Hefei Institutes of Physical Science, Chinese Academy of Sciences	Accurate, Robust and Interpretable Property Prediction Framework for Reduced-Activation High-Entropy Alloys with Small Datasets
P3-132	**Jingyi Shi** University of Science and Technology of China	Study on the Mechanism of Nanoparticle Induced Void Nucleation in ODS Steel under Complex Stress States
P3-133	**Zhixi Zhu** Beijing Institute of Technology	Distinct ion irradiation responses in refractory multi-principal-element alloys

Tentative

P3-134	**Sophie Blondel** University of Tennessee, Knoxville (USA)	Cluster Dynamics Simulations of Cascade Overlap Model for 800H Alloy
P3-135	**Neelam Meena** Karlsruhe Institute of Technology	In-situ TEM investigation of irradiation induced defect evolution in neutron-irradiated Tungsten
P3-136	**Ashley Dickson** Lancaster University	A high-fidelity machine learned interatomic potential for radiation damage in $YBa_2Cu_3O_7$
P3-137	**Ning Gao** Shandong University	Effect of radiation defects on grain boundary evolution under shock loading
P3-138	**Somei Ohnuki** University of Science and Technology Beijing	Evaluation of internal sink intensity of nano-particles in neutron-irradiated ODS steels
P3-139	**Artem Kozlovskiy** Satbayev University	Study of the influence of irradiation temperature on the destabilization mechanisms of the near-surface layer of BeO ceramics
P3-140	**Mingxin Chen** Hefei University of Technology	Study on the Influence of Alloy Elements on the Radiation Effect of Vanadium based Alloys by Positron Annihilation
P3-141	**Linyun Liang** Beihang University	Machine learning-based multi-scale simulation approach for predicting effective thermal conductivity in tungsten monoblock
P3-142	**Sijie Hao** Beihang University	In-situ Scanning Electron Microscopy Study of the Thermal Creep Bebavior in Body-Centered Cubic Metals
P3-143	**Ousmane Ndiaye** University of Illinois Champaign Urbana	Microstructural evolution study of ion irradiated Laser Direct Energy Deposition and Laser Powder Bed Fusion 316H.

Tentative

P3-144	**George Apostolopoulos** National Centre for Scientific Research "Demokritos"	Damage saturation in low-temperature ion irradiation of Fe thin films observed by in-situ electrical resistivity measurements
P3-145	**Yasuhiro Kamada** Iwate University	Effects of cavity formation on the magnetism of iron and iron-chromium alloys
P3-146	**Maria Caturla** Universidad de Alicante	Damage by MeV ion implantation in Fe and FeCr from OKMC simulations
P3-147	**Nima Fakhrayi Mofrad** Aalto University	Computer study of sputtering in fusion-related materials: An in-depth analysis of sputtered species
P3-148	**Jason Trelewicz** Stony Brook University	Interactions of Transmutation Products with Radiation Induced Point Defects in Tungsten from First Principles and Machine Learning
P3-150	**Pengbo Zhang** Dalian Maritime University	Migration mechanisms of mono-/di-interstitials in vanadium with alloying Ti/Cr
P3-151	**Tymofii Khvan** National Centre for Nuclear Research	Predicting irradiation hardening and microstructural evolution in ion-irradiated Eurofer97: a nanoindentation study supported by CPFEM and TEM
P3-152	**Mingliang Wei** Dalian Maritime University	The mechanism of titanium suppressing the clustering behavior of vacancy in vanadium alloys
P3-153	**Jianyang Li** Institute of Modern Physics, Chinese Academy of Science	Study of the effect of self-ions on the microstructural evolution of irradiated Fe-Cr alloys
P3-154	**Seungho Lee** Seoul National University	Machine Learning Interatomic Potential for Atomistic Simulations of Nanostructure Formation on He-Irradiated Tungsten Surfaces

Tentative

P3-155	**Chonghong Zhang** Institute of Modern Physics, Chinese Academy of Sciences	Small punch test of deformation behavior of ultra-thin steel specimens irradiated with high-energy heavy ions
P3-156	**Taku Ishida** J-PARC Center / KEK	Contrasting Irradiation Behavior of Two Phases in Titanium Alloys at Low-Temperature Due To Omega-Phase Precursors in the Beta-Phase Matrix
P3-157	**Yoshitaka Matsukawa** Kumamoto University	Phase diagram of the Γ2-phase: C15 Laves phase in RPV steel
P3-158	**Jin-Li Cao** China Institute of Atomic Energy	Resolving Cr segregation at grain boundaries in Fe-Cr alloys by Ising-machine accelerated Monte Carlo optimization
P3-159	**Kiyohiro Yabuuchi** Institute of Advanced Energy, Kyoto University	Dynamics study of transformation of a stacking fault tetrahedron to a void in FCC alloy.
P3-160	**Giacomo Po** University of Miami	A coupled Dislocation Dynamics and Cluster Dynamics model of irradiation creep and growth
P3-161	**Ane Lasa Esquisabel** University of Tennessee Knoxville	Assessing the long-term performance of the ARC divertor
P3-162	**Ryuta Kasada** Institute for Materials Research, Tohoku University	High Mn austenite steel revisited: novel alloy design concepts and material properties
P3-163	**Cody Dennett** Commonwealth Fusion Systems	Materials development for the ARC fusion power plant
P3-164	**Lee Aucott** UK Industrial Fusion Solutions Ltd	An overview of the STEP programme, materials strategy, and materials R&D activities.

Tentative

P3-165	**REBECA HERNANDEZ PASCUAL** CIEMAT	Development of advanced manufacturing technologies to produce large-scale components to enhance the competitiveness of the Spanish industrial sector in Fusion(RODAS)
P3-166	**Patrick McCarthy** Gencoa Ltd	Design and Development of an Optical Gas Sensor for the ITER DRGA System
P3-167	**Fatimah Sanni** United Kingdom Atomic Energy Authority	Performance Qualification of a Vacuum Pump Train with Modified Scroll Pump
P3-168	**Ming Chen** University of Science and Technology of China	A grain boundary engineering approach to enhance intergranular damage resistance in reduced activation ferritic martensitic steel
P3-169	**Junfeng Yang** Hefei Institutes of Physical Science, Chinese Academy of Sciences	Amorphous FeCrAlTiSi high entropy alloy coating prepared using vacuum ion plating
P3-170	**Min SU Ha** Korea Institute of Fusion Energy	Geometric signals of PAUT in various TIG welding configurations of 316L(N)-IG
P3-171	**Thak Sang BYUN** Oak Ridge National Laboratory	Mechanical Property Assessment of FM and ODS Steels Before and After Irradiation
P3-172	**Luis SEDANO** FUS ALIANZ Science, Engineering and Consulting	A Material Production and Qualification Plan for lead-lithium eutectic Li(6) enriched nuclear grades
P3-173	**Oliver Leys** Karlsruhe Institute of Technology	Continuous Production of Advanced Ceramic Breeder Pebbles

Tentative

P3-174	**Lingfeng Yang** North China Electric Power University	Hardening of T91steel induced by self-ion irradiation after He implantation at 450℃

Tentative

Abstracts

Tentative

Keynote 1

Fusion Energy Innovation Strategy

Kazuhiro Sawada

MEXT

Tentative

Keynote 2

BUILDING BRIDGES – U.S. FUSION ENERGY SCIENCES VISION, STRATEGY, AND ROADMAP

Jean Paul Allain

Associate Director of the Office of Science for Fusion Energy Sciences

United States Department of Energy

Dr. Jean Paul Allain will outline the U.S. Department of Energy's vision, strategy, and ongoing roadmap development effort toward the accelerated development of fusion energy. The discussion will cover a range of topics, including the restructured Fusion Energy Sciences in the Office of Science (SC-FES) and its supporting programs, public-private partnerships and new programs bridging science to supporting a fusion power industry in the U.S., and international collaborations. Emphasis will be placed in outlining key challenges and gaps being defined in the Fusion Science & Technology Roadmap being developed by SC-FES in areas of materials, internal components, and fusion nuclear sciences.

Tentative

29PL1

Status and perspective of the European neutron irradiation program

Michael Rieth[1], Vladimir Chakin[1], Michael Duerrschnabel[1], Ermile Gaganidze[1], Michael Klimenkov[1], Carsten Bonnekoh[1], Nikolai Zimber[1], Hans-Christian Schneider[1], Dmitry Terentyev[2], Ildiko Szenthe[3], Gerald Pintsuk[4], Giacomo Aiello[5], T S Byun[6], Tim Graening[6], X Chen[6], Yutai Katoh[6]

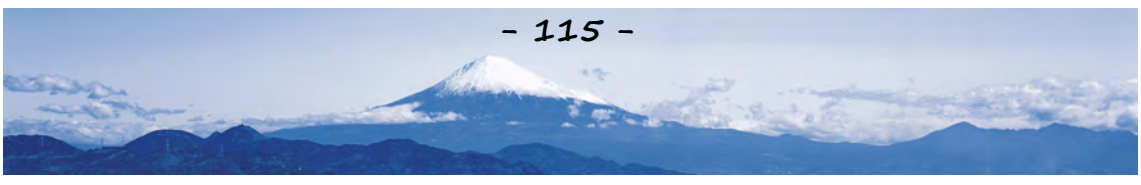
[1]Karlsruhe Institute of Technology (KIT), [2]SCK CEN, [3]MTA Ungarian Academy of Sciences, Centre for Energy Research, [4]Forschungszentrum Julich, [5]EUROfusion, [6]ORNL

One of the key objectives of the EUROfusion Materials Work Package is to conduct neutron irradiation studies to generate reliable material data for the conceptual design of a DEMO reactor. These experiments replicate the anticipated radiation-induced damage effects, focusing on material characterization under breeding blanket and divertor conditions. Neutron irradiation at lower doses is performed in the BRR (Hungary) and BR2 reactor (Belgium), while higher-dose experiments are conducted in HFIR (USA) and BOR60 (Russia) through subcontracting.

This work provides an overview of the European irradiation program, including recent findings and future perspectives. Screening studies on W and Cu materials are reviewed, along with initial results for EUROFER97-2 steel in a condition simulating technological production. Additionally, a project investigating the full range of construction-relevant properties (strength, creep, fatigue, fracture toughness, impact toughness, swelling, and hardness) of EUROFER97-3 steel is presented. The samples are irradiated at 325 °C and 550 °C, representing the lower and upper blanket operating temperatures. Initial findings indicate no softening at 550 °C after 20 dpa, while at 325 °C, the toughness loss in the maximum-strength condition is critical. Furthermore, the in-situ irradiation creep of EUROFER97-3 has been quantified.

The results of this unique project will enhance understanding of EUROFER97 steel's radiation resistance and contribute significantly to the design and licensing of future fusion power plants.

Invited



29PL2

Development of Fusion-Grade Tungsten Materials at Industrial Scale

Shuhei Nogami[1], Naoya Matsuta[1], Kei Miura[1], Shigekazu Yamazaki[1], Tomohiro Takida[1], Akira Hasegawa[2], Carsten Bonnekoh[3], Michael Rieth[3], Marius Wirtz[4], Gerald Pintsuk[4], Dmitry Terentyev[5]

[1]A.L.M.T. Corp., Japan, [2]Tohoku University, Japan, [3]Karlsruhe Institute of Technology, Germany, [4]Forschungszentrum Juelich, Germany, [5]SCK−CEN, Belgium

Since longer time operation with neutron irradiation is expected in fusion reactors after ITER, tungsten materials with microstructural stability, improved mechanical properties, resistance to high heat flux (HHF), and tolerance against irradiation embrittlement have been developed making full use of the microstructure control technologies. In this presentation, the development status of “fusion-grade” tungsten materials by A.L.M.T. Corp. at industrial scale and the results of durability verification under the expected environment of fusion reactors will be shown.

The second phase dispersion and solid solution alloying accompanied with the control of grain and grain boundary features were the major method applied to the development. Especially, the effect of doping by potassium bubbles (K-doping) and lanthanum oxide particles and the alloying by rhenium and tantalum were evaluated. In the non-irradiated state, each modification method was effective in improving recrystallization resistance, ductile-to-brittle transition temperature (DBTT), strength and ductility, and resistance to HHF. Further improvements were observed when doping and alloying were applied simultaneously. However, even if the chemical composition is the same, the grain structure could change depending on the manufacturing conditions, such as the rolling ratio and heat treatment conditions. Hence, the properties also changed. Therefore, based on the systematic evaluation by small-scale materials, the manufacturing conditions were optimized for mass production, and the prospects for manufacturing materials with properties equivalent to those of small-scale materials were obtained. The presentation will show details of the material development and durability verification, including the effects of the aforementioned modification methods on the irradiation tolerance.

Tentative

29PL3

Developing predictive materials models for the virtual fusion reactor from atomistic simulations - proof of concept application to irradiation creep

Max Boleininger[1], Alexander Feichtmayer[2], Daniel R Mason[1], Johann Riesch[2], Luca Reali[1], Sergei L Dudarev[1]

[1]UK Atomic Energy Authority, [2]Max Planck Institute for Plasma Physics

Invitation

The realisation of economically viable fusion reactors requires developing a “virtual reactor” representation of an operating tokamak device [1], informed by predictive materials models spanning the parameter space of simultaneous thermal, mechanical, and radiation loads. As we presently lack materials data under fusion conditions, we must develop predictive models from first principles.

Here, we present a combined in silico and in vivo study focused on quantifying deformation in tungsten under stress and irradiation at low temperature [2]; conditions found in actively cooled components. While it is understood that irradiation accelerates creep deformation in metals, the phenomenon of irradiation creep becoming more pronounced at low temperature defies conventional understanding. To address this apparent contradiction, we developed of a virtual representation, or “digital shadow”, of a custom-designed experiment for measuring irradiation creep in a tensioned tungsten wire. We show how atomistic simulations can inform a multiscale model capable of quantitatively predicting deformations in a material under irradiation - ahead of the actual experiment, without any adjustable parameters.

We discover that, under irradiation at room temperature, internal stresses of up to 2 GPa relax within 0.5 dpa. As opposed to conventional notions of radiation creep, the effect arises from the self-organisation of nanoscale defects, athermally coalescing into polarised dislocation networks that compensate the external stress. Actively cooled materials, in which stresses critical to structural integrity are expected to form, would relax through the favourable creep mechanism explored in this study.

[1] Ellis et al. (2025) Nuclear Fusion (accepted)

[2] Feichtmayer et al. (2024) Communications Materials 5.1: 218

29A1

Neutron-irradiation effects on the mechanical and electrical properties of bimodally precipitate strengthened Cu alloys

Alice E. Perrin[1], Dai Hamaguchi[2], Josina W. Geringer[1], Steve Zinkle[3], Ying Yang[1], Steve Skutnik[1], Jon Poplawsky[1], Yutai Katoh[1]

[1]Oak Ridge National Laboratory, [2]National Institutes for Quantum Science and Technology, [3]University of Tennessee, Knoxville

High strength, high conductivity copper alloys that can resist creep at high temperatures are candidate materials for efficient heat exchangers in fusion reactors. Cu-Cr-Nb-Zr (CCNZ) alloys were designed for improved strength and creep life compared to ITER Cu-Cr-Zr (CCZ) reference alloys by introducing a bimodal distribution of strengthening phases. These CCNZ alloys have comparable electrical conductivity and tensile properties to CCZ alloys while demonstrating a substantially longer creep rupture life at 500 °C. However, the effects of neutron irradiation on the properties of these alloys were unknown. This study characterizes and compares the room temperature mechanical and electrical properties of neutron-irradiated CCNZ and ITER reference heat sink CCZ alloys. Tensile specimens were neutron irradiated in the High Flux Isotope Reactor (HFIR) to 5 dpa between 250 °C and 325 °C. Post-irradiation characterization included electrical resistivity measurements, hardness, and tensile tests. Microstructural evaluation used scanning electron microscopy, energy dispersive X-ray spectroscopy, and atom probe tomography to characterize the irradiation-produced changes in the microstructure and investigate the mechanistic processes leading to post-irradiation properties. Transmutation calculations were validated with compositions measured from atom probe data and used to calculate contributions to the increased electrical resistivity measured after irradiation. Although CCNZ alloys showed more irradiation hardening than CCZ, the overall tensile behavior deviated very little from nonirradiated values in the temperature range studied, and the two alloys had comparable post irradiation resistivity.

Tentative

29A2

Advanced In-situ SEM/EBSD Analysis of CuCrZr alloys: Impact of Neutron Irradiation on Deformation Mechanisms

Dai Hamaguchi[1], Maxim Gussev[2], Soyoung Kang[2], Xiang (Frank) Chen[2], Takashi Nozawa[1], Yutai Kato[2]

[1]National Institutes for Quantum Science and Technology, [2]Oak Ridge National Laboratory

CuCrZr is the material of choice as heat sink for the ITER divertor due to their good balance between thermal conductivity and mechanical properties and is also a candidate for DEMO divertor heat sink material. However, deformation mechanism of the alloy is not yet well understood. In addition, unlike the ITER, irradiation damage is an issue for the use in DEMO and the results from past neutron irradiation tests have shown embrittlement especially at low temperatures, which is one of the concerns in setting the coolant temperature of DEMO divertor. Therefore, understanding the mechanism of embrittlement is important in design and for remedial measures. The aim of this study is to elucidate deformation mechanism of CuCrZr and effects of neutron irradiation by evaluating tensile deformation behaviour using in-situ SEM/EBSD analysis and digital image correlation (DIC) techniques. In-situ SEM/EBSD tensile tests of non-irradiated CuCrZr have shown that the deformation is not accompanied by deformation twins as a usual case for pure copper and other copper alloys. The presence of precipitates leads to a complex slip line pattern with dislocation pile ups at the precipitations. The characterization on irradiated materials is also underway to understand the effect of neutron irradiation on the mechanism. In this presentation, the deformation mechanism of CuCrZr alloy and the effect of neutron irradiation will be comprehensively discussed with the materials irradiated in High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory to the neutron doses of 5dpa expected in DEMO applications.

Tentative

29A3

Development of low-energy mechanical alloying technology to enable mass production of ODS-Cu alloys for fusion divertors

Hao Yu[1], Zimo Gao[1], Toshiki Saito[1], Minha Park[1], Yasuyuki Ogino[1], Sosuke Kondo[1], Ryuta Kasada[1]

[1]Tohoku University

Oxide dispersion strengthened Cu (ODS-Cu) alloys fabricated using the mechanical alloying (MA) process have high thermal conductivity, excellent strength, and resistance to neutron irradiation, promising to be heat-sinking materials for the divertors in fusion demonstration power stations (DEMO). The high ductility of Cu powders makes it easy to agglomerate during the MA process, which greatly limits the mass-production of ODS-Cu alloy powders. Considering their future applications in fusion reactors, new MA processes suitable for ODS-Cu need to be developed urgently. This study aims to create a novel low-energy MA approach for the mass-production of ODS-Cu alloys.

Cu-ODS alloys with a nominal composition of Cu-0.85Hf (wt.%) were fabricated using a water-cooled ball-milling machine (Emax, Retsch, Germany) for various MA durations (1, 3, and 12 h) and RPMs. The MAed powders were consolidated by a spark plasma sintering (SPS) at 900 °C with a load of 50 MPa for 15 min. Microstructural observations focused on the dispersed oxide particles in ODS-Cu alloys were performed using STEM. It was found that the recovery of Cu powders prepared by the low-energy MA process exceeded 90%, and the Hf-rich fine oxide particles were uniformly dispersed in the Cu matrix. Besides, the newly developed Hf-added ODS-Cu alloys show comparable tensile strength and better ductility than commercial GridCopR Cu alloys. The microstructure and mechanical property characteristics of ODS-Cu alloys fabricated by the new MA method will be presented, and some new concepts that enable mass production of ODS-Cu alloys will be summarized.

Invitation

29A4

A multi-scale microstructural characterization of additively manufactured ITER-Grade CuCrZr with superior properties

Francisco Canillas[1], Nerea Ordas[2], Marcelo Roldan[1], Wolfgang Pantleon[3], Esmeralda Munoz-Cortes[4], Rebeca Hernandez[4], Edgar Leon-Gutierrez[1]

[1]CIEMAT - National Fusion Laboratory, [2]Ceit-BRTA, [3]Technical University of Denmark, [4]CIEMAT - Division of Materials of Energy Interest

The ITER-Grade (IG) CuCrZr alloy is a key material for High Heat Flux components in ITER, EU-DEMO, and future fusion reactors due to its exceptional combination of high thermal conductivity, mechanical strength and thermal stability. The feasibility of producing dense IG-CuCrZr via Powder Bed Fusion Electron Beam Melting (PBF-EB/M) has been successfully demonstrated, achieving thermal and mechanical properties exceeding ITER specifications. The unique advantages of this additive manufacturing technique not only enable the fabrication of complex geometries while simultaneously promoting a refined hierarchical microstructure with distinct length-scale features, contributing to the material's exceptional performance.

To further understand the underlying mechanisms responsible for the outstanding thermal and mechanical properties and to establish a clear relationship between them and the processing parameters, a detailed microstructural characterization was conducted. Transmission Electron Microscopy (TEM) provided insight into the distribution and stability of nanometric Cr-rich precipitates, offering a basis for assessing their response to thermal exposure and potential evolution at elevated temperatures. Additionally, Electron Backscatter Diffraction (EBSD) revealed the impact of texture evolution, which was correlated with the mechanical anisotropy through computing the Taylor factor.

These findings not only enhance the fundamental understanding of the microstructural mechanisms governing strength and ductility in additively manufactured IG-CuCrZr but also highlights the critical role of microstructure control in achieving superior mechanical performance. This insight establishes a framework for the tailored design of materials with specific anisotropic mechanical properties suited for targeted applications.

Tentative

29A5

Application of ITER-like W/Cu PFCs for high heat flux components in EAST

Da huan Zhu[1], Yang Wang[2], Chun yu He[2], Chuan Xuan[2], Chang jun Li[1], Rui Ding[1], Bin fu Gao[1], Bao guo Wang[1], Jun ling Chen[1]

[1]Institute of Plasma Physics, Chinese Academy of Sciences, [2]Institute of Plasma Physics, Chinese Academy of Sciences / Science Island Branch of Graduate School, University of Science and Technology of China

Invited

In the future fusion devices, the high heat flux on divertor targets can reach up to 20MWm-2, posing high requirements of plasma facing components (PFCs). Thus, several high performance PFCs, i.e W/Cu monoblocks and W/Cu flat-type components have been developed by the ASIPP. And, they have already passed through the harsh high heat flux tests. To identify their performance under actual tokamak plasma conditions, they are tested in the EAST.

The EAST was gradually installed actively cooled ITER-like W/Cu PFCs for its divertors and limiters in the last decade. With the application of high performance W/Cu PFCs for high heat load components, the temperature of PFCs could reach steady state and thus the plasma pulse was gradually prolonged, greatly promoting the development of physics experiments. In spite of good results, some problems have also been exposed. After a long term plasma exposure, many types of damage phenomena, i.e. both melting and cracking at leading edges, the cracking of joint interface and even exfoliation of W plates, the net-like cracking on surface, the melting of cooling tube and thus leakage failure, the recrystallization even abnormal W grain growth and so on, have been observed and inspected on W/Cu PFCs. The morphological and distribution features of these damages are well described. And, their possible generation mechanisms are deeply analyzed and discussed.

The application of ITER-like W/Cu PFCs in the EAST provides a demonstration, especially, their damages are important reference for the future ITER and other fusion devices.

29A6

Tungsten fibre-reinforced tungsten composites - materials qualification at the high heat flux test facility GLADIS

Johann Riesch[1], Yiran Mao[2], Bernd Boswirth[1], Daniel Dickes[1], Katja Hunger[1], Alexander Lau[2], Hans Maier[1], Henri Greuner[1], Jan W Coenen[3], Rudolf Neu[4]

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Tungsten fibre-reinforced tungsten composites (Wf/W) utilize extrinsic mechanisms to enhance fracture toughness and are therefore considered a promising advanced material for plasma-facing components. In this contribution, we present the qualification of this material using high heat flux testing in the neutral beam facility GLADIS.

In the early development stage, when material availability was still limited, small blocks were used in adiabatic loading conditions for an initial qualification. In a second step, actively cooled mock-ups have been equipped with these blocks to investigate the thermo-mechanical behavior of the composites in a fusion-relevant heat flux and temperature range. With the improvement of manufacturing technologies testing of flat tile mock-ups featuring a size similar to the plasma facing surface of the DEMO mono-blocks concepts was possible. In contrast to the first tests, the aim was now to evaluate the macroscopic behavior and thus the feasibility of the material concept in general. Finally, mono-blocks featuring the European DEMO design were tested in two steps: screening to a dedicated load and cyclic loading at the foreseen steady state power density. Successful loading with 100 cycles, 10 s at 15 MW/m^2 confirmed both the maturity of the fabrication technology and heat removal capacity of the composite material.

For all tests, details on the heat flux loading, pre- and post-characterisation and their role in the material development approach are presented. In summary this study on Wf/W provides a show case for the use of high heat flux testing for the qualification of advanced materials.

Invitation

29B1

Computational materials science evaluation of the D/Li-stripping neutron source as a prototypical fusion materials testing facility

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This presentation will highlight activities in the United States over the past 18 months to assess the suitability of the deuteron/Li-stripping neutron source concept as a prototypical fusion materials testing facility. The analysis performed involves the participation of experts in the areas of neutronics, transmutation and chemistry changes, computational thermodynamics, primary damage production, and long-term microstructural evolution. The studies have been conducted for materials catalogued as primary candidates for structural applications in fusion reactors: reduced-activation ferritic-martensitic steels, vanadium alloys, tungsten alloys, and silicon carbide. The neutron spectra evaluated for prototypicality include several DEMO concepts, the ITER spectrum, the IFMIF-DONES concept, and a pulsed irradiation spectrum representative of inertial fusion energy concepts. We have defined quantitative materials performance metrics to evaluate the proximity of the D/Li-stripping neutron source to a true fusion neutron spectrum in terms of He and H to displacement damage ratios, cascade damage production, transmutation inventory, and the formation of transmutation-induced thermodynamic phases. On the basis of our extensive computational materials science evaluation, our main conclusion is that the D/Li-stripping neutron spectrum is largely consistent with reference fusion concept spectra across all material systems considered.

Tentative

29B2

Helium-Hydrogen Synergistic Effects in Structural Materials in Fusion Reactors

Chenxu Wang[1], Jianming Xue[1], Yugang Wang[1], Shijun Zhao[2], Fei Gao[3], Hong-bo Zhou[4], Xiangyan Li[5], Changsong Liu[5], Steven J. Zinkle[6]

[1]State Key Laboratory of Nuclear Physics and Technology, Center for Applied Physics and Technology, Peking University, Beijing, China, [2]Department of Mechanical Engineering, City University of Hong Kong, Hong Kong, China, [3]Department of Nuclear Engineering and Radiological Science, University of Michigan, Ann Arbor, USA, [4]Department of Physics, Beihang University, Beijing, China, [5]Hefei Institutes of Physical Science, Chinese Academy of Sciences, Hefei, China, [6]Department of Nuclear Engineering, University of Tennessee, Knoxville, USA

Invited

In fusion reactors, 14 MeV neutron irradiation will produce large amounts of helium and hydrogen simultaneously with displacement defects in structural materials. These He and H atoms interact with displacement defects, leading to He-H synergistic effects and aggravating the irradiation damage. However, there exist no available high-flux fusion neutron sources for material performance evaluation. To date, multiscale simulation and multi-ion beam irradiation have been widely used to study the radiation effects in structural materials.

A hybrid-style Fe-H-He ternary potential was successfully developed. Based on this potential, the atomic-scale fine structure and the evolution mechanism of H-He-V cavities in BCC-Fe were investigated using molecular dynamics (MD) simulations. Comprehensive fundamental parameters for the interactions among He, H, and cavities were developed and employed for further mesoscale simulations. With these parameters, we performed kinetic Monte Carlo (OKMC) simulations on cavity evolution in RAFM-like steels under neutron and ion irradiations, focusing on the effects of irradiation particle type, temperature, and dose rate. Cluster dynamics (CD) simulations were also performed to investigate the influences of He and H on the long-term evolution of cavities under irradiation at high doses.

On the other hand, to comprehensively study the synergistic effects and confirm the mechanism obtained from simulation results, a triple-beam irradiation facility has been established at Peking University. In this talk, we will discuss the main findings regarding the irradiation dose, dose rate, temperature, and H/He concentration dependence of cavity swelling based on our experimental results and those obtained from other multi-ion beam irradiation facilities.

29B3

Understanding the Role of Hydrogen on Cavity Evolution in Triple Ion Irradiated RAFM Steels

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The mechanisms behind the synergistic influence of H, He and radiation damage on the cavity microstructures of reduced activation ferritic/martensitic (RAFM) steels were investigated using ion irradiation and advanced characterization techniques. The RAFM steel F82H-IEA was subjected to dual (Fe^{2+} + He^{2+}) and triple (Fe^{2+} + H^{+} + He^{2+}) ion irradiation across a wide spectrum of damage levels (0.5 to 150 dpa), H:He ratios (0:10 to 80:10 appm/dpa) and irradiation temperatures (400 °C to 600 °C) in single parameter variations about a nominal case of 50 dpa, 1x10^{-3} dpa/s, 40:10 appm/dpa H:He and 500 °C. Post-irradiation microstructural characterization was conducted via transmission electron microscopy to evaluate the effects of H co-injection on cavity microstructure and swelling. A 40:10 appm/dpa H:He ratio was shown to enhance bubble nucleation, void growth, and swelling on the order of 1.6-2.4x compared to the dual ion condition. Conversely, 80:10 appm/dpa H:He decreased void growth and swelling by 0.5x, below even the dual ion condition due to over-nucleation of bubbles by a factor 1.2-2.3x. Electron energy loss spectroscopy (EELS) was then utilized to observe a core-shell structure of helium and hydrogen at small bubbles (5 nm) and larger voids (5 nm) across a wide range of temperatures. This observation is attributable to the interaction of hydrogen with the helium-filled cavities and subsequent trapping. Combining the EELS results and the cavity microstructure into a rate theory model, the mechanism by which hydrogen modifies the cavity microstructure and swelling was determined.

Tentative

29B4

Multiscale studies on the synergistic effects of helium and hydrogen on cavity evolution in ferritic steels in the fusion environment

Fengping Luo[1], Denghuang Chen[1], Haoxuan Huang[2], Bowen Zhang[1], Yuxin Liu[1], Jin Wang[1], Hong-Bo Zhou[2], Guang-Hong Lu[2], Fei Gao[3], Chenxu Wang[1], Yugang Wang[1]

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Transmutation atoms He and H produced by fusion neutrons can synergistically alter cavity evolution, while the underlying physical mechanisms remain unclear. We perform systematical multiscale studies to reveal the microscopic and mesoscopic mechanisms for synergistic effects of He and H on cavity evolution in ferritic steels. First, the trapping of He and H by a nanocavity and the He-H interaction around the nanocavity in BCC iron are investigated using ab-initio based lattice kinetic Monte Carlo simulations, establishing a fundamental kinetic picture for the ternary He-H-nanocavities interactions. Furthermore, the microscopic mechanisms for He and H to influence cavity evolution in BCC iron are examined using atomic simulations. He and H stabilize nanocavities, reduce the nanocavity bias, and hinder the diffusion of vacancy clusters. Predictive models for the energetics of He-H-nanocavities ternary interactions and the capture radius of the ternary He-H-nanocavities for displacement defects, and a database for diffusion coefficients of He-H-vacancy clusters are established. This provides energetic and kinetic basis for mesoscale simulations on cavity evolution. By informing these microscopic mechanisms into cluster dynamics, the mesoscopic mechanisms for He and H to influence the long-term cavity evolution in ferritic steels are revealed. He and H enhance the bubble-to-void transformation, thus shortening incubation period for swelling and enhancing cavity nucleation. These microscopic mechanisms exhibit strong coupling behaviors. Synergistic effects of He-H co-existence on cavity evolution are dominated by He. These results facilitate a deep understanding of irradiation effects of fusion neutrons.

Tentative

29B5

Unraveling damage accumulation due to pre-existing irradiation defects interaction with newly generated vacancies upon low-energy H plasma exposure

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Invitive

Understanding damage accumulation in metals under irradiation asks to uncover the influence of pre-existing damage upon the production of new defects and the involved temporal/spatial microstructure evolution. Herein, poly-crystalline tungsten (W) were irradiated with 1 MeV self-ions to introduce defects with various damage levels and subsequently exposed to hydrogen (H) plasma with 415 eV ion energy to a fluence of 5×10^{22} m^{-2}, where no collision cascade is expected but nm-thin hydrogen supersaturated surface layers (HSSL) comprising high density of vacancy-type defects can be generated [1, 2]. Our focus is the effects of pre-existing self-ion damage on vacancy production/evolution upon the non-cascade HSSL formation. Dedicated transmission electron microscopy and H profile measurements with high depth resolution using ^{15}N nuclear reaction analysis were applied to tackle the defect microstructures evolution and to quantify the vacancy-type defects produced.

Preliminary results show a strong influence of pre-existing defects on the early-stage HSSL microstructure. The number of newly generated vacancies during HSSL formation adds to those pre-existing from self-ion damage, leading to a trapped H content of ~15 at.% in W. This record-high H concentration is one order of magnitude larger than that in W undergoing MeV self-ion irradiation followed by H decoration at a few eV ion energy [3] and indicates that vacancy production during the non-cascade HSSL formation is much more effective. Our finding is crucial to resolve the complex damage accumulation process in W PFMs under irradiation. Extrapolation to future fusion reactor conditions is also discussed.

29B6

Molecular dynamics study on hydrogen-helium induced hardening effect in RAFM steels

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Reduced activation ferritic/martensitic (RAFM) steels are the most promising candidate structural material for future fusion reactors because of their excellent irradiation resistance. The helium and hydrogen produced simultaneously by high energy fusion neutrons significantly influence microstructure evolution and degrade the mechanical properties of RAFM steels. Normally, the hydrogen content in the samples is relatively low, but the hydrogen effect cannot be ignored. In order to study the synergistic effect of helium and hydrogen, molecular dynamics (MD) simulations were used to study the evolution of hydrogen-helium-vacancy (H-He-V) clusters in αiron with an intrinsic dislocation line under the effect of cascade damage. The results indicate that the presence of hydrogen promotes the nucleation of He-V clusters, while inhibiting the growth of helium bubbles. The evolved model was subsequently used as the initial configuration for hardening model. The presence of a hydrogen atmosphere significantly increases the shear stress required for dislocation unpinning. To evaluate the helium content inside bubbles in samples that under the irradiation of the Swiss spallation neutron source (SINQ), the barrier strength of helium bubbles and hydrogen-helium bubbles with varying hydrogen- and helium vacancy ratios at various sizes were calculated based on the dispersed barrier hardening model. By comparing the barrier strength obtained from simulations and experiments, the helium content in the bubbles was derived. These results help to understand the H-He synergistic effect in RAFM steels performed in future fusion reactor.

Tentative

29C1

Simultaneous Neutron Irradiation and Hydrogen exposure on Tungsten: Hydrogen Retention

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Fusion materials are subjected to extreme and complex environments in fusion reactors, where tungsten, as a primary plasma-facing material, must endure simultaneous neutron irradiation and hydrogen isotope exposure. This study introduces a novel methodology that simultaneously exposes tungsten to neutron irradiation and a hydrogen environment using the High Flux Isotope Reactor at Oak Ridge National Laboratory. Radiation generated defects that could interact with hydrogen, and this novel capability enables investigation of such dynamic interactions between irradiation-induced defects and hydrogen. The hydrogen atmosphere was established through hydrogen release from vanadium hydride in the first-of-its-kind irradiation capsule. Tungsten was irradiated to 0.03 dpa at ~300-390 ° C. The hydrogen pressure during irradiation was estimated as 1900 Pascal through the post-irradiation analysis on vanadium hydride samples.

Thermal desorption spectroscopy was conducted on tungsten samples subjected to simultaneous neutron irradiation and hydrogen exposure to evaluate hydrogen retention under different hydrogen pressures. For comparison, unirradiated tungsten samples exposed only to a hydrogen environment were also analyzed. The results revealed significantly higher hydrogen retention in tungsten subjected to simultaneous irradiation and hydrogen exposure, with an additional desorption peak attributed to trapping at irradiation-induced defects. Furthermore, post-irradiation deuterium charging was performed on the hydrogen-containing irradiated tungsten samples to investigate post-irradiation deuterium trapping behaviors and potential hydrogen-deuterium exchange mechanisms. This study successfully demonstrates the first-of-a-kind capability to simultaneously expose fusion materials to neutron irradiation and a hydrogen environment, and demonstrate the hydrogen trapping behaviors under simultaneous irradiation and hydrogen environment.

Invitation

29C2

Low Hydrogen Isotope Retention and High Irradiation Resistance in Columnar-Grained Tungsten Prepared via Chemical Vapor Deposition for Fusion Applications

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Chemical vapor deposited tungsten (CVD-W) is a promising plasma-facing material (PFM) for fusion devices, offering high density, ultra-high purity (99.99997%), and ease of fabrication. In the current work, millimeter-thick CVD-W with columnar grains was used, exhibiting a strong [001] grain orientation and radial grain sizes ranging from 15 to 50 μm. The results highlight the exceptional low hydrogen isotope retention and high irradiation resistance of CVD-W under various conditions. Low-energy deuterium plasma exposures (~500 K, up to 2.2E27 D m^{-2}) show that CVD-W retains only 12-30% of the deuterium compared to ITER-like W, mainly due to its ability to suppress deuterium-induced blistering, which reduces dislocation formation and, in turn, decreases deuterium capture. Additionally, its vertical grain boundaries, which enhance outward deuterium transport, were confirmed by permeation experiments. Even under combined high-energy ion and low-energy plasma irradiation, CVD-W still retains less deuterium (38% and 82% of ITER-like W), demonstrating superior hydrogen retention. TEM and nanoindentation tests show that its resistance to displacement damage is comparable to ITER-like W. CVD-W exhibits excellent structural thermal stability, with no significant grain growth or loss of columnar structure after annealing at 2173 K for 3 hours. Under simulated ELM-like thermal shock (1 ms, 100 pulses), CVD-W shows a higher cracking threshold (0.33-0.44 GW m^{-2}) than ITER-like W (0.19-0.33 GW m^{-2}), indicating superior resistance to thermal shock cracking. In conclusion, CVD-W outperforms ITER-like W in hydrogen isotope retention, blistering resistance, high-temperature stability, irradiation damage resistance, and thermal shock resistance.

Tentative

29C3

Investigation of deuterium trapping by helium bubbles in tungsten

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Irradiation of tungsten (W) plasma-facing components with 14 MeV neutrons will result in the creation of displacement damage and the production of H and He. The He atoms will agglomerate into bubbles, which size will depend on the amount of He and temperature. These He bubbles will act as traps for the H isotopes. However, the exact trapping mechanism is still under discussion.

W single crystals were implanted with 1 MeV He ions at 290 K. One set of samples was implanted to a peak He concentration of 1 at.% and then annealed at different temperatures (1400, 1700, 2000 K). The same was performed with the samples that were additionally irradiated with 20 MeV W-ions to 0.5 dpa at 290 K. Another set of samples was implanted to various peak He concentrations (1, 2, 5 at.%) and then annealed at 1700 K. The He bubble sizes and densities in selected samples were characterized using transmission electron microscopy. The samples were exposed to a deuterium (D) plasma at 370 K. D concentration profiles in the samples were measured using D(^{3}He, p)^{4}He nuclear reaction analysis.

The trapped D concentration was proportional to the He concentration. Concurrently, it was weakly dependent on the annealing temperature. Pre-irradiation with 20 MeV W ions resulted in almost two-fold increase of the trapped D concentration, even though the displacement damage is usually annealed at 2000 K. Thermal desorption spectra from He-implanted samples have a different structure compared with 20 MeV W-irradiated samples annealed at the same temperature.

Invited

29C4

Effect of He irradiation on hydrogen isotope retention in W-Re

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Tungsten (W) is a promising candidate for plasma facing materials in fusion reactors. It will be exposed to high fluxes of deuterium (D), tritium (T), helium (He) ash and neutrons. In addition, W will be irradiated by 14 MeV neutrons, leading to the transmutation to rhenium (Re) in W matrix. For steady-state operation in a fusion reactor, it is crucial to understand the dynamics of hydrogen isotopes, including desorption and dissolution. Furthermore, He irradiation is expected to introduce defects and generate He bubbles, both of which can affect the hydrogen isotope dynamics. In this study, we investigate the effect of He irradiation to hydrogen isotope desorption in W-Re.

The W-10%Re alloy disks with 6 mm diameter and 0.5 mm thickness were used. The samples were irradiated with 6 MeV Fe^{2+} to introduce irradiation defects with damage levels of 0.1 and 1.0 dpa at TIARA facility in QST. Subsequently, D + He mixed plasma irradiation was performed at the Deuterium Plasma Exposure at Sandia National Laboratory. The hydrogen isotope desorption behavior was evaluated by high temperature thermal desorption spectroscopy. The results show that D retention increases with accumulated damage, whereas He desorption exhibits no significant change between the samples with 0.1 dpa and 1.0 dpa. This suggests that irradiation-induced defects have a limited impact on He retention for W-10%Re. In this presentation, we will discuss the D and He retention behavior for damaged W-10%Re under various D + He exposure conditions.

Invited

29C5

Oxidation of tungsten and its implications to tritium retention in decommissioning, maintenance and accident scenarios in future devices

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The oxidation of tungsten can result in significant loss of first wall materials and dispersion of tungsten oxide in form of sublimated gas molecules or dust spalled off from the oxide scale. Recent modelling work has shown that one of the oxide phase (WO_2) could acts as a tritium barrier, which will impact the tritium retention of tungsten-based components. Therefore, it is important to understand the microstructure of the tungsten oxide scale (oxide phase formed and morphology). In this presentation, we will report on the results of tungsten oxidation experimental campaigns performed using a thermogravimetric analysis in a range of experimental conditions: temperature ranging between 450 °C and 1200 °C, varying relative humidity (0%, 30%, 60%) or various oxygen content (<5 ppm (N_2 $N_{4.8}$), 0.01%, 0.1%, 1% and 20%). Raman spectroscopy shows that the oxide scale is formed of three layers: (1) a thin 30-50 nm thin WO_2 layer growing on metallic tungsten, followed by (2) a 5-10 µm thin $WO_{2.72}$ and (3) the rest of the oxide scale is formed of the WO_3 phase. The formation of cracks and voids have been observed in the WO_3 layer, leading to dust formation. Current work focuses on the impact of oxygen impurities in vacuum or inert gas to assess the potential impact of tungsten oxidation on plasma operation. Early results indicates that the WO_2 phase is preferentially formed and measurement of hydrogen-isotope permeation through the oxide is in progress to assess its potential impact on tritium permeation and retention in first wall component.

Tentative

29C6

Impact on He seeding for D and He retention in W alloys by DPE D+He plasma exposure

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Understanding hydrogen isotope dynamics in plasma facing walls is a top R&D priority for tritium fuel management. In the Japan-US collaboration programs PHENIX and FRONTIER, it was shown that neutron damage changes the hydrogen isotope behavior in tungsten (W), including the creation of stable trapping sites. However, DT fusion also creates He ash, which nucleates nm-sized defects in W. Evaluating He seeding effects on hydrogen isotope dynamics is essential.

In this study, D + 3-20% He plasma exposure for W-Ta, W-Mo and W- La2O3 was performed by DPE (Deuterium Plasma Experiment) device at SNL (Sandia National Laboratories) at 250 °C. The flux of D+He was ~1.0 X 10^{20} m^2 s^{-1}, allowing a total fluence of 1.0 × 10^{25} m^2 to be reached in 3 hrs.

Thereafter, HT-TDS (High Temperature Thermal Desorption Spectroscopy) was applied to observe D_2 and He desorption separately using a high-resolution QMS.

A D_2 desorption peak from undamaged W was found at 800 K. With 3% He seeding, a large reduction of D retention was observed. For the advanced plasma facing materials, we observed two major desorption stages at 600 K and 800 K. The largest D retention was found for the W- La_2O_3 specimens, which may be influenced by the existence of oxides. An increase in He retention was directly correlated with a reduction in D retention among these specimens. This indicates that He occupies the same stable trapping sites as D and the trapped He works as a near-surface D diffusion barrier.

Tentative

30PL1

Mechanical Properties and High Temperature Helium Embrittlement Mechanisms in Irradiated Alloys

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Recent fundamental radiation effects studies on prospective fusion materials are summarized. Scanning transmission electron microscopy and atom probe tomography correlative spectroscopy techniques have been employed to investigate <100> and ½<111> dislocation loop evolution and Cr segregation at dislocation loops in ion irradiated Fe-Cr alloys. The potential role of preferential loss of glissile ½<111> clusters to foil surfaces on the observed loop evolution vs. temperature will be discussed. The phase stability of binary Fe-Cr alloys has been examined using ion irradiations with different primary knock-on atom (PKA) energies, dose rates and temperatures. Implanted ions, low temperatures, high PKA energies, and high dose rates are demonstrated to suppress alpha prime precipitate formation. Significant heterogeneous Cr-enriched clustering can occur during irradiation at low temperatures (<300C) even for low-Cr alloys that are nominally in the single phase portion of the Fe-Cr phase diagram. The importance of including both the unirradiated and irradiation-induced microstructural components in appropriate hardening superposition models will be shown to be essential for obtaining reliable microstructural predictions of radiation hardening. The effect of applied stress, He implantation rate, and temperature on grain-boundary cavitation has been quantified for multiple structural alloys using low dose rate He implantations at high temperatures with stress applied during the implantation. The key role of plastic deformation (dislocations) on stimulating high temperature helium embrittlement will be emphasized.

Tentative

30PL2

Hydrogen isotope retention and removal in plasma-facing materials

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Deuterium (D) and tritium (T) are the primary fuels for future fusion reactors. However, T tends to become trapped in plasma-facing materials, leading to radioactivity contamination and safety concerns. Moreover, high-flux plasma and high-energy neutron irradiations generate defects that act as deep T traps, thereby increasing T inventory. While high-temperature baking can promote the release of trapped T, its potential to damage critical components limits its practical application. This study proposes hydrogen isotope exchange as an efficient, low-temperature alternative for T removal. To simulate T retention under fusion-relevant conditions, tungsten samples were pre-irradiated with D plasma. We systematically investigated three removal methods: vacuum baking, hydrogen (H) gas baking, and H plasma exposure. Among them, H gas baking proved the most effective due to isotope exchange effects, achieving 86% D removal at 240 °C and 95% at 320 °C, with efficiency saturating after 8 hours. Additionally, H glow discharge demonstrated six times higher D removal efficiency compared to helium glow discharge. Notably, this isotope exchange method reduced the required baking temperature by ~200 °C compared to conventional vacuum baking and more effectively reduced T concentrations at greater depths, as H-T exchange proved more efficient than D-T exchange. These findings highlight hydrogen isotope exchange as a promising strategy for low-temperature T removal, enhancing material integrity and operational safety in fusion reactors while addressing key engineering challenges in T management. In this presentation, the underlying mechanisms will be further explored, offering insights into its practical implementation and potential for integration into future fusion reactor designs.

Tentative

30PL3

Role of grain boundaries on irradiation damage evolution in a reduced activation ferritic/martensitic steel F82H

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In order to investigate the sink strength of grain boundaries for point defects in a reduced activation ferritic/martensitic steel: F82H electron- and ion-irradiation experiments were carried out using the multi-beam high voltage electron microscope at Hokkaido University and the high-fluence irradiation facility at the University of Tokyo, respectively. F82H typically contains various types of grain boundaries (GBs), such as prior-austenite GBs, packet GBs, block GBs, and lath boundaries.

The irradiation experiments resulted in the formation of SIA clusters, dislocation loops, and cavities, which exhibited a gradient in their sizes and number densities. Furthermore, defect-free zones (DFZs) were observed around GBs in some cases, and it seemed that the width of DFZs depended on the character of the GBs. The atomic density ratio (ADR) of GB planes is a key parameter in determining the width of DFZ; a GB with a higher ADR typically exhibits a wider DFZ compared to one with a lower ratio. Additionally, in the case of irradiation with He and H, the cavity formation was strongly affected by the concentration of He and H, resulting in reduced DFZ formation even around a GB with a higher ADR.

These results clearly indicate that the distribution of irradiation-induced secondary defects in ferritic/martensitic steels is not uniform and is likely controlled by the sink strength, as characterized by the ADR of GB planes.

Tentative

30A1

Towards a Comprehensive Understanding of MX-Type Precipitate Behavior Under Irradiation in Castable Nanostructured Alloys

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Castable Nanostructured Alloys (CNAs) are promising for fusion energy systems due to their high density of MX-type precipitates, which act as dislocation pinning sites and radiation sinks. This research employs both ex-situ and in-situ TEM ion irradiation techniques to better understand MX-type precipitate stability. Our study examines seven CNA variants across a wide range of irradiation variables: temperatures from 300-600 °C, doses of 1-100 displacements per atom (dpa), dose rates of $1x10^{-4}$ - $1x10^{-3}$ dpa/s, and helium concentrations of 0-25 appm He/dpa. First, we will demonstrate parity in results between in-situ and ex-situ ion irradiations in terms of precipitate stability, enabling our studies to obtain both high temporal fidelity and increased statistical significance. Using both techniques, we will demonstrate crucial insights into precipitate-sink evolution, with helium playing a significant role in modifying sink evolution and diffusional processes. These factors balance the delicate equilibrium between ballistic dissolution and back diffusion that contribute to MX-type precipitate stability by altering the concentration of migrating point defects and clusters. Our in-situ irradiation techniques also highlight dynamic restructuring in MX-type precipitates, with observed restructuring of TiC precipitates being influenced by morphology, size, and more significantly, their spatial correlation to existing and irradiation induced features. We will contrast our ion irradiation results with recent neutron irradiation results (6 dpa, 521 °C). Finally, we will cast the overall findings to existing mathematical models for precipitate stability to support the experimental findings while also advocating that a database now exists for parameterized high fidelity multi-scale modeling.

Tentative

30A2

F82H Development - Past and Future Issues

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Reduced Activation Ferritic/Martensitic (RAFM) steels are the benchmark structural materials for fusion reactor in-vessel components based on the chemical composition of high Cr heat resistant steels, with the added elements replaced to reduce long-lived induced radioactivity.

F82H steel (Fe-8Cr-2W-0.2V-0.04Ta-0.1C) is a RAFM steel developed in Japan, and has been in the leading position since it was proposed around 1985. The basic study of the comprehensive performance of F82H in a non-irradiated environment has been almost completed. It has also contributed to the international development of RAFM and has been used in round-robin tests under the international collaboration of the International Energy Agency (IEA), which began in 1992. In the Japan-US Cooperative Program, high-dose irradiation tests were carried out in FFTF at PPNL and HFIR at ORNL, and F82H has accumulated an irradiation database over a wide temperature range up to a maximum dose of 80 dpa. In addition, in the Broader Approach (BA) activities that began in 2007, the development of a Material Property Handbook (MPH), the development of advanced joining technologies, and the investigation of design technologies for pressurized in-vessel components used in the complex extreme environments inside the magnetic confinement fusion reactor have been carried out with the aim of using them as DEMO blanket structural materials.

This paper reviews the current status of F82H development and evaluation, with emphasis on its key microstructural feature, the properties achieved, and its fabricability. The technical challenges, potential risks, and R&D priorities for the practical application of fusion energy are discussed.

Invited

30A3

Effect of helium on low-temperature hardening/embrittlement (LTHE) in neutron irradiated isotopically tailored RAFM steels

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Reduced activation ferritic-martensitic (RAFM) steels are the most promising candidates for water cooled, and PbLi blanket systems. A major technical gap in understanding fusion-specific radiation damage in RAFM steels is the high helium (He) generation rate, ~10 atomic parts per million (appm) He/dpa, under the fusion neutron spectra. But in absence of a high flux 14 MeV neutron facility, the effect of He on mechanical properties degradation is challenging to estimate with test reactors due to a lack of He production in RAFM steels in a fission neutron spectra. Here, isotopic tailoring using 58/60Ni was used to stimulate helium generation in F82H steels neutron irradiated in the high flux isotope reactor (HFIR) from ~3 dpa to >85 dpa, >900 appm He. Based on HFIR data, the effect of He on low-temperature hardening embrittlement (LTHE) and overall tensile properties will be elucidated. Irradiation hardening and ductility loss results are compared with historical He effect data from spallation neutron (SPN) irradiations, highlighting significant differences or over-prediction of the effect of He on tensile properties after SPN irradiations as compared to test reactor irradiations. Microstructural understanding of the effect of He on LTHE of HFIR irradiated steels will also be discussed.

Tentative

30A4

Improving the balance between strength and DBTT in industrial-scale production of Castable Nanostructure Steels

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Advanced Castable Nanostructured Alloys (CNAs) are new reduced-activation ferritic/martensitic (RAFM) steels developed at ORNL. They are leading candidates for blanket structural materials in US fusion power plants. The state-of-the-art production of laboratory-scale CNAs have achieved a high volume-fraction and number-density of MX (M=Ti, Ta, V) nanocarbides in the microstructure, using conventional, economical, and scalable casting, forming and thermomechanical treatment technologies.

A major challenge encountered in the scale-up production of CNAs is to improve the balance between the high strength and low DBTT of materials in the industrial settings. Various heat-treatment parameters, including the normalizing and quenching temperatures, heating/cooling rates, and soaking times, were carefully investigated to reveal their effects on strength and DBTT. Such understanding enabled us to design an optimized heat treatment that complies with the industrial production requirements and simultaneously achieves high strength and low DBTT in ton-scale CNA plate. In this presentation, we will report the viability and advantages of optimized industrial-scale production of CNAs through a comprehensive evaluation of properties that include tensile, creep-rupture, low-cycle fatigue, and fracture toughness.

Tentative

30A5

Ion emulation of neutron irradiation effects on ferritic-martensitic steels

Zhijie Jiao[1], Logan Clowers[1], Kevin Field[1]

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Ion irradiation has been a favorable alternative in selection of candidate materials for nuclear applications thanks to its many advantages over neutron irradiation, including low cost, low to zero radioactivity and fast turnaround time. This talk reviews the efforts that have been made on the understanding of irradiation effects in F-M steels (T91, HT9, etc.) using the single ion, dual-ion (Fe+He) and triple-ion (Fe+He+H) irradiation facilities at the Michigan Ion Beam Laboratory for the past 10 years. Ion irradiated microstructures including dislocations, precipitates and cavities will be compared with available neutron microstructure data. Selection of proper ion irradiation parameters will be discussed. Dr. Klueh's book on ferritic and martensitic steels for nuclear applications has been frequently referenced and has provided invaluable guidance for this study.

Tentative

30A6

Accomplishments of the Japan-US FRONTIER Project -Fusion Research Oriented to Neutron Irradiation Effects and Tritium Behavior at Material Interfaces-

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The FRONTIER collaboration was launched in April 2019 to establish the scientific foundations for understanding reaction dynamics at the surfaces and interfaces of plasma-facing components (PFCs) for DEMO reactors. The project comprises four tasks to achieve this objective.

Task 1 conducted neutron irradiation experiments in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) to investigate the effects of microstructural modifications on the mechanical and thermal properties of plasma-facing and structural material interfaces. Accomplishments include the development of new dissimilar joints and layered structures and their irradiation responses.

Task 2 measured hydrogen isotope retention in neutron-irradiated tungsten using the Tritium Plasma Experiment (TPE) at Idaho National Laboratory (INL), revealing a quantitative correlation with the number density and size distribution of vacancy clusters. Additionally, this task examined the oxidation behavior of neutron-irradiated tungsten to construct a database of material behavior under accident conditions. The rhenium generated by neutron-induced transmutation significantly impacted the oxidation behavior of tungsten.

Task 3 focused on the corrosion characteristics of liquid tin as a divertor coolant. In-pile corrosion tests in HFIR, utilizing a specially designed irradiation capsule, demonstrated that a thin Al_2O_3 layer formed on FeCrAl alloys was not significantly damaged during neutron irradiation in liquid Sn (up to 1 dpa between 450 and 510 °C).

Task 4 integrates the findings of these task to develop an interdisciplinary understanding of phenomena in PFCs, ultimately advancing towards an engineering model. Detailed project accomplishments will be discussed.

Invited

30B1

Development of a Multiscale Modeling Platform for Fusion Neutron Irradiation Effects of Materials

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Due to the absence of a fusion-relevant neutron source, computer simulations have become as an essential and legitimate tool for understanding microstructural evolution and thermo-mechanical property of materials under fusion neutron irradiation. Herein, we present recent progress in developing a multi-scale modelling platform of materials under fusion neutron irradiation, using tungsten as a prototypical example. This platform, named NINUM3 (Neutron Irradiation of Nuclear Fusion Materials via Multiscale Modelling), includes two comprehensive databases and a suite of computational tools within a multiscale framework.

Using the NINUM3 platform, we simulate the microstructural evolution and irradiation-induced hardening of tungsten under various irradiation conditions. The simulation results are quantitatively consistent with experiments from both fission neutron irradiation and ion implantation with a specific helium-dpa ratio and transmutation content. Furthermore, we explicitly reproduce the self-assembly process of neutron irradiation defects, showing how chaotic displacement cascades evolve into observable void lattices. Additionally, we predict the microstructures and thermo-mechanical properties of tungsten under fusion neutron irradiation. Our results reveal that the number density of voids and dislocation loops under fusion neutron irradiation is significantly higher than that under fission neutrons, leading to a substantial increase in hardness and a decrease in thermal conductivity.

The NINUM3 platform demonstrates a robust approach to capturing the inherently multiscale nature of fusion neutron irradiation. It exemplifies the standard practice of multiscale modeling by passing parameters through simulations at different scales and is expected to play a critical role in the development of advanced fusion materials.

Invited

30B2

Multiscale modelling of defect evolution in fusion relevant materials

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Tungsten is utilized as the material in the harshest environments of fusion test reactors and proposed power plants. Iron and steel, on the other hand, will play a crucial role as structural materials. Efforts have been made both computationally and experimentally to understand the defect build-up and evolution under the conditions foreseen in future power plants. Computationally, mainly single impact simulations have been carried out to study the defects produced. More recently, high-dose irradiation simulations have been utilized, to obtain higher doses and to study how the produced primary defects are evolving. However, most studies are on small samples without grain boundaries, which limits both defect growth and ignores defect migration to sinks, and the atomistic simulations are limited in timescale.

We have developed and validated both speed-up techniques, which enables faster access to highly damaged cells, and multiscale modelling frameworks, which can take into account the long-time scale phenomena. Additionally, work on polycrystalline materials has been done, to understand the effect of grain boundaries on defect evolution in such systems. All of these have been done with atomistic simulations, to obtain the full atomistic details. Utilizing large cells shows distinct results differing from smaller samples and adding grain boundaries will greatly affect the defect evolution inside the grains. Utilizing multiscale models, we can understand the long-term evolution of the systems, inaccessible by conventional methods. The atomistic configurations are directly used as input in various analysis simulations, which enables direct comparison to experiments.

Tentative

30B3

Computational framework for modeling TiB2 under fusion conditions

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We develop a comprehensive computational framework for modeling the behavior of ultra-high-temperature TiB_2 ceramics under melting conditions, considering both plasma material interactions (PMI) and radiation damage due to fusion neutrons. The framework incorporates multiple levels of phenomena from atomic to mesoscopic and is based on a machine learning (ML) approach used both training empirical interatomic force fields (EIFFs) and coupling the different scales. Critical mechanisms and parameters are verified by tailored experiments. Extensive ab initio modeling of systems, bulk and defect properties, and atomic transport has developed a database of material and radiation defects properties, including solid, liquid, and amorphous phases. The database is used to train the EIFFs using the developed ML approach. EIFFs are used in large-scale PMI and radiation damage simulations to derive mechanisms and parameters that will be used in higher-level radiation effects simulations using kinetic Monte Carlo, phase field and rate theory. In this presentation, we report the main results obtained in the first stage of developing a framework incorporating EIFF and evolution mechanisms and their parameters obtained for D-plasma interactions with TiB_2.

This research is supported by the ORNL LDRD project “Computational Framework for Modeling Titanium Diboride Chemical Erosion by Plasma”.

Tentative

30B4

Crystal plasticity finite element modelling of thermal and irradiation creep in fusion structural materials

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Creep is irreversible, time-dependent inelastic deformation that occurs when the applied stress is below the material yield stress. The creep resulting from neutron irradiation in the structural materials of fusion reactors has the potential to limit the lifetime of fusion reactors.

Fusion reactors require structural materials to withstand extreme environments, including high neutron flux and elevated temperatures. Experimental data on irradiation creep under fusion-relevant conditions are scarce, making physically based models essential for predicting creep behaviour in structural materials. Crystal plasticity finite element (CPFE) models are well-suited for this purpose, as they can capture microstructural evolution, which is critical for accurately modelling creep.

A physically based CPFE model for thermal and irradiation creep is presented that has a dependence on temperature, applied stress, and radiation dose rate. This creep model is based on the glide of dislocations enabled by a dislocation climb mechanism that is controlled by the diffusion of point defects. The model response is calibrated against experimental data to ensure predictive capability. The model is applied to Grade 91 steel, as a candidate structural material for fusion applications.

Tentative

30B5

The Influence of Stacking Fault Tetrahedra on Void Swelling in Irradiated Copper

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Void swelling in irradiated structural materials presents a long-standing challenge, with face-centered cubic (FCC) materials exhibiting significantly higher swelling rates (~1% dpa^{-1}) compared to body-centered cubic (BCC) materials (~0.2% dpa^{-1}). Despite extensive research, the fundamental mechanisms driving this difference remain incompletely understood. Here, through a combination of atomistic simulations and STEM imaging, we identify stacking fault tetrahedra (SFTs) as the primary contributors to the high swelling rate in pure FCC copper. Contrary to conventional understanding, our findings reveal that SFTs are not neutral sinks but highly biased defect absorbers. Specifically, SFTs preferentially absorb mobile self-interstitial atoms (SIAs) at their faces and vertices while exhibiting weak vacancy absorption along their edges. By comparing predicted swelling rates with experimental data across various irradiation conditions, we demonstrate the distinct impact of SFTs and other microstructural features on swelling behavior. These insights enhance the fundamental understanding of irradiation-induced swelling in structural materials and may inform the development of materials with improved radiation resistance.

Tentative

30B6

New Insights into Quantifying Vacancy Distributions in Self-Ion Irradiated Tungsten: A Combined Experimental and Computational Study

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In light of the potential future energy shortage, it is crucial to explore sustainable solutions that can also help reduce greenhouse gas emissions. Thermonuclear fusion power plants could be a key solution. Refractory metals are considered the best compromise to cover the plasma-facing components of these reactors. However, under harsh operational conditions, material properties can degrade over time. To guarantee the safety and efficiency of future tokamaks, it is essential to predict the impact of various factors such irradiation on materials. Multi-scale modeling can help in simulating the effects of irradiation on microstructural evolution. To validate these models, experimental data are required, particularly with regard to the action of radiation-induced defects.

In this study, we characterized a set of self-ion irradiated tungsten samples, ranging from those containing primary defects to those with oversaturated defects, using positron annihilation spectroscopy (PAS). We present a novel approach for estimating the fraction of vacancy-type defects induced by self-irradiation at various temperatures. By employing an optimization algorithm parametrized with a positron trapping model, we extracted complete vacancy distributions from experimental data. This model includes positron annihilation characteristics for different vacancy clusters calculated by the Two-component density functional theory. The method was validated against simulation results and Transmission electron microscopy (TEM) observations. After high-temperature irradiation, we identified small clusters that were undetectable by TEM, with concentrations significantly exceeding the defects visible by TEM (10^{24} m^{-3}). Taking into account positron trapping in the Oxygen-vacancy complex faithfully reproduces the experimental results for high-temperature irradiation.

Template

30C1

Understanding the limitations and potential of micro tensile testing of tungsten and needs for crystal plasticity modelling

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Tungsten is the leading plasma facing material candidate due to its exceptional properties. Understanding the response to neutron irradiation is crucial for the lifetime evaluation of tungsten. The limited space in nuclear reactors and the high levels of radioactivity of the specimens after irradiation are significant barriers to accurately measuring the mechanical properties after neutron irradiation. Testing of micro tensile specimens is one approach to reduce the total amount of irradiated material needed for a set of mechanical testing experiments. However, micro samples are not necessarily measuring the bulk material properties, as size effects produce higher measured mechanical properties than is observed in engineering size counterparts. Determining the minimal sample size for reliable bulk property measurement, combined with modeling efforts, is essential.

We conducted room temperature tensile tests on tungsten micro tensile specimens fabricated with focused ion beam, plasma focused ion beam and femto-second laser ablation system to dimension of 2×2×7 μm^3, 7×7×18 μm^3, and 80×100×233 μm^3 (width×thickness×gauge-length), respectively. Elevated tensile testing was performed up to 475°C on 5×5×18 μm^3 specimens fabricated with focused ion beam. The smallest specimens exhibited a high degree of ductility and strength, whereas the largest specimens demonstrated behavior akin to bulk tungsten. To investigate the effects introduced by the micro specimen fabrication processes and to obtain the necessary bulk dislocation density for the crystal plasticity model, we employed X-ray diffraction depth profiling. This measurement was performed using different X-ray sources with varying penetration depths. Finally, the potential and limitations of micro mechanical tests for tungsten will be discussed.

Tentative

30C2

Comparison of the mechanical properties and recrystallization temperature of W-K materials prepared by two plastic-deformation technologies

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The application of pure tungsten (W) as a plasma-facing material faces critical challenges due to intrinsic brittleness at low temperatures, mechanical property degradation from high-temperature recrystallization, and neutron irradiation induced embrittlement. Potassium (K) doping is an effective strategy to suppress recrystallization and enhance the mechanical performance of tungsten. Although K doping has been successfully implemented in tungsten filament production, the fabrication of W-K bulk materials remains a significant challenge. In this study, we fabricated W-K bulk materials using two distinct hot plastic-deformation techniques: hot-rolling and hot-swaging. The effects of plastic-deformation methods on the mechanical properties and recrystallization behavior of the prepared W-K materials were investigated. Tensile tests using large specimens were conducted at temperatures ranging from room temperature to 1500 °C in a vacuum environment. Both materials demonstrated exceptional tensile ductility at low temperatures, with the swaged W-K rod exhibiting remarkable room temperature ductility. However, significant variations in mechanical strength were observed among the different deformation methods. The hot-rolled plate had superior tensile strength, especially under high temperature conditions, compared to its swaged rod. The recrystallization behavior were investigated through isothermal annealing at temperatures ranging from 1300 °C to 1700 °C for 1h. The Vickers hardness and microstructure of the annealed specimens revealed that the recrystallization temperature of the swaged rod and the rolled plate exceeded 1500 °C. To better understand the mechanical properties and thermal stability of the prepared W-K materials, the microstructures and the K bubbles morphology were analyzed and the relationship between properties and microstructures was explored.

Tentative

30C3

Atomistic simulations of hydrogen dynamics on tungsten surfaces using machine learning potential

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For plasma-facing materials of fusion reactors, such as tungsten (W), understanding the behavior of hydrogen isotopes is important for predicting tritium retention. However, little is known about the detailed surface processes due to the difficulty of performing in-situ measurements in experiments. In addition, it has been challenging to efficiently and accurately describe complex surface processes in atomistic simulations. Recently, the accuracy-efficiency tradeoff of atomistic simulations has been greatly improved due to the advent and advanced implementation of machine learning potential models (MLIPs), which can achieve accuracy comparable to density functional theory (DFT) calculations at a much lower computational cost. In this study, after constructing and validating an MLIP for W-H systems, we performed molecular dynamics (MD) simulations using the MLIP for various surface processes, such as (i) the sticking and reflection behavior of hydrogen isotopes irradiated on W surfaces at 0. 1-100 eV, including the effect of pre-existing surface hydrogen atoms, (ii) the thermal adsorption and desorption behavior of hydrogen gas, including the estimation of reaction rate constants for various temperatures and surface coverages, and (iii) the behavior of hydrogen atoms continuously implanted on the surface at 10-100 eV. These MD simulation results provide not only a comprehensive understanding of the behavior and fate of energetic hydrogen in plasma-facing tungsten, but also kinetic models for fundamental processes that can be used to predict the long-term behavior of hydrogen isotopes through a macroscopic rate model.

Invited

30C4

High-Velocity Dust Impact on Tungsten Fuzz as a Plasma-Facing Component in Fusion Reactors

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Tungsten (W) fuzz, a nanostructured morphology formed under helium (He) plasma exposure, significantly alters the properties of plasma-facing components (PFCs). While fuzz growth mechanisms are well studied, its response to high-velocity dust impact-critical in fusion reactors-remains unclear. Tungsten dust, generated by plasma erosion and transient events, can impact fuzz-covered surfaces at extreme velocities, influencing PFC integrity, erosion dynamics, and tritium retention.

This study investigates W dust impacts on W fuzz using large-scale molecular dynamics (MD) simulations, incorporating more than 500 million atoms at velocities of 1-3 km/s with advanced interatomic potentials. Results reveal impact-induced densification, nanorod collapse, and partial ablation, with thicker base regions exhibiting greater resistance. A nanorod thickness gradient influences impact response, aligning with experimental observations. These findings suggest that fuzz evolution is driven not only by plasma exposure but also by dust redeposition, leading to modifications in erosion rates and mechanical stability.

Understanding the coupled effects of plasma-driven nanostructuring and impact-induced modifications is essential for predictive PFC performance modeling. This study provides new insights into tungsten fuzz resilience under extreme conditions, informing erosion mitigation strategies and optimizing fusion reactor materials.

Tentative

30C5

Neutron-based techniques for stress measurement and defect inspection: round robin tests for high-heat-flux components

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Tentative

High-heat-flux components (HHFCs) are critical in-vessel components in a fusion reactor in terms of thermal and nuclear loads. The severe heat loads demand an extraordinary heat exhaust performance together with mechanical robustness. A typical HHFC consists of plasma-facing armour (tungsten) and water-cooled heat sink body (copper) to cope with different functional requirements. Such a bi-material construction necessarily requires joining of the two materials with dissimilar properties. Joining quality and structural integrity pose a fundamental impact on the reliability and durability of HHFCs. Here, defects and residual stress play a key role and thus need to be scrutinized. Neutron-based nondestructive tests offer effective inspection tools for detecting defects or measuring stress in a HHFC with a significant penetration depth even for highly absorbing materials like tungsten.

In the framework of the EUROfusion divertor project, we applied neutron-based tomography and diffractometry for tungsten monoblock target mock-ups. For stress measurement, extensive round robin tests were conducted using an identical reference mock-up at four different neutron source facilities (FRM-II, SINQ, HANARO, ISIS) in a comparative round robin test campaign. For tomography and SANS, ISIS and PSI were employed. In this contribution, an overview of the application cases of neutron-based examination techniques for HHFCs is given and the latest results of the tests are presented. Residual stress profiles measured at four different diffractometers are compared with FEM-based computational prediction and also with other independent measurement methods. Potential benefits and technical limitations are discussed and prospects for other joined components (e.g. welded pipes) are given too.

30C6

Crack elimination and mechanical property of 3D printing pure tungsten and tungsten alloys

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Tungsten and tungsten alloys are the candidate plasma facing materials in future nuclear fusion power reactor, due to their high melting temperature, strength and thermal conductivity. However, tungsten has poor processability due to its general lack of ductility [1]. Against this limitation, emerging three-dimensional (3D) printing or additive manufacturing (AM) technique, such as laser powder bed fusion (LPBF), allows direct fabrication of geometrically-complex tungsten structure [2]. However, tungsten and tungsten alloys suffer from serious cracking problem during LPBF processing [3]. Existing crack suppression methodologies of nucleant addition, composition modification and preheating show limited success in eliminating tungsten crack. Here we report successful fabrication of crack-free LPBF pure tungsten and tungsten alloys through novel scanning strategy, which not only eliminate the tungsten crack but also refine the microstructure to enhance mechanical properties. High temperature tensile property of LPBF tungsten is reported with ductile-to-brittle transition temperature down to 250 °C. The strategy is also tested in carbide-dispersion-strengthened and solid-solution-strengthened tungsten alloys, showing similar crack suppression capability. Our study opens a new avenue for 3D printing of crack-free tungsten and refractory metal and alloys.

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Invitation

30D1

EFFECTS OF HFIR RB-19J THERMAL NEUTRON-SHIELDED IRRADIATION IN TUNGSTEN METALS AND ALLOYS

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Tungsten is the ongoing focus as the material for fusion reactor plasma-facing components (PFC). As the PFC materials, tungsten and tungsten-based materials present a unique set of advantages. However, while the pristine properties of these materials may be adequate for the PFCs in early technology demonstration fusion devices, it is widely perceived that the neutron irradiation may significantly alter their properties, reliability, and service life due to the synergistic effects of atomic displacement damage and nuclear transmutations. Indeed, studies conducted using fission neutrons have revealed that many factors defining the design allowables of tungsten-based materials, such as the ductile-to-brittle transition temperature, recrystallization envelope, and mechanical properties at the intermediate temperatures, are severely impacted by neutron irradiation.

Within the US-Japan collaboration framework, RB19J irradiation project was conducted using the High Flux Isotope Reactor (HFIR) of ORNL. This campaign explored the effects of irradiation in over 800 specimens of a variety of tungsten materials at temperatures 600-1,200 °C to the maximum fluence (E>0.1 MeV) of ~3.8E+25 n/m^2. The thermal shielding in the RB19J capsule not only realizes the transmutations-to-dpa ratios more relevant to fusion conditions than in some of the previous experiments but also enables comparisons with accelerated transmutation experiment conducted in the same framework using unshielded capsules in HFIR. RB19J results have been reported in several publications, allowing more comprehensive understanding of the mechanisms controlling the irradiation effects phenomena in fusion nuclear environment. This paper discusses the fundamental insights and remaining key questions related to fusion nuclear performances of tungsten materials.

Tentative

30D2

Study of W- based high entropy alloys for nuclear fusion applications

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The production of clean energy from the fusion of light mass particles, using nuclear fusion reaction facilities, has been under study for the past 50 years. One major challenge is the developing of materials for the reactor's divertor, which supports high thermal flux. Tungsten was chosen as the plasma-facing material, while a CuCrZr alloy will be used in the cooling pipes. However, the gradient between the working temperatures of these materials requires the use of diferent plasma facing components. A proposed strategy to solve this issue is to produce funtional gradient materials (FGMs) envolving the refractory high entropy alloys (CrFeTiTa)70W30 and VFeTiTaW. This individual layers were prepared by mechanical alloying and sintering, and their thermal and irradiation resistance was evaluated. Both alloys showed phase growth after annealing at 1100 °C for 8 days, being more pronounced for higher temperatures (1300 °C and 1500 °C). The VFeTiTaW alloy presented greater phase growth, suggesting lower microstructural stability, however no new phases were formed. Both (as-sintered) alloys were irradiated with Ar^+ (150 keV) with a fluence of $2.4x10^{20}$ at/m^2, as well as He^+ (10 keV) and D^+ (5 keV) both with a fluence of $5x10^{21}$ at/m^2. The morphology of the surface of both samples was analyzed before and after irradiation showing no severe morphologic changes, indicating high irradiation resistance. Additionally, the VFeTiTaW alloy presented a lower deuterium retention (8.58%) when compared to (CrFeTiTa)70W30 alloy (14.41%), making it a more promissing FGM.

Tentative

30D3

Recent progress on coating-based W/steel first wall structure through detonation spraying in Shenzhen University

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The manufacturing and in-situ repair of tungsten (W)/steel first wall structure for the blanket are crucial for the advancement and sustained operation of nuclear fusion reactors. Shenzhen University has made significant strides in recent years by developing a coating-based W/steel first wall structure utilizing detonation spraying technology.

The process began with the preparation of a pure W coating on a steel substrate, which exhibited excellent comprehensive properties including density, purity, bonding strength, hardness, thermal shock resistance and so on. Subsequently, a pure and dense iron (Fe) coating was deposited as an interlayer between the W and steel to further enhance performance under high heat flux (HHF) conditions. The primary thermodynamic properties of both W and Fe coating materials have been thoroughly characterized. Based on these characterizations, the thicknesses of the W and Fe coatings in the W/Fe/steel first wall structure were optimized through finite element analysis to ensure optimal performance under steady-state HHF conditions. Following this optimization, W/Fe/steel first wall structure modules with the optimized thickness design were manufactured and successfully tested under 1000 cycles of 1 MW/m^2 HHF loading and long-term steady-state HHF conditions. Additionally, preliminary efforts have been made to develop W/steel functional gradient coatings without the formation of FexWy intermetallics through detonation spraying.

These developments collectively demonstrate that detonation spraying is a viable method for manufacturing and in-situ repair of W/steel first wall structures for the blanket, offering distinct advantages in terms of efficiency and effectiveness.

Tentative

30D4

LHD divertor plasma induced surface modifications on unirradiated tungsten and pre-irradiated tungsten

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Tungsten (W) was selected as the plasma facing material of ITER. Understanding and predicting of the plasma induced W surface modifications are important for ITER from the viewpoints of dust formation and plasma facing component lifetime. One characteristic of divertor plasma on a divertor tiles is the non-uniform distribution of plasma parameters at the scrap off layer (SOL) region, the strike point, and the private flux region. Moreover, fusion neutron will form irradiation defects uniformly along the depth direction of W. A critical question is what kind of impact will irradiation defects have on the plasma induced W surface modifications at different divertor regions. In the present study, the unirradiated W and iron (Fe) ion pre-irradiated W were exposed to deuterium (D) plasma, helium (He) plasma, and D/He mixed plasma separately at LHD divertor leg position. Thereafter, the surface modifications on these two kinds of W samples at the SOL region, the strike point, and the private flux region were characterized by scanning electron microscopy (SEM) and transmission electron microscope (TEM). The typical surface structures after D, He and D/He mixed plasma exposure are oxygen enhanced amorphous W, undulating structure, and pinholes, separately. No significant difference in the He plasma induced surface modifications was found between the unirradiated W and the pre-irradiated W. In the case of D plasma and D/He mixed plasma exposed W samples, pre-irradiation defects have impact on the formation of plasma induced surface structures.

Tentative

30D5

Nuclear Transmutation and its Impact on In-Service Performance of Tungsten

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Invited

Tungsten is among the most promising plasma facing materials for future fusion reactors. When in service in fusion reactors, exposure to the 14.1 MeV-peaked neutron spectrum and high-energy particle flux produces significant microstructure alteration. The synergistic interactions between radiation damage, evolving composition, and helium or hydrogen isotope particles impose significant challenges to the performance of W when subjected to fusion-relevant environment.

Over the past decade, the community has accumulated knowledge of transmutation in W following neutron irradiation in various thermal fission nuclear reactors under two major fusion material science programs (US-Japan joint projects and EUROfusion). This presentation will summarize the key insight regarding the irradiation dose, temperature and neutron spectrum dependence of the configuration and size distribution of defect features consisting of transmutated Re and Os in neutron-irradiated W and the consequential thermomechanical property degradation. Additionally, the in-service performance of W containing transmutation elements, or transmutation-induced second phase precipitates, as a result of exposure to helium plasma will be reviewed, including the impact on hydrogen isotope retention. Atomistic simulations of the interactions between the transmutation elements and helium will also be discussed to facilitate the interpretation of the experimental results. In addition, recent experimental and theoretical investigation of the influence of the transmutation elements on deuterium retention in W will be summarized.

Finally, the presentation on the knowledge gap and research needed to fully understand the transmutation-induced microstructural evolution in W as well as the impact on W performance in fusion-relevant environment will be discussed.

30D6

Ductlisation of Tungsten using Rolling and Rhenium Alloying

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Tungsten is the primary candidate material for the plasma facing components of future fusion reactors. The high activation energy required for the movement of screw dislocations in tungsten leads to poor room temperature ductility which limits its use in load-bearing components. Whilst the neutron irradiation induced precipitation of Re/Os-rich sigma and chi phases are known to be detrimental to mechanical properties, the generation of a high density of defect sinks can be achieved through rolling to reduce grain boundary spacing. This can both promote exceptional room temperature mechanical properties and could potentially decrease the concentration of point defects that facilitate precipitation. Atom Probe Tomography has been used to reveal the formation of Os and Re rich clustering under both neutron and ion irradiation in unrolled material. Rolling is also understood to improve the toughness of W and reduce the DBTT. In this study both rolling and Re alloying are used to demonstrate the excellent room temperature ductility in sub-scale tensile testing and 3-point bending tests.

Tentative

30E1

Irradiation temperature dependence of hardening and microstructure in Eurofer97/3 using thermal gradients

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Reduced activation ferritic-martensitic steels are a leading candidate structural material for use in the blanket of future fusion reactors, where they will be subjected to high neutron fluxes. The damage microstructure resulting from irradiation of these materials is sensitive to the irradiation temperature, causing changes to the mechanical properties which can define operational limits.

The study of irradiation temperature effects using materials test reactors can be challenging due to limited control and instrumentation, with expensive experimental campaigns lasting years to reach desired doses. In collaboration with the Ruđer Bošković Institute (RBI), the United Kingdom Atomic Energy Authority (UKAEA) has developed a rig to facilitate charged particle irradiation of samples subjected to a thermal gradient, allowing the simultaneous irradiation over a wide range of temperatures in a single experiment, at greatly reduced cost and with superior instrumentation. Here we present results of such an experiment on Eurofer97-3, covering the whole range of potential operating temperatures.

Dual beam irradiation was used with Fe ions to produce displacement damage, and He implantation to mimic transmutation gases at fusion relevant He to dpa ratios. Nanoindentation and TEM were used to study the mechanical properties and microstructure as a function of irradiation temperature and compared to results of neutron irradiation studies with good agreement.

Tentative

30E2

Multiscale computational modeling of the helium effect on cavity swelling in F82H steel under different irradiation fields

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A blanket structural material in a fusion DEMO reactor suffers from 14-MeV neutron irradiation bombardment which induces lots of production of irradiation defects. Especially helium (He) is known to strongly enhance cavity swelling. A realistic design of a fusion DEMO reactor requires a methodology that can theoretically interpret the existing experimental irradiation data and can predict the material behavior in a fusion DEMO environment.

The objective of this study is to theoretically understand and visualize He-cavity swelling in F82H steel, focused on the nucleation and growth processes of the cavities, under different neutron irradiation fields: HFIR, HFR, SINQ and a fusion DEMO. The formation kinetics of He-cavities and SIA-clusters was described by a mean field cluster dynamics method based on reaction rate theory with atomistic calculation results; this can trace the irradiation-condition dependent nucleation free energy of defect clusters.

In the calculation, the cavity swelling shows a peak at each irradiation dose; the peak swelling and the peak temperature depend on irradiation parameters such as temperature, dose rate and He/dpa ratio. However, the He effect on cavity formation enhancement has a threshold at low irradiation doses; the value is about 0.1 appm He/dpa. The obtained calculation results provide a good interpretation and reproducibility of some of experimental results. Furthermore, correlation maps of He-cavity swelling between the different neutron irradiation fields were then constructed. For irradiation conditions (<400 °C, <20 dpa), the relative values between the fusion DEMO and the other facilities were estimated to be less than 20 %.

Tentative

30E3

Effects of Mn content and normalisation temperature on radiation induced clustering in ferrous alloys for fusion.

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Through systematic variation of material composition, primarily looking at manganese content, this work explores the effects of life-limiting factors, including low-temperature embrittlement, on breeder blanket structural materials and aims to expand the operational window of current steel options. Existing literature shows manganese contributes to irradiation induced degradation and enhances clustering effects in steels, which will result in embrittlement upon the cool-down of a reactor. There has also been work to show that increased prior austenite grain size, resulting from changes to heat treatment routes, may reduce the change in stress post-irradiation. To further this knowledge, Fe-based model alloys have been produced with variations of Mn content and normalisation temperatures. These samples have been subjected to stimulant reactor irradiation (1.5 MeV protons), and the effects on the formation of clusters, and the resultant low temperature embrittlement, analysed. In addition to standard metallography analytical techniques, due to their atomistic scale, transmission electron microscopy (TEM) and atom probe tomography (APT) will be used to determine the formation, location and size of the clusters, particularly their vicinity to irradiation induced defects and dislocation loops. In addition, due to the relatively near surface effect of proton irradiation, nano-indentation and grazing incidence x-ray diffraction (GIXRD) will be used to determine changes in mechanical properties and crystal structure respectively. Verification of critical content of Mn will then aid development into industrially applicable materials and set significant boundaries on the composition of fusion steels.

Invitation

30E4

Effect of high displacement damage at different dose rates and temperatures on Deuterium retention in Eurofer97

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Open volume defects in irradiated EUROFER97 are responsible for gas retention and embrittlement. EUROFER97 exhibits minor swelling, but quantitative data at large damage doses are scarce. In this study, deuterium (D) retention is used as an indicator for the evolution of open volume defects at different temperatures and dose rates. EUROFER97 samples were damaged with a high dose rate of 2.5 - 16.5 dpa/hr using 11.6 MeV Au ions to high displacement damage doses (50, 100 dpa) at (100 °C- 400 °C). Rutherford backscattering (RBS) with 4 MeV protons was used on iron surrogates to characterize the Au implantation profile. Focused ion beam cuts in the sample coupled with energy-dispersive X-ray spectroscopy match the RBS results. The Au depth profile shows a region of high displacement damage of 500 nm that is gold-free for uncontaminated defect analysis. X-ray diffraction analysis showed clear differences in both the lattice constant and the width of the Bragg peaks between the highly damaged samples and the undamaged reference. This verifies the novel method of creating highly damaged EUROFER97 samples in a short time using Au ions. The samples were subsequently exposed to a D plasma at 100°C and low energy (< 5 eV/D) until full D defect decoration. D depth profiles were measured using 3He nuclear reaction analysis. Further information on D de-trapping energies is obtained using Thermal Desorption Spectroscopy. Presently, irradiations at lower and constant damage rates (2 dpa/hr) are underway. D retention results will be compared with previous low-dose (0.5 dpa) irradiated samples.

Invitation

30E5

The role of temperature in radiation hardening of neutron-irradiated Fe-(8-10%)Cr model alloys

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The exact mechanisms driving low-temperature radiation hardening and embrittlement-which result in a loss of ductility and lowered fracture toughness-in reduced-activation ferritic/martensitic (RAFM) steels for fusion structural applications remain uncertain. In the present study, ultra-high-purity Fe-(8-10%)Cr binary alloys, representing the simplest model with minimized impurity effects, were neutron irradiated at the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) over a temperature range of 200-450 °C, with neutron damage of ~5 dpa. Mechanical testing of the neutron-irradiated Fe-(5-15%)Cr samples performed at ORNL, revealed an increased peak hardening of up to 160% at ~350 °C, accompanied by a clear decrease of up to 8% in the room temperature electrical resistivity for irradiation temperatures between 200 and 350?°C. Notably, the hardening for Fe-(8-9%)Cr was comparable to Fe-Cr alloys with slightly higher or lower Cr content, contrasting with some prior reports.

A comprehensive microstructural study was conducted to investigate the hardening mechanisms in Fe-(8-10%)Cr. Foils for transmission electron microscopy characterization were prepared using focused ion beam liftout techniques for comprehensive dislocation loop analysis. Detailed information on the effect of irradiation temperature on the dislocation loop parameters, cavity formation, and Cr-rich solute segregation and/or precipitation will be summarized. The microstructural results will be used to quantitatively compare the predicted versus measured hardening. These findings will provide insights into the fundamental irradiation effects on Fe-Cr material behavior and support the development of advanced radiation-resistant materials.

Tentative

30E6

Characterization of the radiation hardenings of Fe9Cr and Eurofer97 by means of ball-indentation and micropillar compression

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Fe-9Cr model alloys consist of basic chemical compositions and microstructures of reduced-activation ferritic/martensitic steels (RAFM), e.g. Eurofer97, which are considered as the candidate structural materials for the first wall/blanket of fusion reactors. In the structural component, helium atoms, which are commonly produced through the (n, α) transmutation reactions up to about 15 appm/dpa, can worsen the mechanical properties of materials.

Because it is pretty challenging to scientifically obtain the mechanical properties of the shallow damaged layer of several μm, 28 MeV He ions were chosen for the irradiation at room temperature in this work. The energy degrader was applied to obtain the damaged layer of ~ 50 μm in depth having around 1.2 dpa and nearly homogenously distributed He atoms of ~ 190 appm.

Indentation tests with around 2 and 10 μm (in diameter) ball tips, micropillar compression, as well as transmission electron microscopy (TEM) were done to study the He effects on the hardening and microstructures.

After the ion-irradiation, spherical He bubbles of around 1.83 ± 0.27 nm/3.72 × 10^{23} m^{-3}, 1.14 ± 0.18 nm/2.43 × 10^{23} m^{-3} were formed in Fe9Cr and Eurofer97, respectively. The indentation stress obtained by the ball tips shows the strong dependance of tip radius but weak dependence of indentation strain. The underlying mechanism has been discussed in comparison with the results of micro pillar compressions.

Tentative

30F1

TEM characterization and statistical analysis of radiation-induced defect microstructure in tungsten and tungsten alloys

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Tungsten (W) and W-based alloys are key materials for the first wall applications in fusion technology due to their outstanding properties including a high melting point and excellent spattering resistance. The alloys, such as W-Y_2O_3, W-TiC, W(K), and W-Re, have been developed to enhance performance in terms of strength, ductility, and resistance to thermal cycling and erosion. To assess their microstructural behavior under neutron irradiation, the microstructure of the irradiated alloys was compared with that of pure tungsten subjected to a damage dose of up to 1 dpa.

Here, we provide quantitative analysis of neutron radiation-induced defects, such as dislocation loops, voids, and precipitates formed in various W alloys. It shows the evolution of these defects as a function of the damage dose at different temperatures. It was found that the size of the loops and voids increases, while their number density decreases with damage dose. The study also addressed the segregation behavior of Re and Os at voids, loops, dislocations, and grain boundaries, helping to identify the conditions for the formation of Re/Os precipitates. The results were compared with those obtained from W-based alloys irradiated at various temperatures. The study highlights also the effect of Re, TiC, Y_2O_3 particles and K bubbles on formation of radiation induced defects.

These insights contribute to the ongoing development of new W materials for fusion applications, emphasizing the need for optimized microstructural stability and irradiation resistance in extreme environments.

Tentative

30F2

Feasibility study of a TEM/STEM analysis for the characterization of irradiation-induced heterogeneous microstructures

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Fine particles in RAFM steels such as M23C6 and MX contribute to improved thermal aging and creep properties, making them important material factors in F82H steel. Despite this, these particles become unstable under irradiation, especially M23C6, which undergoes amorphization due to relatively low-temperature irradiation. To better understand this mechanism, it is necessary to clarify the heterogeneous microstructure formed by irradiation. For example, it is known that crystallographic, chemical, and topological disordering strongly influence the manifestation mechanism of radiation-induced amorphization, but many challenges remain in its quantitative evaluation. Therefore, in this study, high-resolution ABF/ADF, STEM-EDS, and EFTEM observations were performed for the ion-irradiated F82H as well as for $SrTiO_3$, which served as a standard material for TEM/STEM observations, with the aim of proposing advanced microstructure characterization techniques.

The results of column resolution ABF/ADF imaging of irradiated SrTiO3 show a decrease in Sr and Ti column intensities with increasing irradiation dose. This decrease is initially monotonic, then slows down above 2 dpa and accelerates again above 20 dpa. This behavior suggests the accumulation of topological and chemical disordering, which is consistent with previous studies. In the high-resolution STEM-EDS mapping, even with a relatively low spatial resolution probe, periodic concentration modulation due to the atomic column arrangement was observed, and the irradiation-induced heterogeneous microstructures were also observed. This finding allows the characterization of irradiation microstructures in various materials, and the results of irradiated F82H steel using a similar technique will also be presented in my talk.

Tentative

30F3

Synchrotron-based characterization of radiation damage in Ultra high temperature ceramics

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Tungsten is the leading candidate for plasma-facing components (PFCs) due to its attractive properties, resistance to sputtering, and compatibility with tritium. However, prolonged exposure to fusion plasmas and 14 MeV neutrons raises concerns about its resilience against plasma-induced surface damage and degradation of bulk properties. Rapid embrittlement from neutron exposure is a potentially insurmountable issue for the first-wall, which will experience the most extreme neutron fluence. As a result, ultra-high temperature ceramics (UHTCs) with unique thermal and mechanical properties have emerged as promising first-wall materials. Nevertheless, significant technological and knowledge gaps exist in their manufacturing and radiation response, both of which are essential for producing integrated PFCs for viable fusion energy. This work describes a fundamental study detailing the fabrication, properties, and initial neutron irradiation performance of primary candidate UHTCs for fusion, contrasting the behavior of carbide (TiC, ZrC, NbC) and boride-based (TiB_2, ZrB_2) UHTCs. While process parameters to achieve dense materials will be discussed, focus will be on resulting microstructure and implications for performance. Through advanced synchrotron-based scattering methods (XRD and PDF), we gain insights into the effects of irradiation on atomic and microstructural evolution. We find that lattice swelling in hexagonal borides is significantly higher than in cubic carbide counterparts. The proliferation of radiation-induced defects is apparent in all UHTCs, with broadening of XRD peaks clearly indicating an increase in two-dimensional defects. Our initial results based on fabrication and performance testing have positive implications for future PFC research.

Tentative

30F4

Micro Fracture Strength of Neutron Irradiated F82H by using μ-Tensile Testing

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Reduced activation ferritic/martensitic steels (RAFM) are leading candidates for blanket structural materials of the fusion DEMO reactor. It is especially important to evaluate the effects of 14 MeV neutron irradiation on material properties. However, fusion neutron irradiation facilities are not presently available. As well known, multi-ion irradiation is the most convenient and accurate way to simulate the 14 MeV neutron irradiation conditions, but this irradiated layer is limited in a small volume. The key is to extract more characterization from such a small volume. Notched micro tensile tests have been used to evaluate the micro fracture strength of single-block F82H after multi ion irradiation. However, comparison with macroscopic fracture characterization data has been difficult.

In this study, micro fracture strength is obtained from neutron irradiated F82H, for which macro strength data have already been obtained, using a micro tensile test method. The specimen used is the end of 1/3CVN specimens of F82H IEA irradiated at RB-11J at HFIR in ORNL. These specimens have an absorption energy of approximately zero at room temperature. The presentation will discuss the results of a comparison of micro fracture strength between ion irradiated and neutron irradiated F82H irradiated at 300 °C.

Tentative

30F5

Tensile and Permeation Properties of Austenitic Stainless Steel Tubing with Internal Hydrogen/Tritium

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Tritium storage and transport is one of the major challenges to fusion reactors. Tritium, being an isotope of hydrogen, embrittles most structural metals that typically meet design requirements. Additionally, helium-3, tritium's decay product, has been shown to cause further embrittlement. Structural integrity of tritium containment materials is of utmost importance, not only for the effectiveness of the reactor system but also the safety of personnel and facilities. The effects of tritium and helium-3 on the mechanical properties of stainless steel have been studied at Savannah River National Laboratory for decades due to its use for tritium handling systems. The present work features results from tensile testing 304L tubing that was tritium pre-charged and aged at -80 °C to allow tritium decay while minimizing losses to off-gassing. Tensile strength, reduction of area, and deuterium permeation was measured in an attempt to understand the hydrogen kinetics related to hydrogen-assisted fracture. Six different microstructures were studied, ranging from cold worked with fine grains and high yield strength to fully annealed with large grains and relatively low yield strength, to determine differences in tritium effects related to grain morphology. Upon tritium pre-charging and aging, the highest strength microstructures exhibit the greatest loss of ductility, and the lowest strength microstructures exhibit the greatest increase in yield strength.

Tentative

30F6

Anisotropic irradiation hardening in MeV-ion damaged tungsten

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The 14 MeV neutrons produced by the deuterium-tritium fusion reaction are a serious threat to the structural integrity of the tungsten plasma-facing components, which consist of a myriad of grains with various crystallographic orientations. There is evidence that the irradiation hardening behaviour depends on the grain orientation, but our current understanding of it is lacking. In this study, the (100), (110), and (111) surfaces of high-purity single-crystal tungsten specimens were respectively irradiated by 6 MeV Cu ions near room temperature to create radiation damage. Depth-resolved hardness before and after ion irradiation was measured and a strong orientation dependence was observed. Analysis of the hardness-depth profile revealed that irradiation hardening caused the deviation from the Nix-Gao model (which described well the depth dependence of the hardness before ion irradiation) after ion irradiation. The deformation microstructure underneath the indent, enabled by a combination of focused ion beam milling and transmission electron microscopy imaging, supported the observation. Based on this information, a volume fraction model-considering the influence of the damage gradient from ion irradiation and the emanating plastic zone of the indenter-was used to correlate displacements per atom with hardness changes. We found that the irradiation hardening coefficient on the (111) surface was 2.7 times that on the (100) surface, whereas the value on the (110) surface was nearly zero. The anisotropic irradiation hardening effect is mainly attributed to orientation-dependent displacement threshold energy and resolved shear stress. Our results are valuable to the understanding of deformation mechanisms of ion-irradiated materials.

Tentative

01PL1 "Presenting Author: Marta Serrano"

Cross-cutting material issues for fusion and fission nuclear systems: connection with the unified European strategic research agenda of CONNECT-NM

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Fusion applications impose very severe requirements on materials in terms of resistance to prolonged irradiation with helium production, thermal shocks, high temperature and compatibility with corrosive coolants. While in many respects specific of fusion systems, several of these issues have much in common with fission materials. This paper summarizes these issues. Next it describes the materials science research lines that are adopted in the CONNECT-NM European partnership, which are common to all nuclear materials, for fission and fusion alike, highlighting the benefit for fusion of adopting the relevant approaches and their goals. These include the development of structured test-beds for accelerated qualification of materials, as well as platforms to accelerate the development of materials that operate under harsh conditions. Another goal is the development of multi-parameter-based approaches for materials health monitoring, based on different non-destructive examination and testing techniques. Models that suitably combine physics-based and data-driven approaches can valuably support these developments, together with the creation and population of a centralised, “smart” database for nuclear materials.

Tentative

01PL2

Qualification of EUROFER97 base metal and weldments for ITER TBM: EUROfusion contribution

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In future fusion power plants (DEMO, CFETR or SPARC) the Breeding Blanket is the component in which tritium, one of the fuel elements for the fusion reaction, is generated. Various Test Blanket Module (TBM) concepts are planned for installation in ITER, where each TBM is considered as nuclear pressurized equipment according to French regulation. The structural material for the European TBM is EUROFER97, a Reduced Activation Ferritic Martensitic (RAFM) steel. Fusion for Energy (F4E) and the EUROfusion Consortium are collaboratively engaged to codify EUROFER97 for incorporation in RCC-MRx.

The scope of the tests required for codification is provided in Table 1. The task breakdown structure and implementation timeline are provided in Table 2, which include (i) mechanical properties according to existing standards; (ii) bespoke tests to validate the design rules; (iii) assessment of neutron induced ageing. In the review of the program, attention is given to the properties of EUROFER97 Batch-3 and Batch-4, used for the production of TBM mock-ups, in relation to existing data for previous batches, neutron irradiation studies and first results for the welded joints. Results from collaborative test campaigns are given for fatigue, creep and fracture mechanics properties as well as from the first neutron irradiation campaigns extending our knowledge to high-temperature range (450-550 °C). Progress in design rules validation (plastic flow instability in non-irradiated and irradiated condition and creep fatigue) is highlighted. Challenges faced during the 5 years implementation, preliminary lessons learned and further planning for the completion of high priority tests are presented.

Template

01PL3

Tritium production from thermal neutron irradiated solid breeder materials

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As nuclear fusion progresses towards becoming a viable energy source, one of the critical requirements for its success is the ability to breed tritium, a key component of the fusion fuel. One approach to achieve this is through the use of a solid breeder blanket. One of the key metrics for the performance of a solid breeder is how well it can release the bred tritium. A large body of legacy research has been conducted on potential breeder materials; however, these data are inadequate due to the inconsistent results, the lack of testing of new and novel breeder materials, and the lack of current capabilities to perform such tests. We present the design and calculations for solid breeder materials that will be irradiated in the NRAD (Neutron RADiography) reactor, a facility capable of providing controlled neutron flux conditions. Post irradiation examination will focus on the tritium release behavior of these materials at the Safety and Tritium Applied Research (STAR) facility utilizing a new tritium thermal desorption spectroscopy (T-TDS) system. The tritium release profile and release fraction will be characterized for the leading and novel candidate solid breeder materials. This study will provide a comprehensive analysis of the tritium release dynamics for a wide number of candidate solid breeder materials, thereby contributing to the development of efficient breeder blankets for future fusion reactors.

Tentative

01A1

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

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To meet the design and construction requirements of the China Fusion Engineering Test Reactor (CFETR), it is essential to develop high-temperature and radiation-resistant candidate structural materials and establish efficient evaluation methods for assessing their radiation performance. The use of small/micro-sample testing techniques to evaluate the radiation resistance of structural materials offers advantages such as reduced irradiation space, lower radioactive doses, and simplified sample preparation.

A project founded by Ministry of Science and Technology of the People's Republic of China aims to systematically investigate the mechanical properties of low-activation ferritic-martensitic steel (CLF steel) and oxide dispersion-strengthened (ODS) steel, examining their variations with sample size and the correspondence to standard samples.

The project is progressing smoothly and has yielded valuable results, summarized as follows:

Ion irradiation micro/nano mechanical testing and evaluation methods: using nanoindentation techniques to evaluate the hardening effects of materials after ion irradiation, this study reveals phenomena such as stress-affected zones, size effects, and pile-up effects, along with their changes with irradiation dose.

Standardization of small-specimen testing techniques and failure theory constitutive models: the impact of small specimen shape, size, machining methods, applicable testing equipment, and necessary fixtures on test results was analyzed. A preliminary full-process technology for small specimen mechanical property testing was established.

Standardized mechanical testing of small specimens in the hot cells: a testing device for tensile, creep, fatigue, small punch, and fracture toughness tests of small specimens was designed and constructed in a hot cell. Testing of non-irradiated samples has been conducted, and the draft of four standards has been completed.

Tentative

01A2

Development of C&S for fusion in-vessel components: from ITER to DEMO

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Invitive

In the fission industry, nuclear installations are designed and constructed following established Codes and Standards (C&S), developed over decades based on operational feedback and regulatory requirements. In contrast, fusion facilities lack a harmonized regulatory framework and their safety hazards are different from those of fission plants. Additionally, while mission-critical components - such as In-Vessel Components (IVCs) - may not be classified as safety-related and might be exempt from regulation, they must still adhere to the highest quality standards for investment protection and operational reliability.

To address these challenges, ITER adopted a multi-code approach, combining industrial codes - such as ASME and RCC-MRx - with ITER-specific design criteria and guidelines. Despite the comprehensive documentation generated by ITER, key challenges still remain due to regulatory uncertainties and extended operational domain of IVCs in future fusion power plants.

To bridge these gaps, EUROfusion, in collaboration with Fusion for Energy (F4E), is actively working on the development of C&S for DEMO IVCs. A major focus is the development of the RCC-MRx code for the EU Test Blanket Modules, which will provide critical information for the design of the Breeding Blanket. In parallel, EUROfusion is developing specific DEMO Design Criteria for In-Vessel Components (DDC-IC), introducing new design rules and alternative damage assessment methodologies to enhance performance while reducing excessive conservatism.

The DDC-IC aims to serve as a structured guide for DEMO IVC design, establishing a robust foundation for future fusion reactors. This work provides an overview of EUROfusion's efforts and the current status of DDC-IC development.

01A3

Development and validation of small specimen testing technique for ductile fracture toughness

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Availability of a qualified irradiated database on the structural materials is one of the preconditions for a successful realization of DEMO in-vessel components. The limited irradiation volume of emerging irradiation facilities with fusion relevant neutron spectra, such as IFMIF-DONES, will make the application of the Small Specimen Testing Techniques (SSTTs) indispensable. A proper accounting of specimen size effects related to the use of the SSTTs is thus required for the qualification of the data obtained with small scale specimens. The acting international testing standards, e.g., ASTM E1820, disqualify the application of the SSTT specimens for investigation of J-R curve behavior in the ductile regime due to violation of the specimen size requirement. We examined a hybrid approach combining experiments on the SSTT specimens with Finite Element (FE) simulations of ductile crack growth by using a Cohesive Zone Model (CZM). A procedure for the determination of the flow curve by tensile testing the small scale smooth round specimens as well as a methodology for the calibration of the CZM by testing and simulation of small scale notched round tensile and small scale three-point-bend specimens will be presented. Validation of the proposed hybrid approach will be done on the base of testing results obtained at RT on the reduced activation ferritic/martensitic steels EUROFER97 and F82H within the IAEA CRP and EUROfusion projects. Simulations with CZM parameters calibrated at RT enabled good prediction of the J-R curve behavior on large fracture-mechanical specimens.

Invitive

01B1

Corrosion kinetics of parent and welded F82H by LTZO ceramic breeder pebbles

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In ceramic breeding blanket, reduced activation ferritic/martensitic steel (RAFM) structural steel is corroded by vapor gas released from tritium breeder (Li-containing ceramic pebbles), resulting in a significant reduction of the fatigue lifetime. To understand the corrosion kinetics and the effect of moisture absorbed in the breeder pebbles, the corrosion test of parent and welded F82H by LTZO (Li2+xTiO3+y solid solution with 20 wt% Li_2ZrO_3) pebbles were conducted at 773-998 K. Element depth profile and structural analysis using glow discharge optical emission spectroscopy (GD-OES) and X-ray diffraction (XRD) identified the corrosion products has cubic, spinel, and rhombohedral structures with the compositions of Li-TM-O (TM: transition element in F82H). In the long-term compatibility test for up to 672 h, the growth of the corrosion layer thickness followed a parabolic law at 833 K, yielding apparent diffusion coefficient of 6.95 × 10^-13 cm^2/s. A rapid growth of the corrosion layers was observed at 993 K after 380 h which could be triggered by failure of the protective layer. The growth of the corrosion layer thickness after short compatibility tests simulating a post welding heat treatment (PWHT) followed the parabolic law in the both parent and welded F82H, while thicker layer formation was observed in the welded F82H corroded by the LTZO pebbles exposed in a wet atmosphere. The results using pebbles with different moisture absorptions indicated the corrosion of F82H during PWHT can be suppressed by removing physically and chemically absorbed water in the pebbles.

Tentative

01B2

Phase Stability and Irradiation Resistance of the Octalithium Ceramics (Li_8MO_6, M = Zr, Sn, Pb and Ce) for Tritium Breeding

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In order to most efficiently produce tritium from a high energy neutronic reaction, lithium dense tritium breeding materials (TBMs) are required. TBMs must operate under high temperatures and neutron radiation, whilst producing extractable tritium and being compatible with the surrounding materials. Ceramic TBMs offer material compatibility and do not suffer from magnetohydrodynamic (MHD) effects, however, traditionally they have lower tritium breeding ratios (TBRs) in addition to concerns over radiation damage.

With the current industrial interest in spherical tokamak arrangements with less space for TBMs, materials with higher TBRs are required. Neutronics simulations suggest that the octalithium compounds, with their high lithium densities, offer significantly higher TBRs than Li_4SiO_4 and Li_2TiO_3 which are designated for use in ITER - however most of these compounds lack basic physical data (melting points, phase stability, mechanical properties) and none have been subject to micro mechanical and ion irradiation testing.

This work presents the mechanical properties (Youngs modulus, hardness and fracture toughness) of dense octalithium ceramics (Li_8MO_6, M = Zr, Pb, Sn and Ce) from nanoindentation, how these experimental values correspond with those predicted using density functional theory modelling (DFT), and the impact of high energy (12 MeV, 1e17cm^{-2}) He ion irradiation on these properties. Further we examine how the octalithiums will perform in the hostile environment of a future reactor, by exploring the phase stability at high temperatures (500 °C, 700 °C and 900 °C) using X-ray diffraction and mass loss.

Invited

01B3

Molecular dynamics simulations of tritium diffusivity in lithium orthosilicate crystals using a machine learning potential

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Li_4SiO_4 is a promising solid tritium breeding material due to its high lithium density, thermal stability, and excellent tritium release performance. Tritium extraction involves diffusion through the bulk and desorption from the surface. Therefore, the extraction rate is controlled by both the diffusion and the desorption mechanisms. For the diffusion coefficients, there is a large discrepancy between the reported experimental values, indicating that the diffusion behavior of tritium in Li_4SiO_4 crystals needs further study.

Atomic simulations such as density functional theory (DFT) calculations and molecular dynamics (MD) calculations help to uncover diffusion mechanisms and accumulate diffusion data. In this study, to realize accurate MD simulations, we construct a machine learning potential using DFT calculation data as the training set. The moment tensor potential (MTP) is adopted.

Previous reports have shown that T^+ (positively charged tritium) is the most dominant charge state of tritium in Li_4SiO_4, so we consider tritium as T+ in our calculations. We divide the diffusion of T+ into two cases: (1) the diffusion of T+ in the perfect crystal as an interstitial atom (Dint); (2) the diffusion of T+ interacting with Li vacancy (DLi). At low temperatures (400 K - 800 K), Dint exhibits anisotropy, with faster diffusion along the Y direction, while DLi remains nearly immobile due to strong T^+-Li vacancy binding. At high temperatures (800 K - 1000 K), Dint converges to DLi because Li atoms start to diffuse significantly and T^+ is no longer deeply trapped in a Li vacancy.

Tentative

01C1

Progress and mid-term achievements of the INNUMAT project

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The Horizon Europe project INNUMAT is aimed at the development and qualification of innovative structural materials / material solutions for applications in fission lead-cooled and molten salt fast reactors as well as fusion DEMO. Co-free High entropy alloys (HEAs), a new class of materials with a vast development potential and very promising properties, as well as alumina forming austenitic (AFA) steels, already identified as prospective structural materials for Gen IV and Small Modular Reactors, are in the main focus in which advanced material solutions are considered as well, in particular weld overlay and coated 15-15Ti for lead-cooled fast reactors, among others MYRRHA and ALFRED, and coated EUROFER and advanced oxide dispersion strengthened (ODS) steel for fusion DEMO. Some of these structural materials are also potentially applicable outside the nuclear sector, e.g. in concentrated solar power and/or in H_2 confinement.

Halfway through the project, remarkable progress could be made towards its goal in rapidly increasing the technology readiness of the materials regarding corrosion resistance, high temperature strength, thermal stability and irradiation tolerance via computational and experimental high throughput material screening methods. Besides performing and evaluating respective experiments on the REFERENCE variants of all materials, predictive modeling at different scales in different areas was adapted and utilized to design and produce NEW materials, in particular HEAs and coated materials, with optimized resistance to irradiation, creep and corrosion. In addition, a pre-qualification plan has been developed paving the way toward pre-industrialization assessment and qualification roadmap for the considered new materials.

Tentative

01C2

Development of FeCrAl-based Oxide Dispersion Strengthened (FeCrAl-ODS) Alloy Free from Ageing Embrittlement

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FeCrAl-ODS alloy has been developed as an accident-tolerant fuel cladding material for light water reactors due to its high-temperature strength and steam oxidation resistance. Recently, it has been proven that this alloy is also useful as a piping material for liquid metal-cooled breeder blankets in fusion reactors. In this alloy system, appropriate control of the Cr and Al concentrations has been studied to avoid Cr-rich α' embrittlement. However, it has also been found that the Ti added to refine the oxide particles promotes β' precipitation during long-term ageing. In this study, we developed a new FeCrAl-ODS alloy free from undesirable precipitation caused by long-term ageing

The composition of the test material is Fe-12Cr-6Al-0.4Zr-0.5Y_2O_3, 0.24 Ex. O (in wt.%). Hot-extruded material and cladding tubes were prepared using the standard fabrication process of ODS alloy. These test pieces were subjected to ageing treatment at 748 K for up to 1988 hours, and microstructure observation, hardness testing and tensile testing were carried out before and after the treatment. After 744 hours of ageing,

Cr enrichment was observed at the grain boundaries, and after 1988 hours of ageing, Zr compounds of around 200 nm in size precipitated. Nevertheless, the change in Vickers hardness was less than Δ10 HV even after the 1988 hours of ageing. There was almost no change in the strength or elongation of the tensile test after ageing for 744 hours. The properties of oxide film after pre-oxidation annealing will be reported at the conference.

Tentative

01C3

Irradiation effects in the mechanical model of a full fusion tokamak

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A virtual model for a tokamak device involves concepts and algorithms spanning atomistic to macroscopic length scales and many domains of physics and engineering. At its core, a virtual tokamak model should implement the response of irradiated materials to spatially varying stress, neutrons with given energy spectrum, and temperature. An in-silico model for a tokamak prior to its construction is expected to enable significant technological developments, reducing both time and costs involved. It should enable screening reactor concepts on a scale and at pace not attainable using design-by-experiment approaches, the consideration of radiation damage at neutron energies and component sizes that are not yet accessible to experiment, and the treatment of phenomena such as the enhanced tritium retention in irradiated reactor materials. Here, we develop a finite element model for the entire UKAEA's MAST-U spherical tokamak, using 127-million finite element simulations, parallelised over hundreds of processors. Structural calculations have been performed under purely mechanical loads such as gravity or a dynamic earthquake-like excitation. The model is also coupled to highly parallelized neutron transport Monte Carlo simulations, on the same full-device scale and the same discretised geometry. To represent radiation damage effects in the form of spatially varying swelling, we parametrise a surrogate model using massively parallel atomistic simulations of tungsten, copper, and stainless steel with recoil spectra matching high-energy fusion neutrons. We explore how to transfer information from the domain of neutron transport simulations to the domain of structural calculations through the detailed analysis of fundamental radiation effects in materials.

Invited

02PL1

China's Vision for R&D of Fusion Reactor Materials at Different Stages

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This talk will present the community activities, the status and recent R&D progresses on fusion reactor materials (FRMs) in China, including the large-scale fabrication of RAFM/ODS-RAFM steels and copper alloys, advanced tungsten materials, Li-ceramics, superconducting materials, etc. The fabrication of plasma-facing components, especially divertors, superconducting magnets and modular breeding blankets will also be introduced. Recently, several testing facilities for FRMs, such as CRAFT (for magnets and divertor testing), ITER first wall-relevant high heat flux platform and small DT neutron sources, have been constructed and are now operational for FRMs testing. Additionally, about 15 dpa experimental data after fission neutron irradiation have been obtained, and the synergistic irradiation effects of fusion neutrons in FRMs have been evaluated preliminarily based on multiscale simulations and triple-beam ion irradiation. Meanwhile, the projects of high-flux fusion neutron sources have been proposed.

Considering the stages of fusion energy development in China, the service environments and the requirements for FRMs at different stages are analyzed. A phased roadmap and the strategies for the R&D of FRMs have been proposed, emphasizing material testing via BEST and HL-3 tokamaks and the involvement of manufacturers for large-scale, cost-effective fabrication of FRMs and components. Our roadmap outlines three phases: the 2020's focus on BEST/HL-3 testing, the 2030's support the Chinese DEMO reactor, and the 2040's target the Prototype Fusion Power Plant. This phased approach aims to meet the evolving demands of fusion energy development at different stages in China.

Invitation

02PL2

Advancements and Perspectives in the Development of SiC-Based Materials for Fusion Applications

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SiC-based materials, namely SiC fiber-reinforced SiC matrix (SiC/SiC) composites, show potential for structural and functional applications inside future fusion power plants, owing to their ability to operate at high temperatures and with a range of coolants/breeders, thereby enabling higher energy conversion efficiency. This paper presents recent advancements in the development of SiC-based materials within the international community, focusing on processing techniques and research related to their resistance in fusion-relevant environments. Recent processing activities have focused on near-net shaping, improving the composite's mechanical properties by incorporating high-strength SiC fibers, and developing a pressure-less processing method to form ceramic-based, leak-tight joints. Neutron irradiation research continued with the High Flux Isotope Reactor (HFIR). Post-irradiation examinations were extended to include statistical analysis of the mechanical properties after irradiation at intermediate temperatures (e.g., ~600 °C) to high neutron damage levels exceeding 30 dpa, emphasizing the importance of statistical reliability in the mechanical properties to expand the design margin. A synergistic irradiation experiment for a fission application investigated the dimensional stability of a SiC/SiC composite channel structure with dimensions of 30 mm × 30 mm × 380 mm and a wall thickness of 1.25 mm in the HFIR. This experiment provides insights into the radiation stability of a flow channel insert made of SiC/SiC composite. Remaining critical knowledge gaps include the effects of gaseous (e.g., He) and metallic (e.g., Mg and Al) transmutations on the bulk thermo-physical properties. Current efforts to address these gaps will also be presented.

Tentative

02PL3

Development and Irradiation Effects of Welds and Joints in Reduced-Activation Ferritic/Martensitic Steel F82H

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The low-activation ferritic/martensitic (RAFM) steel used in the blanket of the fusion DEMO reactor is, in principle, manufactured by welding due to the complexity of its structure. In the ITER-TBM in JA, only tungsten inert gas (TIG) welding is used, but in the fusion DEMO reactor, laser and electron beam (EB) welding techniques are also being considered, and advanced welding/joining technologies are being developed. The operation environment for these welds and joints is designed to minimize neutron irradiation caused by fusion reactions, but it is not possible to completely avoid it, and they will always be irradiated. Therefore, the integrity of the joints after irradiation is an important issue. In this study, we conducted irradiation tests at HFIR up to 50 dpa with the aim of clarifying the irradiation resistance of TIG and EB welds for Japanese RAFM, F82H. Note that weld metal (WM) and heat-affected zone (HAZ) themselves were selectively irradiated especially for TIG weld materials. The initial test results suggest that WM and HAZ undergo hardening and loss of ductility to the base metal level in tensile tests with miniature SS-J3 specimens irradiated to 50 dpa at 300 °C. In this paper, irradiation data at varied temperatures and to different dose levels will be summarized. In addition, we will also report the results of screening evaluations using ion accelerators for advanced joining technologies such as linear friction welding (LFW), eventually clarifying the prospects and challenges of F82H welding/joining toward the DEMO design.

Invited

02A1

Retention and release of deuterium in displacement-damaged and helium-implanted EUROFER97 at reactor-operating temperatures

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One of the principal challenges is understanding radiation damage effects on hydrogen isotope (HI) retention, particularly concerning safety and tritium self-sufficiency. Lattice defects and helium (He) bubbles induced by high-energy neutrons in plasma-facing components act as HI trapping sites, significantly affecting HI transport and retention.

This study investigates deuterium (D) retention in displacement-damaged and He-implanted EUROFER97. The objective is to characterize He-related trap sites for D and study their thermal evolution. EUROFER97 samples are irradiated with 20 MeV tungsten ions at temperatures ranging from 290 to 770 K, inducing a primary peak damage dose of 0.6 dpa. Selected samples are additionally implanted by 0.5 MeV He ions to a peak concentration of 0.6 at.% to form He bubbles. These samples are then exposed to D plasma to fill the radiation-induced defects with D. Nuclear Reaction Analysis (^{3}He-NRA) is applied to determine D depth profiles. Thermal Desorption Spectroscopy (TDS) is employed to investigate D desorption.

The presence of He bubbles, characterized by transmission electron microscopy, significantly increases the local D concentration. He-related traps decrease in density with increasing temperature but they are not completely recovered even after annealing at 720 K, unlike displacement-damage without He that recovers at 620 K. Irradiation at reactor-operating temperatures leads to lower D retention compared to annealing at the same temperatures after irradiation at 290 K. D is de-trapped from He-related defects at about 520 K. Simulations of TDS spectra utilizing a macroscopic rate equation code (TESSIM-X) estimate the D de-trapping energy from He-related defects.

Tentative

02A2

Detritiation in tungsten and its oxides: Prediction from multi-scale materials modelling and engineering implication

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Invitation

A fundamental understanding of tritium permeation and retention behaviour in fusion energy components is essential for efficient tritium recovery, waste classification and performance of in-vessel materials under neutron irradiation. Tungsten is one of the most promising materials for plasma-facing components for future fusion devices such as STEP and DEMO. Tungsten used in JET has been exposed to D-T campaigns and is expected to have high tritium inventory. Tungsten oxidises readily at temperature above 500 °C and its oxidation is a potential concern for maintenance, accidents, tritium recovery and waste handling scenarios. There is little literature investigating the detritiation of tungsten under these fusion-specific conditions.

In this work, a systematic multiscale materials modelling investigation has been undertaken to predict the microstructural properties of tungsten and its oxides relevant to tritium retention and transport. Using a combination of density functional theory, machine-learned potentials and phase field methods, our results reveal that tritium permeability is low in WO_2, intermediate in W, and relatively high in WO_3 [1]. Tritium transport and prospects for efficient tritium recovery were shown to depend on W/oxide interface topology, oxide microstructure, crack nucleation and growth processes in the W oxidation film. A dynamically metastable WO_2 -rich barrier layer could dramatically influence tritium retention and efficient tritium recovery. The current modelling results are assessed and compared with experimental data using tritiated W samples at the UKAEA to understand the detritiation behaviour of WOx films grown under different exposure conditions.

[1] M. Christensen et al., Nuclear Materials and Energy, 38, 101611 (2024)

02A3

Influence of reproducible trapping consideration and W-RAFM bonding region on hydrogen isotope transport behavior in the first wall

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The hydrogen isotope transport behavior (HITB) in the first wall (FW) of WCCB blanket has been extensively studied through simulations. These models typically use effective diffusion parameters and incorporate with trapping effect of the RAFM steel which is designed to bond with tungsten (W) containing a W-RAFM steel bonding region. However, the effective diffusion coefficient is acknowledged to include the information of traps and the W-RAFM steel bonding region will inevitably affect the HITB. Neglecting these two issues would generate inaccurate predictions of HITB. This work investigates the HITB in the FW with the consideration of (1) the repeated adoption of trapping and (2) W-RAFM steel bonding region through a combined experimental and simulation approach. The diffusivity and solubility parameters were obtained by GDP experiments and FESTIM was applied to simulate the HITB of the FW. Two diffusion coefficients, optimized diffusion coefficient (D1) and effective diffusion coefficient (D2), are considered for RAFM steel. An error function is employed to determine the unknown D1. Compare the total amount of retention of FW with D1 and D2 alongside three traps respectively, indicating that the reproducible consideration of trapping effect will result in an overestimated prediction. Furthermore, the effect of W-RAFM steel bonding region on HITB of FW has been discussed. The retention in the bonding region is one order of magnitude higher than in the adjacent bulk materials. Taking the two neglected issues into account, an optimized prediction model for HITB in the FW of WCCB blanket is provided.

Invitation

02A4

Mass production of boron-containing steels for fusion shielding application

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Fusion devices require high-performance shielding materials to protect superconducting magnets from nuclear heat deposition and radiation damage caused by neutrons and gamma rays. Currently, ITER's In-Vessel Shielding (IWS) relies on 304B4 and 304B7 boron steels as the primary shielding materials. However, the embrittlement and cracking issues associated with the processing and forming of high-boron steels require the use of smaller plates for these steels in IWS designs, while larger forgings used for the shielding blanket are produced from steel ingots. To address the need for high-shielding boron steel in fusion devices and overcome the challenges of processing-related embrittlement, this study explores the development of 304B7 high-boron steel for fusion application through two technological approaches: powder metallurgy and melting methods. The preparation process is optimized to ensure a uniform distribution of boron phases, thereby enhancing mechanical properties and facilitating mass production. Furthermore, to integrate both structural and functional characteristics in shielding blanket forgings, a new tungsten-boron steel, which offers superior shielding performance compared to 304B7, is developed, leading to significant improvements in mechanical properties. A 9-ton tungsten-boron steel forging is achieved, and the processing capabilities of the forging-such as deep-hole drilling and welding performance-are systematically evaluated. This advancement paves the way for the manufacture of ton-scale shielding blankets, significantly improving the shielding performance of fusion devices.

Tentative

02A5

Development of Grey Cryo-Steels-Combining Structural and Shield Function to Enable Compact Fusion Design

Lance Snead[1], Abhinav Sharma[1], Mirza Shawan[1], Ethan Peterson[2], David Sprouster[1]

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Recent advancements in high-field, high-temperature superconductors has enabled design of more compact, economic, and higher power density fusion systems. However, compacting these systems result in proportionally greater radiation fields with less physical space for functional systems such as shielding. This suggests the need for innovative solutions to optimize next generation fusion systems, as example for magnet shielding.

Here we describe an approach to improved system shielding by incorporating shielding functions within structural alloys. A composited cryo-steel has been developed incorporating a high fraction of metal hydride (MH). The example MH, HfH2, simultaneously provides neutron moderation, absorption as well as being an effective absorber of electromagnetic radiation. Ideally, such a shield requires access to processing windows that minimize MH decomposition while achieving high MH loading and good adequate mechanical properties. Here we describe grey cryo-steel consisting of 25v%, 35v%, and 45v% HfH2, entrained within a (typical model) cryogenic 304L matrix steel. Relatively low-temperature processing (99%) compact densities of uniform microstructures, as determined with physical measurement supported by X-ray diffraction/tomography, and electron microscopy. In addition to processing and microstructure we present the mechanical performance (tensile properties for as fabricated, rolled, annealed) of the grey cryo-steel from -196 °C to ambient. Finally, beyond presenting microstructure and properties development this paper will present a neutronics analysis of the impact of this grey cyro-steel on compact fusion systems, specifically enhanced magnet life.

Tentative

02B1

Evolution of Vacancy Cluster, Precipitates, and Hardening in Binary Tungsten Alloys after Irradiation at Elevated Temperatures

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Tungsten (W), as plasma-facing material, will transmute into rhenium (Re) by neutron irradiation in a future fusion reactor. In previous studies, the reduction of deuterium (D) retention and precipitation of Re has been observed in W-Re alloys after neutron or ion irradiation. According to the simulation results, the Re atom reduces the mobility of the W self-interstitial atom (SIA) by forming a mixed dumbbell and enhances the chance of recombination with a vacancy. To clarify the impact of the binding energy with a W-SIA on the evolution of defects and irradiation hardening, a comprehensive investigation of D retention, positron lifetime measurements, microstructure analysis, and nanoindentation tests were performed.

Plates of W, W-5Re, W-0.3Cr, W-2.5Mo and W-5Ta alloys were irradiated at 523-1273 K with 6.4 MeV Fe ions to 0.26 dpa at the damage peak in the accelerator DuET, Kyoto University. The results of D retention and positron lifetime measurements showed Ta and Mo had no noticeable suppression effect on vacancy formation in W, while significant suppression was observed for Re and Cr. Irradiation-induced precipitation was found in W-0.3Cr alloy using APT and STEM measurements. These observations suggest that an additive element with strong binding energy with a W-SIA effectively suppresses vacancy formation and has a stronger tendency to precipitation under irradiation. On the other hand, no systematic correlation was found between irradiation hardening and binding energy with a W-SIA; W-0.3Cr alloy showed significant hardening while the degree of hardening of W-5Re alloy was the lowest among all these materials.

Tentative

02B2

Mechanical properties of nanoporous tungsten under irradiation - modelling and experiments

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Nanoporous metals (np-metals) are a class of functional materials that exhibit a bi-continuous network of nanoscale pores and solid ligaments, forming an open-cell porous material. They offer improved mechanical properties, enabled by nanoscale effects. In particular, refractory np-metals have a remarkable potential for nuclear applications. These np-metals combine the excellent mechanical and radiation performance of the base material with an improved tolerance to radiation, thanks to the presence of a large amount of internal surfaces, which act as sinks for radiation-induced defects. In this contribution, nanoporous W with ~35% porosity and average ligament size of 20-25 nm was manufactured using severe plastic deformation of a coarse-grained W-Cu composite followed by selective dissolution of Cu. The resulting np-W was then implanted with H and He at energies of 82 keV and 160 keV, respectively. The implantation fluence for both cases was selected to be 5 x 10^{20} m^{-2}.

Molecular dynamics simulations of radiation damage and FFT-based crystal plasticity simulations were combined to predict the effect of H and He implantation on np-W mechanical properties. The simulations were then validated by nanoindentation and micropillar compression experiments on as-prepared and irradiated samples. Hardness and yield stress increase due to irradiation, and the results are interpreted based on a model accounting for the increased resistance to dislocation glide by helium-implantation-induced defects.

Template

02B3

Hydrogen-Stabilized Dislocation Interactions Drive Blister Formation in Tungsten

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Achieving tritium self-sufficiency is a critical milestone on the path to practical fusion energy. Hydrogen isotope charging in plasma-facing tungsten components is known to induce blistering, which significantly increases tritium retention and jeopardizes self-sufficiency. Despite extensive research, the atomistic mechanisms underlying hydrogen blister formation remain poorly understood, particularly for intragranular blisters observed under sub-threshold hydrogen plasma. Here, we identify the nucleation pathway of such hydrogen-induced blistering. We reveal that hydrogen promotes the interaction between two /2 screw dislocations, facilitating the formation of a edge dislocation junction, which is energetically unfavorable in the absence of hydrogen. This dislocation exhibits exceptional stability and evolves into a vacancy-type dislocation loop under applied strain. The accumulation of these loops during ongoing deformation and dislocation reactions leads to bulk fracture along {001} planes, culminating in hydrogen blister formation. These findings offer atomistic insights into recent experimental observations and advance our understanding of sub-threshold hydrogen-induced blistering in tungsten, a crucial challenge for fusion materials design.

Tentative

02B4

Experimental investigation of the flux-dependence of He clustering in polycrystalline and monocrystalline W

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Ab-initio modeling of helium (He) behavior in tungsten (W) predicts clustering mechanisms, self-trapping and trap-mutation, but to date there is little experimental evidence for isolated He self-trapping in a defect-free lattice. In order to investigate these mechanisms and the influence of intrinsic defects in W on them experimentally, polycrystalline (poly-W) and monocrystalline (sc-W) W samples were probed for open-volume defects by positron annihilation spectroscopy and then exposed to low-temperature He plasma at 292 K surface temperature. A He energy of 50 eV was chosen to avoid kinetic defect introduction. The flux was varied between 10^{17} and 10^{19} He/m^{2}s and the fluence ranged from 10^{19} to 10^{23} He/m^2. Elastic recoil detection analysis (ERDA) was used in combination with an electrochemical stripping method to derive absolute He areal densities and He depth profiles.

The sc-W data show a flux-dependent threshold behavior: While there is no measurable He signal at lower fluxes, significant He retention is observed at a flux of 1×10^{19} He/m^{2}s. This flux-dependence indicates that self-trapping dominates the He retention. Exposed with the same parameters, the poly-W retention data exhibit a threshold at much lower fluxes. A comparison of He retention after low-flux exposure of electron-beam irradiated sc-W with undamaged sc-W and poly-W eliminates pre-existing open-volume defects as the cause for the differences in He interaction between the undamaged materials. Thus, impurities emerge as the most plausible promoters for He clustering mechanisms and trapping in poly-W, which is supported by the differences in He depth profiles found between poly-W and sc-W.

Tentative

02B5

The effect of grain boundary on hydrogen isotopes diffusion in tungsten

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Tungsten (W) has been chosen as the candidate material for ITER first-wall and divertor. And the transport of hydrogen isotopes (HIs) in plasma facing materials remains an important issue. The grain boundary (GB), as a type of defect of relatively high volumetric density ratio, was found to have influence on HIs diffusion. In this work, single crystalline W, ITER-grade commercial polycrystalline W (ATW) and high rolling deformation W plate were selected to study the effect of GBs on HIs diffusivities, through gas-driven permeation (GDP) method. In detail, for ATW, the rolling direction (RD), transverse direction (TD) and normal direction (ND) were investigated separately. The applied deuterium pressure was about 1E5 Pa and the temperature range covered 923 K to 1123 K. Results indicated that the effect of GB is significant when temperature lower than 1023 K. Besides, electron backscatter diffraction (EBSD) characterization was utilized to analyze the influence of high-angle grain boundaries (HAGB) and low angle grain boundaries (LAGB) on deuterium diffusion. Furthermore, to figure out the micro-scale mechanisms of GBs impacting diffusion, molecular dynamics (MD) simulations were launched in a series of GBs. Both the influence of hydrogen binding energy to GBs and GBs density on diffusion have been elucidated. This study gives a view to the anisotropy diffusion characteristic of HIs in ITER-grade W, which might be taken into consideration for the evaluation in the field of fuel recycling and radiation safety.

Tentative

02C1

Leveraging the Fusion ENergy Integrated multiphys-X (FENIX) Computational Tool for Fusion Materials Research

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Tentative

Addressing the materials development needs of the accelerating fusion energy deployment timeline requires a deep understanding of the interaction between materials and their harsh environment. Plasma facing materials, for example, are subject to extreme thermal loads, repeated thermal shocks, and bombardment by 14 MeV neutrons, plasma ions, and neutral particles (deuterium, tritium, and helium), whose interactions with material performance are complex. Advanced multiphysics modeling and simulation capabilities offer great potential for accelerated material studies through a high-fidelity, multiscale approach resolving these convoluted interactions. The open-source Fusion ENergy Integrated multiphys-X (FENIX) framework addresses these needs by leveraging the Multiphysics Object-Oriented Simulation Environment (MOOSE) framework to model the environment and response of plasma facing materials. FENIX integrates various MOOSE capabilities, including the heat transfer, solid mechanics, and thermal hydraulics modules, along with MOOSE-based applications such as Cardinal (neutronics and computational fluid dynamics) and TMAP8 (Tritium Migration Analysis Program, version 8). FENIX also supports dedicated plasma edge modeling capabilities for plasma-material interaction simulations. By capturing these physics, FENIX can accurately predict the fusion environment and simulate material performance.

We will present FENIX's current capabilities and an example case for a divertor monoblock focusing on interactions between its environment and material performance. This example will highlight FENIX's modular design and how it enables collaborations for massively parallel multiphysics simulations. FENIX is being developed following the INL Software Quality Assurance plan PLN-4005, ensuring rigorous verification, continuous integration and adequacy for purpose. These features support FENIX in advancing fusion materials research, design, and performance evaluation.

02C2

Perspectives and challenges of ultra-high temperature ceramics for fusion plasma-facing applications

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A major challenge in fusion energy is developing materials for plasma-facing components (PFCs) capable of withstanding the extreme conditions of proposed fusion reactors. This presentation explores the potential of ultra-high temperature ceramics (UHTCs) as alternative plasma-facing materials (PFMs), emphasizing their advantages and challenges for enabling breakthroughs for PFCs. UHTCs offer several potential advantages as PFCs due to their extreme melting points, tailorable thermal conductivity, and attractive unirradiated mechanical properties at elevated temperatures, including fracture toughness comparable to or superior to tungsten. However, the effects of neutron irradiation and fusion plasma exposure on UHTCs remain poorly understood. This work examined the effects of neutron irradiation on dimensional changes, lattice swelling, microstructures, and thermal properties of ^{11}B-enriched TiB_2, ZrB_2, and TiC irradiated at HFIR, ORNL, to 5-10 dpa at 200-1000 °C. Building on these results and a review of UHTC properties, along with review of limited studies on their responses to neutron irradiation and plasma-surface interactions, five key research pathways for advancing UHTCs in PFC applications will be discussed: (1) assessing irradiation effects on coupled thermal-mechanical performance; (2) addressing gaps in research on irradiation, plasma exposure, and their synergy; (3) studying high-temperature neutron irradiation effects; (4) optimizing multi-component UHTCs or composites to enhance material properties; and (5) improving radiation resistance.

Tentative

02C3

Thermal ageing study of liquid metal based plasma facing material: tungsten capillary porous system wetted by tin

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Most of the existing reactor-class devices propose to operate tungsten as plasma facing material (PFM). Due to complex in-vessel environment tungsten lifetime is constrained by several aspects: neutron embrittlement, dust generation, local melting, cracking, …. This motivates alternative PFM solution developments in the prospects of high availability nuclear fusion power plant. In this context, liquid metal based divertors (LMD) development could enable operation while reducing target lifetime issues through liquid replenishment. One approach to develop a LMD aims at developing in-vessel components using Capillary Porous System (CPS), wetted by a low melting point metal. This concept demonstrated its reliability in a semi-inertial use during the FT-U operation, notably under off-normal events, thus promoting CPS concept development as sacrificial targets or as first wall/divertor components.

To increase our understanding concerning the thermal ageing of such technology, a tungsten CPS sample was thus produced and wetted with tin at ENEA, then tested under high heat flux in the e-beam facility HADES (CEA-Cadarache). A total of 40 thermal cycles imitating either stationary (1.3 MW/m^2, 2 s On) or transient heat loading (46 MJ/m^2, 0.6 s On) was performed. Although tin evaporation and splashing occurred resulting of a total weight lost equal to 0.77 g, this study shows that CPS sample survived with no failure. A comparison was also made between pre & post ageing experiments using optical microscopy, SEM and pycnometry. These analyses emphasize large deformation of the CPS mesh which should be reduced and/or optimized in the perspective of in-vessel components using CPS.

Tentative

02C4

Detailed analysis of gas retention behavior of bubbles in fusion reactor materials using in-situ TEM and STEM-EELS

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It is well known in many metals that helium atoms introduced by bombardment of helium ions or (n,α) nuclear reaction generally precipitate as helium bubbles, causing a significant effect on hydrogen isotope retention behavior. However, it is still unclear where, how, and how much helium and hydrogen isotope exist in metals with bubbles. In this study, we demonstrated the gas retention behavior of bubbles in several fusion reactor materials such as tungsten, beryllium and reduced-activation ferritic/martensitic steel (F82H) using two types of high function TEM: in-situ ion irradiation TEM with a high-resolution quadrupole mass spectrometer, and an aberration-corrected scanning TEM combined with electron energy-loss spectroscopy (STEM-EELS).

In-situ TEM enabled quantitative analysis of gas species released from a 3 mm diameter samples and simultaneous observation of microstructural evolution under annealing. For instance, in tungsten sequentially irradiated with helium and deuterium, it was shown that deuterium is desorbed at relatively low temperatures without any apparent change, whereas helium is desorbed at high temperatures accompanied by bubble growth and migration. STEM-EELS analysis also revealed the locations of helium and deuterium atoms and their densities within individual bubbles. While most of the deuterium is localized in the bubble for tungsten and beryllium, only a small amount of deuterium is present in the bubble for F82H.

Quantitative data on hydrogen and helium retention will be presented and discussed in the conference. The results clearly indicate the direct impact of helium bubbles on the hydrogen isotope retention behavior in metals.

Invitation

02C5

Low-Z materials coating to enhance long pulse H mode plasma performance in EAST

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Tentative

Low-Z materials are widely employed for wall conditioning to enhance plasma performance in fusion devices. In EAST, lithium (Li) evaporative coatings significantly reduced both the impurity concentration and fuel recycling in plasmas [1]. However, it also demonstrated routine Li coating has short lifetime about 300 s. Therefore, real-time wall conditioning via feedback Li powder injection was used to further control fuel recycling and reduce the core high-Z metal impurities during >1000 s long pulse H-mode discharge in 2025. In addition, for supporting ITER new baseline, a series of boronizations were also carried out in EAST. It was found that the impurity radiation including oxygen and heavy impurities decreased significantly, which results in the slightly increased plasma stored energy. The lifetime of boronization was about 1700 s [2]. Moreover, boron (B) powder was injected into plasma for real time B coating[4], and the W impurity content could be decreased to 10-5.

Compared to Li coating, the B coating has lower H trapping capacity [5] and a plasma confinement degradation of >15%. These results provide a very valuable reference for evaluating low Z materials application in ITER and future fusion reactor devices.

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02D1

Swelling resistance in reduced activation ODS medium-entropy alloys under triple ion irradiation

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Swelling resistance of high/medium entropy alloys (H/MEAs) is believed to be due to frequent I-V recombination, as severe lattice distortion suppresses the migration of interstitial-type clusters. However, the synergistic effect of transmutation gas atom and point defect on swelling could be an issue in H/MEAs. In this study, the swelling in oxide dispersion strengthened (ODS) MEA was investigated under triple ion irradiation to clarify the sink effect at the particle interface in MEA.

FCC-type MEA Cr0.8FeMn1.3Ni1.3 powder powder, 0.5 wt% Y_2O_3 powder, and 0.2 wt% Ti powder were mechanically alloyed with a planetary ball mill and then consolidated at 1000 °C by spark plasma sintering method. Triple ion irradiation (10.5 MeV Fe^{3+}, 15 appmHe/dpa, 40 appmH/dpa) at 500 °C was performed up to 20 dpa to investigate cavity formation. Comparisons were made between specimens without Y_2O_3 (0Y), with Y_2O_3 (0.5Y), and with Y_2O_3 and Ti (0.5Y0.2T).

Nano oxides distributed in 0.5Y0.2T were smaller in size and higher in number density than those of 0.5Y, which was attributed to the formation of Y-Ti complex oxides as indicated by elemental analysis. Cavities were formed in matrix and near the oxide particles, and detailed analysis revealed that cavity size tended to correlate with the size of adjacent particles. As a result, 0.5Y0.2T showed the smallest swelling. These results indicate that increasing the number of sink sites, i.e. oxide/matrix interface, can be effective to suppress cavity swelling in MEA matrix as well as traditional ODS alloys, under a fusion relevant irradiation condition.

Tentative

02D2

^{11}B isotope enhanced strength-ductility synergy in ODS steel

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The strength-ductility synergy is one of the major issues concerning the application of ODS steel in nuclear reactors. Here we developed a ^{11}B doped 13Cr ODS steel with outstanding strength-ductility synergy via mechanical alloying and subsequent spark plasma sintering. A combination of high UTS up to 887.3 MPa and total elongation of 21.7% was obtained in ^{11}B doped 13Cr ODS steel. 60 ppm 11B was added into the 13Cr ODS steel that achieved an improvement of total elongation by over 40% at room temperature compared to its counterpart without ^{11}B addition. The atomic scale high-resolution characterization techniques including nano secondary ion mass spectrometry, atom probe tomography and transmission electron microscopy revealed that 11B was concentrated within the interior of the nano-oxides, which played a key role in accommodating the strain gradient and sustaining a low strain-hardening rate during plastic deformation by the tunable pinning effect of ^{11}B-containing particles to dislocations. This work provides a simple yet substantially effective new pathway for designing ODS steel with superior strength-ductility synergy.

Tentative

02D3

A novel RAFM steel with superior high-temperature creep lifetime prepared by Additive manufacturing

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The structural materials for fusion reactor blanket face extremely harsh service conditions. Reduced activation ferrite/martensite (RAFM) steel strengthened by MX phase or oxide dispersion strengthened (ODS) phase is the preferred blanket material. The MX phases generally precipitate from matrix and form a coherent interface that showed high stability under irradiation. However, MX-reinforced RAFM steels (e.g., CNAs), still fail to meet the high-temperature service requirement. ODS-RAFM steels have longer high-temperature creep lifetime, but nano-oxides larger than 20 nm may suffer from structural deterioration and chemical changes under neutron irradiation, leading to performance decline. New RAFM steel that can meet both service conditions and has good processability needs to be developed.

In our previous work, ODS-RAFM steel had been successfully prepared by a special LPBF strategy with oxygen atmosphere, which has comparable high-temperature mechanical properties to those prepared by powder metallurgy (PM). In this study, we further developed this technology, in conjunction with compositional design of key elements like Ti, C, and Y, fabricating superior LPBF-RAFM steel. With nano-dispersoid that had the size of 2-10 nanometer and number density as $10^{23}/m^3$, the high-temperature creep lifetime of this steel achieved 210 h at 160 MPa and 650 °C, much higher than the ODS-Eurofer97 prepared by PM (29 h). Further characterization indicated that this nano-dispersoid was neither MX nor Oxides, and it was in good lattice agreement with the matrix, which means this LPBF-RAFM steel can not only possess outstanding thermal mechanical performance, but also have the potential to exhibit high neutron irradiation stability.

Invitation

02D4 "Presenting Author: Jiming Chen"

Fabrication of ODS steel via a Z type Homogenized Blending (ZHB) route

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This work reports an attempt of fabricating ODS steel for large scale via a new route. We used rolling on RAFM-Yttrium composites followed by stacking and hot isostatic pressing to bond them, after repeating such operation several times and final heat-treatment, we obtain the ODS steel. This is an accumulative-roll-bonding (ARB)-like route, but Y content so low (~0.5%) that the control of uniform Y distribution is dramatically different from ARB, so we call it Z type Homogenized Blending (ZHB) process. Where, “Z” means stacking structure and repeating processes.

With above ZHB route, the as received ODS steel has the tensile strength of ~1.4 GPa and total elongation of ~14%. After a final annealing at 650 °C for 1hr, the work hardening is largely eliminated, and the material still keeps a tensile strength of ~1.0 GPa and total elongation of ~14%. These properties are nearly comparable to traditional ODS steel via mechanical alloying, let alone the grain sizes of traditional ODS steel are comparatively smaller. Moreover, micro-structural observation shows almost no Y segregation, indicating a uniform distribution of Y in the alloy.

In contrast, when similar accumulative-roll-bonding was applied for RAFM steel and Y foils, after final annealing at 650 °C, its tensile strength falls down to around the value of original RAFM steel. This indicates a potential advantage of ZHB route for fabricating ODS steeling in large scale.

Template

02D5

Investigation of in-situ nano oxide formation in ODS F/M steels produced by L-DED

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Oxide dispersion strengthened ferritic/martensitic (ODS F/M) steels are considered as a candidate structural material in fusion reactors because of good thermal and mechanical properties and superior neutron irradiation resistance. Laser-directed energy deposition (L-DED) is regarded as a promising process for producing ODS F/M steels due to its short process time and to create near-net shape components. ODS F/M steel powders are conventionally fabricated with ball milling to disperse nano yttria in F/M steel powder. However, nano yttria inside the ODS powders are frequently agglomerated during laser-directed energy deposition (L-DED) process. Recently, dissolving yttrium during the gas atomization process is an alternative way to produce ODS powders but yttrium is highly reactive with ceramic crucible. In this study, ODS F/M steel powder was produced by electrode induction melting gas atomization (EIGA) process without using crucible. Also, Ar and O_2 mixed co-axial gas was used to form nano oxides during L-DED. The microstructure of powder and as-built samples were observed using a scanning electron microscope (SEM) and SEM-energy dispersive X-ray spectroscopy (EDS). The size distribution and number density of yttria were calculated with transmission electron microscope (TEM) and small angle neutron scattering (SANS). The strength and elongation were calculated using uniaxial tensile test with strain rate 10^{-3} /s.

Tentative

02D6

Structural material development and down-selection process for in-vessel components within the European fusion materials program

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Material development for future fusion devices has a decades-long history in Europe, initiated under EURATOM through the EFDA collaborative agreement and further developed in the framework of the EUROfusion Consortium. One particular topic thereby is the development, characterization and upscaling of advanced structural materials, which should allow extending the operational regime and the lifetime in particular of the breeding blanket with regard to the baseline material EUROFER97.

The progress thereby is monitored via assessment of the material technology readiness level of the materials for low and high temperature cooled blanket designs, which also includes the identification of showstoppers. In this down-selection process a variety of material data with particular focus on the performance under neutron irradiation is required to take a sound decision.

Based on basic characterization and results of several neutron irradiation campaigns, the portfolio of low temperature steels has been reduced to one with a thermo-mechanical treatment at high temperature and lower austenitization temperature with the option for a second relying on cold working and subsequent recrystallization. For the high temperature design the focus is on a steel with high tungsten and isotopically doped boron content with a second alternative resulting from a model alloy search algorithm. In addition. ODS alloys with varying Cr content, to tackle both application as surface plating as well as fabrication of the full First Wall component are developed with significant progress in the industrial upscaling of mechanical alloying. Herein, the rationale for the down-selection will be explained and next steps outlined.

Invited

02E1

Toward emulation of neutron damage with ions: effects of helium on microstructural evolution in ion-irradiated V-4Cr-4Ti alloy (Heat US832864)

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Vanadium-base alloys are promising candidate structural materials for modern commercial-scale fusion power plant designs due to their low activation, high-temperature strength (>600 °C) and resistance to swelling. However, concerns about lifetime dimensional and structural integrity remain unresolved, particularly radiation embrittlement and loss of ductility below 400 °C, and high-temperature helium embrittlement. A critical challenge in fusion materials is replicating neutron-irradiated microstructures using ion irradiation to rapidly qualify structural materials. A precise neutron-ion “handshake” (microstructural match) remains elusive; complex helium and hydrogen interactions during irradiation and heat-to-heat variability between studies introduce additional uncertainty in elucidating key mechanisms driving microstructural evolution from neutrons and ions.

This study investigates effects of helium on microstructural evolution of V-4Cr-4Ti (General Atomics heat US832864) under ion irradiation. Microstructures resulting from varying helium/hydrogen injection rates, and multiple permutations of helium/hydrogen/ion irradiation steps (multi-stage sequential gas/ion steps, dual/triple beam) are evaluated, discussed, and compared with historic neutron irradiations of the same heat. Resultant microstructures are characterized using scanning and transmission electron microscopy. Real-time elastic property changes are measured using in-situ ion irradiation transient grating spectroscopy (I3TGS) to characterize the onset of void swelling, when observed. These findings contribute toward a systematic approach for emulating neutron damage with ion irradiation, informing future efforts to establish ion-based screening methodologies for rapid development and qualification of fusion materials.

Work supported by Commonwealth Fusion Systems. Vanadium alloys (heats US832864 and US832665) for this work were procured and furnished by the United States Department of Energy, Office of Fusion Energy Sciences.

Invitation

02E2

High-Dose Irradiation Response of SiC Fibers

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The irradiation stability of SiC/SiC composites is generally excellent, making them an attractive structural material for fusion applications. However, at intermediate and low temperatures under high doses, significant strength degradation has recently been observed following neutron irradiation. This could be due to the smaller swelling in the SiC fibers compared to the SiC matrix, leading to strain mismatch and interfacial debonding. Unlike high-purity SiC matrix that expand under irradiation, fibers, which contain minor impurities, could also contract; however, the underlying mechanisms remain unclear. This study aims to elucidate the mechanisms mitigating swelling in SiC fibers.

Highly crystalline SiC fibers (Hi-Nicalon type S, Tyranno SA3rd) were irradiated with 5.1 MeV Si-ions at either RT or 300 °C up to 100 dpa, followed by dimensional and strength measurements as well as microstructural analysis.

The results showed that room-temperature irradiation induced progressive amorphization and significant strength degradation in both fiber types. The amorphization threshold damage level of SiC fibers differed not only from that of the matrix but also between fiber types. In contrast, no amorphization occurred at 300 °C, where the fibers exhibited expansion rather than contraction, though the degree of expansion was smaller compared to SiC matrix. HRTEM revealed that, in fibers irradiated at 300 °C, carbon packets dissolved into the SiC grains, where shorter π bonds replaced the typical σ bonds predominantly found in unirradiated SiC. This suggests that the excessive formation of CSi antisites with shorter bonds occurs within SiC fibers and may be one of the key mechanisms responsible for swelling mitigation.

Template

02E3

Mechanistic exploration of insulation materials low-temperature irradiation damage through multiscale simulation coupled with experimental validation

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The failure of insulation materials in fusion reactors can directly lead to the quenching of the superconducting magnet system, which poses significant risks to operational safety. To replicate real operating conditions, low-temperature γ-ray irradiation platforms at 77 K and 4.2 K are designed and established, enabling successful γ-ray irradiation experiments at low temperatures. The experimental findings are corroborated through Density Functional Theory (DFT). At an irradiation dose of 5 kGy, the concentration of free radicals in polytetrafluoroethylene (PTFE) irradiated at 77 K is found to be 29% of that observed at room temperature. At 30 kGy, the free radical concentration increases to 128%, and the molecular weight of PTFE decreases by 45.5%. These results demonstrate that as the irradiation dose increases, PTFE undergoes a "cleavage - crosslinking" process. At low temperatures, bond cleavage and free radical motion are inhibited, preventing crosslinking and resulting in a high concentration of short-chain free radicals in the material. DFT calculations indicate that the reaction enthalpy change for C-C bonds in PTFE increases by 20% at 77 K compared to room temperature. The presence of free radicals generates singly occupied molecular orbitals (SOMO) with an energy of -0.2852 eV, which facilitates charge migration. This results in an 18% reduction in the mean elastic modulus and an 8.9% decrease in the dielectric constant of PTFE under 15 kGy low-temperature irradiation. These findings suggest that investigating low-temperature γ-ray irradiation damage mechanisms is essential for accurately guiding the development of new insulation materials for fusion reactors.

Tentative

02E4

Impact of High Neutron Fluence on Mechanical Properties of FBG Sensors for Nuclear Applications

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Tentative

The increasing use of fiber Bragg grating (FBG)-based sensors in fission and fusion reactor environments necessitates evaluating their mechanical reliability after exposure to intense radiation fields. While FBG-based sensors are already employed for in-vessel applications in nuclear reactors, further research is required to ensure their durability in fusion facilities. In ITER, FBG sensors play an essential role in monitoring the structural integrity of in-vessel components. These sensors measure critical parameters such as temperature, displacement, and deformation. Although FBGs themselves demonstrate high radiation resistance up to fast neutron fluences of ~10^{20} n/cm^2, the mechanical properties of FBGs embedded in the sensor body following prolonged neutron-gamma irradiation remain an open question.

In this work, a methodology for assessing the mechanical properties of FBG-based temperature and deformation sensors after high-fluence reactor irradiation has been developed and applied. The sensor is based on a Cu-coated FBG embedded in a sandwich structure consisting of two stainless steel plates brazed with STEMET 1101 solder. The FBG sensors were irradiated in two different research reactors: IVG.1M (Kurchatov, Kazakhstan) and WWR-K (Almaty, Kazakhstan), up to neutron fluences of ~10^{18} and ~10^{20} n/cm^2, respectively. After several years of storage, the irradiated ampule devices containing the sensors were disassembled in hot cells for post-irradiation studies. Tensile tests, elemental microanalysis, and microhardness investigations of the irradiated sensors were performed.

The findings contribute to the assessment of sensor lifetime and the enhancement of their mechanical properties during the manufacturing stage, ensuring their resilience in extreme environments within fusion and fission reactors.

02E5

Radiation damage simulations of FCC high-entropy alloys using machine-learning potentials

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High-entropy alloys exhibit exciting material properties and are currently a highly active field of research within materials science. High-entropy alloys have potential uses in extreme environments and as critical components in future fusion reactors. The creation of new and accurate interaction models is crucial for the accurate simulation of these complex alloys. Machine learning interatomic potentials (MLIP) have increasingly become commonplace in molecular dynamics simulations, as they can provide near quantum level accuracy while allowing for simulations of large systems well above the ones achievable in methods such as density functional theory (DFT), essentially bridging the gap between ab-initio method and MD. In this work, we have used low-dimensional tabulated Gaussian approximation potentials (tabGAP [1]) to simulate radiation damage in AlCrCuFeNi high-entropy alloys (HEA) as well as in other Ni-based alloys. The tabGAPs incorporate a fixed short-range repulsive potential with the intention of better describing short-range interactions relevant for radiation damage simulations. We have investigated the effects of the repulsive interactions on radiation damage simulations using MLIPs. Using the developed ML potentials we have simulated overlapping collision cascades in various compositions. We have investigated fundamental material properties and damage production of the overlapping cascades and compared them between different compositions.

[1] J. Byggmastar, K. Nordlund, and F. Djurabekova, Phys.Rev. B104, 104101 (2021).

Tentative

02E6

Accelerated design of novel RAFM steels and reduced-activation high entropy alloys for fusion applications by machine learning

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To meet the high requirements for materials served in future fusion reactors, the RAFM steels and reduced-activation high entropy alloys (RAHEAs) were optimized and designed based on machine learning (ML). The main developments were summarized as follows.

(1) An intelligent design model is developed to design the compositions and heat-treatment parameters for RAFM steels with the targeted tensile properties. The newly designed RAFM steel has higher ultimate tensile strength (UTS) by 100-400 MPa than the conventional RAFM steels while maintaining comparable elongation at 25-600 °C.

(2) An integrated design strategy for RAFM steels with desired microstructure and performance was proposed by using ML and calculation of phase diagrams (CALPHAD). Two RAFM steels with high MX volume fraction were designed and fabricated with exceptional UTS.

(3) A new strategy based on ML was proposed to accelerate the design of RAHEAs with desired phase and property. The models and phase selection rules were developed to identify single-phase body-centered-cubic (BCC) solid-solution (SS) HEAs with high accuracy. A new single-phase BCC RAHEA with desired hardness was designed and fabricated.

(4) A framework by incorporating symbolic regression (SR) and domain adaptation was proposed to improve the accuracy of property prediction based on the small datasets of RAHEAs. The domain-invariant features related to hardness were captured and used to construct the ML model with improved accuracy and robustness.

These results demonstrate that the data-driven materials development paradigm by ML with high accuracy and efficiency, accelerates the design and development of fusion materials and facilitates their engineering applications.

Tentative

02F1

Progress and Future Perspectives in the European Test Blanket Modules (TBM) Manufacturing

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Within the framework of European fusion strategy, two reference tritium Breeder Blankets concepts are developed to be tested in ITER as Test Blanket Modules: Water-Cooled Lithium-Lead (WCLL) which uses liquid Pb-16Li as a breeder and neutron multiplier, and Helium-Cooled Ceramic Pebble (HCCP) with lithiated ceramic pebbles as breeder and beryllium pebbles as neutron multiplier materials. Both concepts use as a structural material Reduced Activation Ferritic Martensitic steel, EUROFER97. Pressurized water (15.5 MPa, 295-328 °C for WCLL) and helium (8 MPa, 300-500 °C for HCCP) are used for heat extraction.

This paper briefly describes the strategy adopted for development of the TBM-related fabrication technologies, and an approach used for manufacturing and delivery of the TBMs on ITER, considering particularities of EUROFER97 structural material and relevant regulatory aspects (TBM is classified as a Nuclear Pressure Equipment).

The paper presents results related to the phase of consolidation of welding processes and related technologies for manufacturing of EUROFER97 structures and components. It discusses (i) EUROFER97 weldability demonstration for various welding techniques used for the TBM manufacturing (GTAW, EBW, LBW, HIP), (ii) filler material acceptance studies, (iii) assessment of the effect of multiple Post-Weld Heat Treatment on welds mechanical properties, (iv) dissimilar welding between EUROFER97 and 316L(N)-IG steel, and (v) Double-Wall Tubes (DWT) manufacturing.

Future steps and activities related to the TBMs manufacturing are outlined, namely, consolidation of weld matrices, development of welding procedure specifications (WPS) for particular weld joints, instrumentation integration aspects, further improvement of the HIP process applying Failure Mode Effects and criticality Analysis methodology.

Invitation

02F2

Research Progress in Manufacturing and Assembly of Tritium Breeding Blanket Component at ASIPP

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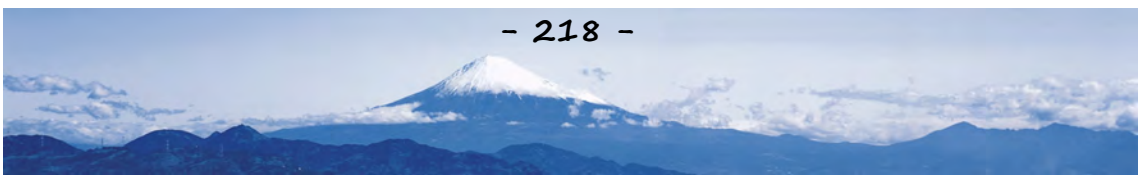
In the design of the tritium breeding blanket component, tungsten (W) is usually selected as the plasma-facing material for the first-wall and the RAFM steel as the structure materials, in which a serial of cooling channels are installed. Therefore, the manufacturing of blanket components involves the forming of internal cooling channels in RAFM steel plates and the joints of W-RAFM steel dissimilar material. In order to prevent tritium loss in the cooling channels, the bonding techniques of tritium permeation barrier (TPB) on RAFM steel also need to be explored.

In this report, some recent research progress will be provided in the manufacturing and assembly of blanket module. Firstly, we will outline the key technology issues in the manufacturing of CFETR-WCCB module [1]. Then the joining technologies of steel-steel similar and W-steel dissimilar materials have been researched and developed, including the process optimization and interface strengthening [2, 3]. At the same time, some new bonding technologies of FeCrAl and RAFM steel have been developed to prepare the reliable TPB in the cooling channels [4, 5]. In the following, the assembly welding process of the WCCB modules was proposed. Finally, the first full-size WCCB module was successfully manufactured at ASIPP. In this processing, we set up a number of large-scale machining, welding device and testing platforms. In addition, the 3D printing technology has also been applied in the manufacturing of blanket component, and some new result will be shown in this presentation.

Invited



02F3

Newly proposed PWHT-free design for reduced-activation bainitic steels

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An alloy modification of the reduced-activation bainitic ferritic steel, based on 3Cr-3WVTa steels, has been proposed to eliminate the need of post-weld heat treatment (PWHT). The material design was optimized for high-temperature, large-scale fusion structural components such as helium-cooled vacuum vessels and blanket support structures, which operate at temperatures up to 450 and 550 °C, respectively. The previously proposed alloy design incorporated increased Mn and reduced C contents, resulting in significant improvements in high-temperature mechanical properties in both base and welded materials without the need of PWHT. However, the increased Mn addition led to embrittlement after short-term thermal exposure at 500 °C, which could be further exacerbated by irradiation. Our research objective is to establish new composition ranges for the steels maintaining the same PWHT-free design concept, with the goal of preventing embrittlement after a thermal treatment or irradiation at operating temperature by minimizing the Mn addition. Since the Cr addition up to 7 wt.% stabilizes austenite phase at elevated temperature and retard the ferrite transformation during the continuous cooling process, the alloy design selected 5 wt.% Cr, rather than 3 wt.%, with expectation of improved hardenability. The newly proposed alloy, with increased Cr content, reduced C content, and a small addition of Ni, successfully balances high hardenability to form fully bainitic microstructure, relatively low hardness in the as-normalized or as-welded conditions, and minimized hardness drop after tempering. This design aligns with the proposed PWHT-free alloy design concept while addressing potential embrittlement issues due to the minimized Mn addition.

Invitation

02F4

Progress towards a mock-up vanadium alloy blanket

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Vanadium alloys are an attractive option for the structural material of a fusion blanket due to their excellent high-temperature mechanical properties, strong radiation resistance, and low activation behaviour. However, a lack of established supply chain for vanadium alloys means that performing component-scale testing is difficult, which in turn limits the technology readiness of vanadium alloys. To advance vanadium alloys beyond a technology readiness level (TRL) of 4, subcomponents must be manufactured and tested.

Kyoto Fusioneering (KF) aims to fabricate a mock-up blanket module made from V-4Cr-4Ti, which is currently the leading candidate vanadium alloy for fusion applications. We have worked with casting, forging, machining and welding vendors to produce test specimens from forged ingots weighing several tens of kilograms. This mock-up will demonstrate the key structural features required for a fusion blanket and will draw on KF's experience of making a similar demonstration module from silicon carbide composites.

The key manufacturing techniques we have tested include: casting of high purity ingots with low interstitial content, thermomechanical treatment to optimise mechanical properties, forging to produce the necessary structures (e.g. pipes, plates, flanges), diffusion bonding between V-4Cr-4Ti and tungsten to form the basis of a first wall structure, and welding techniques for joining individual parts to produce a blanket subcomponent. We will present an overview of such activities and outline next steps towards increasing the TRL of vanadium alloys for fusion applications.

Tentative

02F5

Tungsten Coatings on an Actively-cooled, Additively-manufactured, Reciprocating Probe in the Long-pulse WEST Device

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WEST (Tungsten, W, Environment in Steady-state Tokamak) is an actively-cooled tokamak operating with dual goals of sustaining long-pulses and qualifying ITER-like plasma-facing components (PFCs). A reciprocating probe system is being designed for WEST to support these goals. This system includes embedded temperature sensors, as well as removable sample slots for ex-situ surface analysis of plasma-material interactions. This presentation focuses on the coating methods used to armor the probe. The compact nature and extreme exposure conditions needed for the probe result in novel build solutions coupled with careful design analysis. Overall, this probe is 2.6 m long and ~0.05 m in diameter and can translate linearly into the vacuum chamber up to ~0.5 m. The probe end closest to the confined plasma encompasses 0.35 m of the overall length and houses the samples. In this region, the limited space, integral cooling, and linear actuator needs have pushed the design to additive manufacturing, allowing for precision cooling-channels and compactness. This probe-end is manufactured with a copper-alloy (GRCop-42) coated in a thin W coating. The coating thickness is being optimized for its thermomechanical properties, i.e., a balance between thin coatings for resilience to delamination and thicker coatings for extended lifetime. Following the success of WEST's W coating (~30 um) on a Cu-Cr-Zr alloy, we are following a multi-step validation process to ensure the coating integrity on GRCop that includes off-line high-heat flux exposures on test articles. Initial results of these exposures will be shown.

Work supported by the US DOE under contract DE-AC05-00OR22725.

Tentative

02F6

Development and Testing of a Novel FAST-Diffusion Bonding Process for Joining Eurofer97 Steel to Tungsten for Plasma Facing Components

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Tentative

Tungsten (W) is a leading candidate for the armour material of plasma-facing components (PFCs) due to its high melting point, excellent thermal conductivity, low tritium retention, and high resistance to sputtering. For structural materials, low-activation steels like EUROFER97 (E97) are favoured for their superior mechanical properties. However, joining W to E97 is challenging due to significant mismatch in their melting points, thermal expansion rates, and the formation of brittle intermetallic phases.

To address these issues, approaches such as interlayers and functionally graded layers have been developed and have shown partial success on small-scale joints. A recent study [1] compared these approaches to direct diffusion bonding of W to E97 performed with field assisted sintering technique (FAST), also known as spark plasma sintering (SPS). Surprisingly, joints made by FAST-diffusion bonding (FAST-DB) were able to withstand the highest cyclic heat loads. However, the reasons why this is the case remain unclear.

Therefore, this work identifies the key mechanisms underlying FAST-DB that address the challenges of W-E97 joining. These mechanisms are analysed with finite element modelling and validated through a comparative study using the Magnum PSI device at DIFFER. The joints are subjected to fusion-relevant cyclic thermal loads of up to 30 MW/m^2 produced by hydrogen plasma. Finally, FAST-DB is used to join further dissimilar material systems, including W-CuCrZr joints. Together with the Dr. Fritsch Group, we demonstrated the scalability of FAST-DB by producing tiles up to 10 cm×10 cm using industry-scale FAST machines.

[1] V. Ganesh, Energies 16, 3664 (2023)

03A1

EUROfusion characterisation programme of CuCrZr alloy for the DEMO divertor heat sink: mechanical and thermophysical properties

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Invited

Copper-chromium-zirconium alloy (CuCrZr) has been selected as the baseline divertor heat sink material for the European DEMO reactor. The characterisation of its mechanical and physical properties, which is coordinated by the EUROfusion Work Package Materials, is a pre-requisite for a reliable engineering design of DEMO. The purpose is to produce and process qualified material properties, which will feed into the materials database and subsequently the DEMO Material Property Handbook (MPH), which serves as a reference document for the DEMO Design Criteria (DDC).

The investigated material is a commercial CuCrZr alloy in the form of extruded rods subjected to an additional annealing at 580 °C for 2 h to reproduce the conditions of hot radial pressing, the actual joining process of a CuCrZr pipe and a tungsten block in a divertor monoblock, causing similar changes in the microstructure, mechanical and thermophysical properties.

The test matrix for the characterisation programme has been developed based on the requirements of the design work packages. It includes tests to obtain mechanical and thermophysical properties in a wide range of temperature before and after neutron irradiation with different priority levels assigned to them.

Most of the thermophysical and high-priority mechanical tests on the non-irradiated material have been completed, including tensile, fracture toughness, and low-cycle fatigue testing. This presentation will provide an overview of the CuCrZr characterisation campaign, summarising the first results obtained, and discussing the agreement of the results of different European laboratories, the reproducibility of the data and their contribution to the DEMO MPH.

03A2

Tensile and Charpy properties of EUROFER97-3 steel after neutron irradiation at 325 °C and 550 °C to damage doses of 19-24 dpa

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In December 2021, an irradiation campaign was launched in the BOR-60 fast reactor that includes EUROFER97-3 steel samples with two heat treatments (HT-1: 1100/700 °C; HT-2: 980/780 °C). Post-irradiation examinations will close parts of the gaps in the EUROFER database regarding the DEMO design phase. A comprising set of engineering data including mechanical properties at irradiation temperatures of 325 °C and 550 °C and damage doses of 20 and 40 dpa will be provided by this project, which is substantial for the DEMO blanket and divertor cassette design. In this paper, the tensile and Charpy properties of EUROFER97-3 steel irradiated at both mentioned temperatures to 19-24 dpa are presented and discussed. The tensile and Charpy test results differ strongly depending on heat treatment and irradiation temperature. The tensile samples irradiated at 325 °C show an increase in strength for both heat treatments, while irradiation at 550 °C leads to a slight decrease for HT-1 but to an increase for HT-2. At the same time, for both heat treatments, the ductility decreases for 325 °C and increases slightly after irradiation at 550 °C. Irradiation at 325 °C strongly degrades the Charpy properties of EUROFER97-3 steel: a large DBTT shift towards to higher temperatures and to significant reduction of upper shelf energy for both heat treatments occur. Irradiation at 550 °C leads to slight changes in Charpy properties: a slight DBTT shift to higher temperatures and a small reduction of upper shelf energy for both heat treatments take place.

Invitation

03A3

Multiscale Analysis of Hydrogen Isotope Effects on EUROFER97

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Hydrogen embrittlement (HE) poses a critical challenge for fusion reactor materials, threatening the mechanical integrity of structural components like EUROFER97, a key candidate for breeding blanket applications. This study explores hydrogen isotope effects on EUROFER97, focusing on hydrogen isotope transport and embrittlement mechanisms under fusion-relevant conditions.

EUROFER97 samples are charged with a gas mixture containing 500 ppm tritium (^{3}H) in hydrogen to simulate reactor conditions and enable hydrogen distribution analysis. The global hydrogen content is quantified using Thermal Desorption Spectroscopy (TDS), while Atom Probe Tomography (APT) examines microstructural changes and hydrogen distribution at the atomic scale.The mechanical properties of the non-charged and charged samples are evaluated through in situ micro-compression tests of single-crystal micropillars.

Preliminary experiments conducted to establish the methodology revealed no peak at 3 a.m.u. in the APT mass spectrum for the uncharged state. However, a distinct peak was detected in the charged state, indicating changes attributable to hydrogen isotope charging.

The same protocol will be applied to neutron-irradiated EUROFER97 samples from the SPICE irradiation program to study irradiation hardening and hydrogen effects on the irradiated microstructure. Further, an in situ Multi-Channel Plate (MCP) detector will measure local tritium (^{3}H) content during deformation with high spatial resolution, providing insights into hydrogen embrittlement mechanisms.

These investigations will advance the understanding of HE and radiation damage, aiding in the development of hydrogen-resistant materials for future fusion reactors.

Tentative

03A4

Impact of Manufacturing Processing Control on Irradiation Precipitate Stability and Swelling of Advanced Steels

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Castable Nanostructured Alloys (CNAs) are advanced steels developed for fusion energy systems. Laboratory scale heats of CNAs have shown improved strength due to carefully designed processing, resulting in a high density (>10^22 m^-3) of small (0.25 Ton) may impact the starting microstructure and thus the irradiation response of these alloys. This work evaluates the influence of laboratory to industrial production scaling on the microstructural evolution under ion irradiation in CNAs through a systematic dual ion irradiation campaign. We evaluated CNA alloys across two different compositions and five heat treatments between 1 - 50 displacements per atom (dpa), using dual ion irradiation with a damage rate of 7x10^-4 dpa/s and 10 appm He/dpa at 500 °C. We show all CNA variants studied undergo inter-lath TiC precipitate dissolution and intra-lath growth at different rates regardless of processing or composition history. In one instance, two variants with identical composition but differing significantly in quenching rates showed substantial decreases in inter-lath precipitate density yet large deviations in intra-lath growth rates. At 50 dpa, the mean TiC precipitate size in the high quench rate variant had a 70% increase in size while the non-idealized (slow) quenched material showed a 45% increase in mean size. We will show this disparity, and others found are due to both grain boundary characteristics and starting TiC phase fraction which impacts sink strength evolution and diffusional characteristics in CNA under ion irradiation conditions at fusion-relevant temperature.

Tentative

03A5

Local deformation behavior in proton-irradiated reduced-activation ferritic/martensitic steels under nanoindentation

Sumin Lee[1], Diancheng Geng[2], Yuma Furukawa[1], Sanghoon Noh[3], Yasuyuki Ogino[2], Sosuke Kondo[2], Minha Park[2], Hao Yu[2], Kasada Ryuta[2]

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Tentative

Reduced-activation ferritic/martensitic steels (RAFMS) has been developed for fusion reactors due to their superior radiation resistance by high sink strength of tempered martensite microstructure, which comprises fine laths, high dislocation density, sub-grain structures, and distributed precipitates. These complex and hierarchical microstructural features are also beneficial to the mechanical properties, but can lead to localized deformation and then brittle fracture, both before and after irradiation. However, local deformation behavior of RAFMS is still unclear due to the further complex deformed microstructure. Our preliminary observations suggested that slip band formation and pile-up around the nanoindentation imprint in RAFMS can be an indicator of local deformation in their lath-martensite [Kasada, ICFRM-21]. Hence, this study investigates the slip-band and pile-up formation under nanoindentation with detailed crystalline-level analysis for 2 MeV proton-irradiated advanced reduced-activation alloy (ARAA) RAFMS up to 0.2 dpa.

After the proton irradiation, distinct slip bands formed around imprints, indicating that slip in the lath was suppressed while slip at a vicinity of lath-boundaries was promoted likely due to irradiation-induced slip localization. In order to confirm this, 3D-EBSD analysis were employed to elucidate the deformation mechanisms at the crystalline level. These investigations are expected to contemplate critical discernment into local deformation mechanisms of irradiated RAFMS to establish precise irradiation embrittlement models.

03B1

Development of repair technology for functional ceramic coating

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Functional coatings for tritium permeation barrier and corrosion protection have developed and proved high performance using ceramics. During the long-term reactor operation, however, the degradation of the coating such as cracking and wastage must be taken into account because the coating is exposed to a severe environment such as thermal cycles. The degraded components will be replaced in the periodic inspection, while the maintenance cost should be high. If the repairing technology for damaged coatings is established, the economic efficiency of the reactor blankets will be improved. Therefore, in this study, we investigated on repairing of the ceramic coating through two approaches: recoating and self-healing.

Zirconium oxide coatings were fabricated on steel substrates using a liquid precursor by dip-coating. The coated samples were exposed to thermal cycles with different parameters. After that, we conducted recoating for the damaged samples with the same coating process. Self-healing of the coating was investigated through addition of chromium carbide nanoparticles to the coating precursor with a concentration of up to 5 wt% and dispersion of aggregated nanoparticles using a bead mill disperser. Deuterium permeation measurements, scratch tests, and surface/cross-sectional observations were conducted to evaluate the coating properties.

After recoating, deuterium permeation flux decreased by two orders of magnitude in comparison of that of the damaged coating, proving a recovery of permeation barrier performance. As for self-healing technique, it is proved that the dispersion process is effective to avoid aggregation of nanoparticles. These results indicate that both methods are promising in the coating repairing technology.

Tentative

03B2

Tritium release and defect annihilation of neutron-irradiated lithium ceramics with Pb addition

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To enhance the tritium release without increasing the lithium density in the material, Pb, which is a typical neutron multiplier, was composited in Li_2TiO_3-$0.5Li_4SiO_4$. The lithium ceramics with the Pb content of 5 wt% and 20 wt% were prepared. The neutron irradiation with the flux of 5.5×10^{12} n cm^{-2} s^{-1} was conducted using the Kyoto University Research Reactor (KUR). The tritium generation in ceramics was measured by a tritium thermal desorption spectroscopy (tritium-TDS) system and a liquid scintillation counter (LSC). The tritium desorption rate per volume of the ceramic was increased. Especially, tritium release as HT was significantly enhanced by Pb addition, demonstrating that the chemical bonding amount of tritium by oxygen was reduced. The activation energies for tritium desorbed both as HT and HTO were reduced at higher Pb content. On the other hand, the defects introduced by neutron irradiation, i.e., E'-center and O-center, were measured by electron spin resonance (ESR). The annihilation process of the defects was investigated by heating the ceramic isochronally. The density of O-center was increased rapidly when the Li_2TiO_3-$0.5Li_4SiO_4$ was heated from ambient temperature to 373 K, while that for the Li_2TiO_3-$0.5Li_4SiO_4$-5%Pb increased slowly. With a further increase of Pb to 20 wt%, the initial density of O-center was reduced significantly. It was decreased almost linearly against temperature. The recovery rate of defects was determined by the Pb content. It can be known that Pb bonds with interstitial oxygen, which facilitates the tritium release as HT.

Invitation

03B3

Thermal Stability of Thin-Film Coatings for Breeder Blankets

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Multilayer coatings are being explored for use in breeder blankets of future fusion power plants to provide corrosion resistance and inhibit tritium permeation into structural steels. Rare earth oxides are candidates for tritium permeation barriers and W is being considered as a Li corrosion resistant topcoat. A key challenge for this multilayer approach is the mismatch in coefficient of thermal expansion (CTE) between the steel substrate and the multilayer coating, which can lead to coating failure through cracking and delamination. Therefore, it is important to understand this effect, as well as how other sources of stress can contribute to the thermal stability of these coatings and what can be done to mitigate against it. Here, the effects of interlayers, composition, and deposition processes on the thermal stability of thin-film coatings are investigated.

Coatings of rare earth oxides and W were produced via vapour deposition to create ~2 μm thick coatings on G91 steel substrates. Isothermal holds and thermal cycling, up to 650 °C, were used to test thermal stability. Post heat treatment characterisation using X-ray diffraction and cross-sectional microstructural analysis enabled assessment of different contributions to the coating stress state.

The use of interlayers resulted in excellent adhesion for all coating systems, including for cyclic heating of an alumina coating, and hence improved the thermal stability. The effect of tailoring the microstructure to relieve stress was explored, including by varying the composition and deposition processes. Of particular interest, evidence of nanoscale layered microstructures were seen in TEM analysis.

Tentative

03B4

Current R&Ds on Advanced Breeding Functional Materials for JA DEMO activities

Jae-Hwan Kim[1], Taehyun Hwang[1], Yutaka Sugimoto[1], Hiroyasu Tanigawa[1]

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We have dedicated ourselves to researching and developing granulation process of breeding functional materials for tritium breeders as well as characterization of beryllide blocks as neutron multipliers over the past several years. Through trial and errors, beryllium intermetallic compounds (beryllides) have been successfully fabricated by plasma sintering process while normal Li_2TiO_3 (LTO) and solid solution Li_2TiO_3 with Li_2ZrO_3 (LTZO) as 1 mm-pebble, have been successfully fabricated by an emulsion process. And also the optimization for the fabrication process has been carried out to improve the yield and the specific properties.

Characterization of beryllide blocks, including mechanical properties, reactivity to water vapor, swelling properties, hydrogen retention, and compatibility with other materials, has shown that the new blanket concepts using beryllides are more promising than that using beryllium pebbles.

For LTO pebbles, we found that the tritium release property showed low tritium retention and a high portion of HT when released. Currently in-situ T release test for LTO pebbles with 95% ^6Li enrichment is undergoing. Additionally, we evaluated the gas pressure drop in the pebble bed, indicating no problematic pressure drop.

In this study, current R&Ds on advanced breeding functional materials in JA DEMO activities, regarding not only material fabrication process but also characterizations. Additionally, the refinement and recycling process of these materials established by the QST will be addressed by applying the innovative refining process with chemical reaction and microwave heating.

Tentative

03B5

Progress and Challenges in the Development of Neutron Multiplier Materials for Breeding Blankets

Ramil Gaisin[1], Rolf Rolli[1], Vladimir Chakin[1], Guangming Zhou[1], Francisco A. Hernandez[1]

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Neutron multiplier materials (NMMs) play a crucial role in breeding blankets, ensuring a sufficient neutron supply for tritium breeding while also contributing to energy extraction and neutron shielding. Among all potential candidates, only beryllium and lead are considered viable for this purpose.

This study explores the development and characterization of both beryllium-based (titanium beryllide) and lead-based (plumbides and liquid lead) materials.

Titanium beryllide currently stands as the most technologically advanced NMM, reaching a TRL of 4, with 20 full-size blocks successfully fabricated using an industrially scalable process. Special emphasis is placed on simulating the operational conditions of titanium beryllide blocks within the breeding blanket, particularly their response to long-term thermal cycling (over 2000 cycles) in a helium environment, undergoing rapid heating and cooling between 400 °C and 1000 °C. Additionally, ongoing neutron irradiation campaigns of various beryllides will be discussed.

Lanthanum and zirconium plumbides are investigated as potential backup solutions for NMMs. This study presents their fabrication methodologies alongside a detailed characterization of their properties. Particular attention is given to their corrosion resistance at elevated temperatures, which remains a significant challenge due to the high reactivity of lead. Additionally, stagnant liquid lead contained in steel vessels is examined as a promising alternative. Preliminary findings on the corrosion resistance of the structural material EUROFER97 in contact with liquid lead and other environments under a temperature gradient, representative of breeding blanket conditions, are also discussed.

Tentative

03C1

Materials Irradiations in HFIR for Fusion Relevant Materials

Adrian M Schrell[1], Richard H Howard[1], Christian M Petrie[1], Yuji Hatano[2], Takaaki Koyanagi[1]

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Absent a fusion prototypic neutron source, Oak Ridge National Laboratory's (ORNL) High Flux Isotope Reactor (HFIR) has long been an attractive facility for providing the high steady-state displacement damage rates that are required to simulate fusion conditions. This work describes several recent irradiation concepts that have been designed or tested to support material development and the underlying science for future fusion reactors.

In-situ deuterium capsules have been built to examine hydrogen trapping in plasma facing components. This capability is of importance, as previous ion irradiation studies have reported that irradiation-induced defects can increase the hydrogen retention. A capsule has been built to allow charging from a hydrogen environment during irradiation in the flux trap in a manner that greatly exceeds previous pressures of this type of study.

A high-temperature thermal gradient capsule has been designed and readied for insertion. Traditional capsules target a single temperature, which contributes to the lower throughput of irradiation experiments. Thermal gradients allow test specimens to span up to a 1000 °C, allowing a broader range of temperatures with a single capsule. Additionally, the design facilitates "high" temperatures reaching 2000 °C, thus enabling the evaluation of plasma facing materials under extreme conditions.

Finally, a silicon carbide (SiC) composite channel box was recently irradiated in HFIR to investigate the effects of differential swelling on lateral bowing. The experiment results inform dimensional stability of SiC-based flow channel insert. These three experiments demonstrate HFIR's utility as a materials test reactor for the fusion space.

Tentative

03C2

Overview of the DONES Programme

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The DONES Programme aims to create a database of materials for fusion reactors, focusing on RAFM steels like EUROFER97 under high neutron flux.

Developed within this framework, IFMIF-DONES (International Fusion Materials Irradiation Facility, DEMO-Oriented Neutron Early Source) is a powerful neutron irradiation facility designed for studying and qualifying materials for fusion reactors. It also contributes to the development of tritium and breeding blankets technologies and is part of the European roadmap to fusion electricity.

The main goal is to develop a database of material properties under neutron irradiation similar to a fusion reactor. Neutrons are generated by an accelerated deuteron beam hitting a liquid lithium curtain, mimicking the conditions of a fusion reactor's first wall. Initial experiments will also study tritium behavior in breeding blanket systems.

The IFMIF-DONES facility has moved from the preliminary design to the detailed design phase, with some components already in production. The construction phase officially began with the first DONES Steering Committee on 16 March 2023, ensuring a smooth transition of ongoing activities. This paper summarizes the DONES Programme and the current status of the IFMIF-DONES design, developed through the EUROfusion Work Programme. It also incorporates lessons learned from the IFMIF/EVEDA Project under the BA Agreement between EURATOM and the Japanese Government.

The focus will be on the design status of the DONES Facility and its interaction with the DONES Users Community to define the experimental programme, emphasizing fusion applications.

Tentative

03C3

Synergistic irradiation-thermal-mechanical loading in fusion steels: First results from a new facility

Dave Lunt[1], Ben Poole[1], Thomas Hughes[2], Rhys Thomas[2], Laurence Skidmore[2], Philipp Frankel[2], Ed Pickering[2], Samir de Moraes Shubeita[3], Matt Carrington[1], Cory Hamelin[1], Chris Hardie[1], Allan Harte[1]

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Much of the current understanding of changes to material mechanical properties due to irradiation exposure is obtained with sequential thermal-irradiation and then thermal-mechanical loading. This is not representative of the synergistic nature of irradiation-thermal-mechanical loading in a fusion power plant, leading to inaccuracies in component lifetime assessments. Here, we detail the commissioning of synergistic irradiation-thermal-mechanical testing capability at the Dalton Cumbrian Facility, where samples are subjected to proton irradiation while simultaneously applying mechanical load at elevated temperature, using a testing frame designed for live measurement of temperature and strain during the experiment.

This new facility for in-situ ion irradiation-thermal-mechanical materials testing was used to compare the thermal creep and irradiation creep response of a ferritic-martensitic steel at the operating conditions likely in a fusion power plant. Understanding irradiation-enhanced creep is particularly important for timely qualification of high temperature structural materials for fusion, as it is long-term deformation phenomenon that is not well understood and hence poses a risk to confidence in structural integrity assessments of components. Furthermore, novel scientific experimental methodologies for quantifying the deformation and damage at multiple length-scales were applied; in-situ optical digital image correlation was used to measure macroscopic creep strains, and advanced microscopy for ex-situ high-resolution strain mapping in a FEG SEM was used to gain insight into changes to nano-scale deformation mechanisms at the microstructural length scale when moving from the thermal creep regime to the irradiation creep regime.

Tentative

03C4

Current Status and Plan of Fusion Material Research with Neutron Sources and Heavy Ion Irradiation Facility

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To compare with the actual neutron irradiation effect, the following three facilities and related R&D were introduced in this presentation;

(1) HANARO irradiation test under ITER TBM program: thermal neutron, volumetric irradiation (larger specimen; standard or less), which can be used for irradiation DB up to 3 dpa comparing with unirradiation DB with ARAA of Korean RAFM steel, ARAA (Advanced Reduced Activation Alloy).

(2) A cyclotron based neutron source has been developed with existing 30 MeV cyclotron at KAERI, which is used for radio-isotope production and newly used for developed for neutron radiography. It can supply high energy and volumetric neutron flux but limited dpa, which can be used high energy neutron effect.

(3) KAHIF (KAERI Heavy Ion Irradiation Facility), a heavy ion beam irradiation test facility capable of accelerating heavy ions to 1 MeV/nucleus has been operated at KAERI. A KAHIF was built and named it (KAHIF). From the beam acceleration test, the He+ and Ar9+ ion beams were accelerated to 4.2 and 42.3 MeV, respectively. Then, new Fe13+ ion source was installed for fusion structural material test under Fusion Preceding Program in Korea: Fe-ion irradiation test, surface irradiation (micro-specimen, high dpa effect) which can be used high dpa test, irradiation mechanism study.

Because each of them has its own advantage and limitation, a strategy is also introduced to develop fusion material irradiation research considering how to induce synergy with these ongoing projects.

Tentative

03C5

Development of an Irradiation Test System for Basic Material Studies

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Tentative

Since fusion reactors have not yet existed, irradiation test data have been obtained by using fission reactors and accelerators, and fundamental data on irradiation behavior have been prepared, through which the basic material behavior of various candidate materials for fusion reactors has been understood. The importance of the role of fission reactors and accelerators in the process of acquiring and accumulating irradiation test data is expected to remain unchanged.

JAEA, in cooperation with the Institute for Materials Research (IMR) and the Research Center for Accelerator and Radioisotope Science (RARiS) of Tohoku University, is currently developing a combined irradiation test system that can simulate the in-core environment of a fission reactor, especially a water-cooled light water reactor. The primary objective of this irradiation test system is to create an environment that combines proton beam-induced irradiation damage, high-temperature water corrosion, and radiolysis. The system aims to enable measurement of corrosion behavior of materials under irradiation. Proton beam irradiation will be carried out using the 930-type AVF cyclotron, with beamline development progressing toward a maximum acceleration voltage of 20 MeV and a maximum current of 50 μA.

The first irradiation experiment data using this system is scheduled to be acquired in March 2025, and the details of the irradiation data acquired after March 2025 will be presented at 22nd International Conference on Fusion Reactor Materials (ICFRM-22).

03D1

Development of Boron-strengthened B-RAFM Fusion Steels

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Boron is a key alloying element that enhances the creep strength of steels, notably in grade-92 (30 ppm) and the MarBN-class (100-150 ppm). In MarBN steels, boron modifies the carbide structure, replacing $M_{23}C_6$ precipitates with $M_{23}(C,B)_6$, slowing coarsening under high-temperature creep conditions and delaying the transformation of martensite to ferrite. This modification improves the creep lifetime, which is crucial for steels used in fossil-fuel power generation.

However, the application of boron in fusion breeder blanket steels presents challenges. Boron can degrade toughness and exacerbate radiation damage through $B_{10}(n,a)$ reactions; however, the magnitude of these effects could be tolerable for use in nuclear applications via optimisation of chemistry and heat treatment. To assess the feasibility of transferring this technology from fossil-fuel to fusion energy applications, UKAEA, in collaboration with partners experienced in MarBN steel development, initiated the “TRANSFUSION” project. This study represents the first step in determining whether boron-nitride strengthening can maintain its effectiveness within RAFM (Reduced Activation Ferritic-Martensitic) alloy systems.

In this talk, we will present our ongoing research on Boron-Added RAFM (BRAFM) steels, focusing on our alloy design approach, results from accelerated creep tests, and microstructural comparisons. We will also discuss the implications of these findings for the development of advanced materials for fusion energy applications.

Goodall et al. Development of a boron-containing reduced activation Ferritic-Martensitic (B-RAFM) steel. Ironmaking & Steelmaking. 2024 doi:10.1177/03019233241273484

Tentative

03D2

Developing and testing silicon carbide composites for fusion-relevant conditions

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[1]UK Atomic Energy Authority

SiC fibre reinforced SiC composites have many desirable characteristics as a structural material for the breeder blanket of fusion power stations including mechanical and microstructural stability under high temperature irradiation (~600?1000°C), low neutron absorption cross-section, and low induced activation. They are resistant to many corrosive environments and have low hydrogen isotope permeation. This presentation will include a variety of work undertaken by UKAEA and collaborators on understanding the performance of SiC in fusion environments and finding solutions where there are challenges.

Ion implantations informed by FISPACT calculations have been used to simulate fusion-relevant transmutations simulating the C(n,3α), Si(n,d)Al reactions, and the faster burnup of silicon. A variety of post-irradiation characterisation has been performed including micromechanics (novel micro push-out method), grazing incidence XRD and AFM to measure strain and swelling, TEM, and local electrical property measurements to investigate radiation effects on functional performance. These results are related to microstructure inform new manufacturing methods. In-situ TEM helium implantations using the MIAMI facility have explored extremely high doses of helium, beyond 40 years predicted concentration. Studies of deuterium retention in neutron irradiated SiC have been carried out for the first time, showing that radiation defects and grain boundaries can reduce permeation. First principles calculations have been performed to elucidate and help explain experimental results, particularly around hydrogen and helium interaction with defects.

Progress on joining experiments and nuclear-grade manufacturing will be presented as part of our work on enabling SiC composites to be used in fusion engineering components.

Tentative

03D3

The saga of nonferrous metallurgy in the quest of new radiation-resistant fusion materials beyond pure tungsten

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Innovative materials development is essential for advancing nuclear fusion energy technologies, which require high-performance materials capable of withstanding extreme conditions such as intense heat and radiation exposure. Tungsten (W) has historically been a preferred material due to its high melting point, availability, and cost-effectiveness; however, it remains highly susceptible to radiation damage. This limitation highlights the need for alternative materials that can meet the demanding requirements of fusion reactors, posing a problem for contemporary non-ferrous metallurgy. In this work, we will review the scientific data accumulated all over the decades in the radiation damage of W and its binary alloys. Although recent high-entropy materials, such as multicomponent equimolar refractory alloys and carbides, offer improved performance in energetic particle environments compared to traditional W and W-based alloys, they remain impractical for developing new radiation-resistant materials needed for fusion energy applications. Recent strategies towards new “chemically simplified alloys” will be reviewed to show that the fundamental metallurgical concept of terminal solid solution may still be a suitable approach for the quest of new radiation-resistant fusion materials. This approach aims to bridge the gap between using simple pure W and complex high-entropy alloys, offering a potential pathway for innovation in nuclear fusion research.

Invitation

03D4

Effect of disruption heat loads on tungsten-steel fusion composites for tokamak front wall

Hannah E Tipping[1], James Hargreaves[2], Daniel Wilkinson[3], Mariia Zimina[1], Thomas Morgan[2], David Bowden[4], Tomas L Martin[1]

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The operating conditions of a fusion tokamak pose a significant risk to the structural integrity of the first wall components. Behind the protective layer of tungsten shielding, reduced activation ferritic martensitic steels are used to support the reactor wall and coolant channels. Eurofer-97 is one of these steel candidates, designed to withstand standard operational temperatures up to 550 °C. In addition to the steady state, transient disruptions caused by plasma instabilities are expected up to 2500 times during a fusion reactor's lifetime [1], imposing up to 2 GW/m^2 heat flux for between 1-4 ms [2]. Finite Element Modelling suggests transients will raise Eurofer-97's temperature to 950 °C and vaporise the coolant [3]. To test this, specimens of Eurofer-97, with varying thickness of tungsten shielding, were exposed at DIFFER's Magnum-PSI facility. A steady state 500 °C argon plasma was used to replicate the fusion environment whilst simultaneously the steel was exposed to 1-1000 pulses of a high powered LASAG laser, reaching a maximum ΔT of 1200 °C. Preliminary microstructural characterisation showed a significant topographical damage region with bulk globules of melted material up to 150μm in diameter. Cross sectional electron backscatter diffraction and hardness testing will be compared for different tungsten shielding thicknesses, and the resulting implications for the performance of the tokamak first wall will be discussed.

[1] J. Wesley, in 15th Workshop on MHD Stability Control, Madison, WI, Nov. 2010.

[2] F. Maviglia et al., Fusion Eng. Des., vol. 177, Apr. 2022.

[3] J. Hargreaves et al., Mater Des, vol. 244, Aug. 2024.

Invitation

03D5

Nanoprecipitate strengthened Vanadium alloys for Fusion First Wall and Blanket Application

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Vanadium alloys such as V-Cr-Ti are attractive candidate fusion blanket structural materials whose operating temperatures exceed candidate reduced activation ferritic martensitic steels. However, the current operating temperature of the leading candidate V-4Cr-4Ti (V44) alloy is limited by thermal creep to <700C. Moreover, V44 alloys are susceptible to hardening and embrittlement when irradiated below 400oC due to a lack of microstructural features to mitigate irradiation defect clustering. In this work, we will present newly designed and fabricated vanadium alloys incorporating a high number density of TiC nanoprecipitates. Such precipitates are anticipated to improve high temperature strength and as mitigating sinks to lessen irradiation hardening. Computational thermodynamics was used to design the alloy chemistry and guide heat treatment parameters. Experimental lab-scale alloy ingots were made by arc-melting and casting followed by hot forging, cold rolling and a two-step solution and aging heat treatment. A multimodal characterization will be presented including SEM, TEM, XRD, APT, tensile and Charpy test. Alloy performance is presented as a contrast with the well understood reference V-4Cr-4Ti alloy. The V44C alloy shows the tensile strength double that of V44 at 700C at the cost of reduced impact energy at room temperature. Strength improvement is discussed based on particle/solid solution strengthening and the fracture behavior correlated with the fractography results. Further optimization strategy on the nanoprecipitate strengthened V alloys is discussed.

Research sponsored by the ORNL LDRD program and the Office of Fusion Energy Sciences, U.S. Department of Energy, under contract DE-AC05-00OR22725 with UT-Battelle LLC.

Tentative

03E1 "Presenting Author: Zachary Bergstrom"

Developing Innovative Material Solutions for Fusion Pilot Plants at General Atomics

Tyler Abrams[1], Ali Basaran[1], Zachary Bergstrom[1], Giacomo Dose[1], Brian Grierson[1], Jerome Guterl[1], Carlos Monton[1], Zana Popovic[1], Ruben Santana[1], Gregory Sinclair[1], David Weisberg[1]

[1]General Atomics

General Atomics (GA) is developing innovative material solutions for key components of a fusion pilot plant (FPP). For first wall applications, a new generation of tungsten-silicon carbide functionally graded materials were recently exposed to high heat flux in the divertor of the DIII-D tokamak. Pre-annealed samples were roughened by plasma exposure but the front-facing W coating remained intact in all cases. XRD analysis has indicated the formation of new material phases, including tungsten di-silicide, and DFT modeling is underway to further correlate the observed peaks. For structural applications, SiCf/SiC fiber composites produced at GA via CVI and subsequently sealed with CVD were also recently tested under high heat flux. Some micro-cracking was found in the sealant layer but there was no evidence of macroscopic fracture or other failure mechanisms. For tritium fuel cycle applications, GA is pioneering sputtered thin films as functional materials for selective separation of hydrogen species from a mix of gases to enable direct internal recycling technologies. We have demonstrated the capability to grow palladium (Pd) nanowires directly within porous alumina membranes; combining this with a thin layer of niobium results in a Pd nanowire/Nb thin film "hybrid" metal foil pump that is leak-tight and robust against differential pressures in vacuum environments. General Atomics welcomes public and private collaborations to further develop material technologies for fusion energy.

Tentative

03E2 "Presenting Author: Andrey Litnovsky"

Advanced plasma-facing components for future fusion power plants: an overview and first results of the “a-SMART-FIT” project

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Advanced materials for fusion power plants are under intensive development worldwide. Recently, a project “a-SMART-FIT” supported by German federal government has initiated aiming at supporting the key technologies for a fusion power plant. The main goal of a “a-SMART-FIT” is the industrial realization and testing of water-cooled components from two innovative material concepts: SMART and Wf/W (FIT). SMART - tungsten-based alloys provide a unique self-passivation feature during a severe accident in a power plant, whereas Wf/W, a tungsten fiber-reinforced tungsten addresses the intrinsic brittleness of bulk tungsten by introducing a pseudo-ductility. These two materials are supposed to address issues of the entire plasma-facing area of the main wall and the divertor in a fusion power plant.

The consortium consisting of industrial partners Dr. Fritsch Sondermaschinen GmbH, which is coordinating the project, Zoz GmbH, Forschungszentrum Julich and two universities: RWTH Aachen University and Ulm University, has achieved remarkable results already. The SMART samples sintered from industrial powder alloyed by Zoz GmbH feature the 104-fold suppression of tungsten oxidation in case of accident as compared to that of pure tungsten. SMART materials featuring relative density exceeding 97% became available at Dr. Fritsch Sondermaschinen GmbH. At the same time, industrial Wf/W components have reached the linear dimensions of 100×100 mm^2 and density exceeding 93 %. Bulk Wf/W has withstood the heat fluxes of up to 15 MW/m^2. An overview of the consortium, current results of industrial realization of plasma-facing components along with research activities will be presented.

Invitation

03E3

Multimodal Synchrotron and Neutron Characterization of PFC Tungsten-Steel Interfaces in Laser Additive Manufacturing

Natan Garrivier[1], Malgorzata Makowska[2], Steven Van Petegem[2], Florencia Malamud[2], Markus Strobl[2], Manuel Pouchon[2]

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Advances in fusion reactors core architecture have positioned metallic additive manufacturing (AM) as one of the prime manufacturing processes for components with complex internal structures like the breeding blanket and divertor. Laser Powder Bed Fusion (LPBF) and Laser Directed Energy Deposition (L-DED) are being considered to fabricate these parts, with tungsten (W) serving as the leading plasma-facing material to shield Reduced-Activation Ferritic-Martensitic (RAFM) steel components. Ideally, these materials are directly bonded together as one component with integrated cooling channels. However, joining those two very dissimilar materials using laser AM poses a serious and multifaceted challenge.sIn this study, bi-layered specimens of tungsten on FM steel were produced using LPBF and L-DED. The resulting microstructure, composition, and strain field were analyzed using a combination of advanced characterization techniques, including EBSD, synchrotron μXRD, μXRF, 2D-XRD mapping, 3D-XRD tomography, operando XRD and neutron imaging. Comparative analysis of these techniques revealed process-driven mesostructural variations that were undetectable with conventional methods. This presentation will provide a cross-correlated assessment of synchrotron and neutron-based characterization techniques for metallic AM, demonstrating their effectiveness in uncovering unexpected process-dependent mechanisms that are critical for the fabrication and performance of fusion reactor components.

Invitation

03E4

Irradiation Effect on Tungsten/Silicon Carbide Interface Joined by Hot-press

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Tungsten is expected for a plasma facing material. Silicon carbide composites are expected for structural materials. Tungsten and silicon carbide have close coefficient of thermal expansion. High heat load doesn't cause significant shear strength at the interface. It is also considered high temperature joining process doesn't induce significant residual stress at the interface. The objective is to understand irradiation effect on microstructure, mechanical properties and thermal properties of the interface between tungsten and silicon carbide joined by hot-press. Tungsten and silicon carbide plates were joined by hot-press at 1500~1700 °C with up to 20 MPa. The joined sandwich plates, which had thin tungsten foil between silicon carbide plates were also prepared. The necked specimens were machined from the sandwich plates to evaluate torsion shear strength of the interface. Neutron irradiations were carried out at the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) under Japan-US collaborations, the PHENIX and the FRONTIER projects. Microstructure, mechanical properties and thermal properties of the interface between tungsten and silicon carbide following the neutron irradiation were evaluated at Low Activation Materials Development and Analysis (LAMDA), ORNL.

Silicon carbide and tungsten were joined successfully by hot-press. W5Si3 and WC were formed at the interface during hot-press process. Apparent change of microstructure and shear strength at the interface weren't observed. Details of neutron irradiation effect on the interface will be presented.

Invitation

03E5

Numerical Simulations of the Effects of Residual Surface Stress Distribution on Micro-Hardness Data in Fusion Reactor Structural Materials

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The extreme service conditions of fusion reactors impose higher performance requirements on structural materials. Surface treatment is an effective method to improve fatigue resistance, stress corrosion resistance, and wear resistance. However, the residual stress introduced by machining or other surface treatment has a decisive impact on the mechanical properties and long-term service performance of materials. While compressive residual stress can enhance material durability, excessive or non-uniform stress distribution may lead to premature cracking or deformation, causing adverse effects. Therefore, accurately modeling and evaluating residual stress distribution is crucial for optimizing the design and performance of fusion reactor materials.

In this study, a finite element analysis (FEA) method was employed to model the residual stress field induced by surface treatment and to evaluate its impact on micro-hardness data. Three numerical approaches were used to introduce residual stress distribution below the specimen surface: (1) the stress gradient loading method, which simulates residual stress distribution by applying predefined stress gradients; (2) a deformation approach, which derives the residual stress field by generating an inhomogeneous plastic strain gradient; and (3) a method that creates residual stress with homogenous displacement in a material of inhomogeneous constitutive behavior. A microhardness model was developed based on residual stress simulations to evaluate its impact on material properties. The model's accuracy was verified by comparing simulated values with experimental measurements, enabling assessment of different stress simulation methods. The findings contribute to improving numerical simulation reliability and provide insights for optimizing material performance under extreme conditions in fusion reactors.

Tentative

03F1

R&D of advanced functional materials and engineering qualification for the fusion solid blanket

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The tritium-producing blanket is a key component for fusion reactors to realize tritium self-sustainability. With growing demand on the technology, advanced functional materials and scaling-up productions have been comprehensively investigated in China. It is not only the 50%-higher strength core-shell Li_4SiO_4-Li_2TiO_3 pebbles with 100 °C reduction in T-release temperature but also the porous structural 3D-printing Li_2TiO_3 blocks capable of >80% improvement in effective thermal conductivity are developed after comprehensive studies on various breeders. In further, the tritium release behavior, mechanical performance, long-term annealing and irradiation effect of advanced breeders were investigated. In addition, both beryllium and beryllides as advanced neutron multiplier was developed. The Be12Ti was made into neutron multiplier components with ingot in diameter of 50 mm and beryllium pebbles in size of 0.3~1.5 mm. A production capacity of 1 kg/patch was reached recently. Furthermore, significant effects of irradiation, surface scratching and corrosion on various T-barrier coatings are presented. A non-linear dependence of deuterium permeation on irradiation dose is found for Cr2O3/Al2O3 coating up to 50 dpa. A novel ZrN/ZrO_2 coating exhibits good corrosion resistance to lithium. Moreover, scaling-up fabrication capability of tritium breeding materials progressed a lot. The engineering qualification of Li_4SiO_4 pebbles in SWIP and CAEP and Beryllium pebbles in SWIP have been completed. The continuous production and testing of three batches of products were demonstrated, showing stable performances. The production capacity of Li_4SiO_4 and beryllium pebbles reaches more than 1 and 2 tons/year, providing important support for fusion reactor construction.

Tentative

03F2

Investigating Deuterium Retention and Transport Behavior in Neutron-Irradiated Tungsten by Gas-Driven Permeation, Laser Ablation Mass Spectrometry, and Thermal Desorption Spectroscopy

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Tungsten is the primary plasma-facing material in the WEST tokamak and has been chosen as the divertor material in ITER due to its desirable properties for use in the fusion environment. However, the performance of tungsten under prolonged neutron irradiation remains uncertain, specifically regarding hydrogen isotope retention and permeation. Exposure to the 14.1 MeV-peaked fusion neutron spectrum introduces microstructural defects to plasma-facing materials that can impact hydrogen isotope retention and transport, raising concerns for fuel retention and material performance. To investigate the hydrogen isotope trapping behavior in neutron-irradiated tungsten, we have performed experimental deuterium retention studies on neutron-irradiated tungsten samples exposed in the High Flux Isotope Reactor (HFIR). These samples are then implanted with deuterium in the Tritium Plasma Experiment (TPE). Deuterium gas-driven permeation measurements are also performed on neutron-irradiated tungsten samples to assess the effects of neutron irradiation on hydrogen isotope transport behavior. To evaluate the retained deuterium, the depth-dependent deuterium concentration profile will be characterized by using laser ablation mass spectrometry (LAMS), followed by measuring the retained deuterium desorption spectrum and de-trapping behavior using thermal desorption spectroscopy (TDS). Subsequent to the TDS measurements, the samples will be characterized by an additional post-TDS LAMS analysis to assess any remaining deuterium in the near-surface region. Experimental findings will be compared to available literature and modeling predictions to refine our understanding of hydrogen isotope behavior in tungsten under fusion-relevant conditions.

Invitation

03F3

Water Lead Ceramic Breeder (WLCB) Breeding Blanket (BB) for the EU DEMO: neutronic campaigns for the optimization of its performances

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A novel Breeding Blanket (BB) design named Water Lead Ceramic Breeder (WLCB) BB has been developed under the EUROfusion Programme aimed at overcoming the open issues that emerged at the end of the Pre-Conceptual Design (PCD) phase for the driver blanket concepts: Water Cooled Lithium Lead (WCLL) and Helium Cooled Pebble Bed (HCPB).

Such alternative configuration was conceived as a hybrid blanket and with the best trade-off between the two to avoid: 1) current issues in HCPB with shielding, multiplier technology and costs; 2) current issues with T-extraction technology from PbLi in WCLL variants and 3) use of anti-permeation barriers.

In fact, instead of beryllium, alternative n-multipliers as Pb or Pb-alloys have been explored.

The concept has been analyzed and improved here under the neutronic perspective (tritium breeding and shielding functions) performing a large number of neutronic campaigns that have demonstrated the potentiality of the new concept.

In particular, the optimization of: toroidal and radial layouts, cooling plates size and water content, neutron multiplier zoning and materials (Pb, Be12Ti, C, Zr5Pb3), 6Li enrichment, ceramic breeder material, packing factor of ceramic, porosity and First Wall modified design have been carried out to achieve the best results in terms of TBR.

A preliminary design - based on the best balance between neutronic performances, manufacturing and viability considerations, among others - fulfilling the EU DEMO TBR requirement of 1.15, has been delivered for further iterations.

Tentative

03F4

Experimental validation for the qualification of lithium ceramic breeders.

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A key unknown for lithium ceramic-based breeder blankets is whether a practical blanket configuration can be made which will achieve a tritium breeding ratio sufficient to support operation of a fusion power plant. Initial analysis, internally and through literature, suggests that it is theoretically possible but there remain a large number of practical uncertainties which bring doubt to achieving the modelled TBRs. Project VICE (Validation of Ceramic Experiments) by Oxford Sigma, sponsored by the United Kingdom Atomic Energy Authority (UKAEA) Lithium Breeding Tritium Innovation (LIBRTI) Programme, is an experiment that quantifies these uncertainties to address the key unknown and increases confidence in predicting TBR calculations in any fusion tritium-deuterium fuel cycle-based device. Project VICE has the following objectives: 1) demonstration of lithium ceramic manufacture, 2) demonstration of tritium production, 3) demonstration of lithium enriched ceramics, 4) demonstration of tritium recovery, and 5) validation of modelling. Achieving these objectives will aid substantiation of the breeding performance and validation of tritium recovery modelling. The production of sufficient, high quality, and standardised data is required to qualify claims of material breeding performance. The primary objective of VICE is to demonstrate a scalable supply of lithium ceramic pebbles using existing capabilities, building upon existing breeder experiments in manufacturing at Oxford Sigma and mature existing programs on qualification of breeding performance. The program started in January 2025 and will conclude in March 2026 and this talk will provide an update on the program, scientific approach, detailed modelling, experimental plan, and expected data

Invited

03F5

Innovative FCI Design with Al_2O_3-SiC Fiber-Reinforced CMC Composites: Enhanced Thermal Insulation, Maintained Electrical Insulation, and High-Temperature Mechanical Stability

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The flow channel insert (FCI) performs dual functions: providing electrical insulation to mitigate magnetohydrodynamic (MHD) effects, and ensuring thermal insulation to maintain structural steel-PbLi interface temperatures below allowable limits. Various SiC-based materials are under development for FCI applications, including porous SiC bonded between CVD SiC face sheets and SiC/SiC ceramic matrix composites (CMCs). While SiC/SiC composites demonstrate superior mechanical strength compared to porous SiC ceramics, their elevated thermal conductivity (>15 W/m.K) poses limitations in FCI implementations, necessitating further R&D to address this performance trade-off.

Aluminum oxide (Al_2O_3) fibers exhibit exceptional thermal insulation properties owing to their intrinsic low thermal conductivity (<0.15 W/m.K). Building on this characteristic, we propose an innovative FCI design utilizing (Al_2O_3+SiC) fiber-reinforced CMC composites. The 2D plain-woven (Al_2O_3+SiC)f/SiC composites were fabricated via chemical vapor infiltration (CVI) and systematically evaluated through mechanical testing and thermal-physical characterization over the temperature range of RT-700 °C. Incorporation of Al_2O_3 fibers into the textile architecture results in a significant reduction of thermal conductivity (3-5 W/m.K). Meanwhile, the electrical conductivity remains within relative low levels (1.9-66.7 S/m), while the tensile strength demonstrates sustained stability (~300 MPa at RT-700 °C).

Finally, a 100 cm-long prototypical segment of the FCI structure has been successfully fabricated and is currently undergoing testing in a PbLi loop equipped with superconducting coils. This experimental setup aims to quantify the mitigation effect on MHD phenomena and investigate the influence on the flow and heat transfer characteristics of PbLi under operational conditions.

Invited

P1-001

Irradiation Hardening and Microstructural Evolution of K-RAFM Steel Under Fe-Ion Irradiation up to 20 dpa

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The development of Reduced Activation Ferritic/Martensitic (RAFM) steels with superior irradiation resistance is crucial for their deployment in DEMO and future fusion reactors. This study investigates the irradiation hardening behavior of K-RAFM steel, a Ti/Ta-added RAFM variant, subjected to Fe-ion irradiation at 300 °C to a dose of 20 dpa. Nanoindentation and micropillar compression testing were employed to evaluate irradiation-induced changes in mechanical properties, while Transmission Electron Microscopy (TEM) was used to characterize radiation-induced defects. The results indicate that K-RAFM steel exhibits a minimal increase in yield strength post-irradiation compared to Eurofer97 and other RAFM steels, demonstrating exceptional resistance to irradiation hardening. This behavior is attributed to the refined precipitate distribution, where Ti and Ta additions enhance the density and volume fraction of MX-type precipitates, acting as effective sinks for radiation-induced defects. The limited defect accumulation reduces irradiation hardening, making K-RAFM steel a promising candidate for DEMO structural components. Furthermore, the trends observed in yield strength evolution at 2 dpa, 10 dpa, and 20 dpa align well with literature data on RAFM steels, reinforcing the material's suitability for high-dose applications. These findings contribute to the qualification of K-RAFM steel for future fusion environments, supporting its integration into advanced reactor designs.

Invited

P1-002

Effect of Proton Irradiation under Tensile Stress on Irradiation Hardening Behavior of F82H

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Materials used in fusion reactors are expected to be subject to stress loading due to thermal stress and electromagnetic forces. In fact, in the case of W-coated F82H, which is used for the first wall of the blankets, the mismatch between the thermal expansion coefficients of W and F82H causes residual stress to lead cracking. However, we have reported that this residual stress is likely to suppress neutron irradiation hardening [Seki et al., JNM607(2025) 155657]. To verify this, the present study investigates the effects of tensile stress on irradiation hardening behavior of F82H.

2 MeV proton irradiations were performed at ~100 °C up to 0.1 dpa with and without tensile stress with 230 MPa, which is less than 0.2% proof stress (550 MPa at room temperature) of F82H. Irradiation hardening was evaluated by measuring the bulk-equivalent hardness, which was obtained from nanoindentation tests. Comparison of bulk equivalent hardness after irradiations with and without tensile stress indicated that tensile stress suppressed irradiation hardening of F82H. Mechanism of suppression will be discussed.

Tentative

P1-003

On irradiation damage microstructure around grain boundaries in the reduced activation ferritic/martensitic steel: F82H

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F82H, a reduced activation ferritic/martensitic steel, is expected to serve as a structural material for fusion reactors. The effect of sink sites for point defects, represented by grain boundaries, must be experimentally determined to predict irradiation-induced microstructural damage, embrittlement, and hardening in F82H steel. The objective of this study was to experimentally quantify the sink effect.

Thin film specimens of F82H steel were prepared for electron microscopy and irradiated with electron beams up to 2.0 dpa at 400 °C using an HVEM with an acceleration voltage of 1250 kV. After irradiation, the grain boundary orientation was identified using TEM, and the strain at the grain boundary was further evaluated from lattice images obtained near the grain boundary using a high-resolution transmission electron microscope. The denuded zone width near the grain boundary in the irradiation-damaged microstructure was measured, and its relationship with the aforementioned grain boundary parameters was carefully examined.

It was found that a higher areal density ratio corresponded to a larger denuded zone width, and that a smaller local strain spacing, also resulted in a larger denuded zone width.

These results indicate that both the areal density ratio of grain boundaries and the local strain spacing at grain boundaries can be used as indicators of grain boundary character and sink strength against point defects.

Tentative

P1-004

Machine Learning-Based Jiles-Atherton Modeling of Ferrite Materials Under Strong Stray Magnetic Fields in Fusion Applications

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Ferrite cores are extensively utilized in controlled fusion systems, serving as critical components in transformers, Hall current sensors, and other devices. However, due to the inherent magnetic field sensitivity of ferrite, its material properties are significantly influenced by the strong stray magnetic fields generated during the operation of tokamak devices, potentially leading to operational anomalies or failures. Therefore, accurately modeling the hysteresis properties of ferrite materials under strong magnetic fields is of paramount importance. The Jiles-Atherton (J-A) model has been widely adopted for characterizing the hysteresis behavior of ferrite materials. However, existing studies have primarily focused on qualitative explorations of the relationship between J-A model parameters and magnetic field conditions. For the design of ferrite cores in fusion systems, a quantitative understanding of material property parameters under varying magnetic fields is essential. Addressing this gap, this paper proposes a machine learning-based modeling approach for ferrite materials in strong magnetic field environment. Using ferrite magnetic rings as the research subject, a direct quantitative relationship between J-A model parameters and magnetic field parameters is established by leveraging extensive finite element simulation data. The accuracy of the proposed model is validated through experimental testing. This method provides a valuable reference for the design and optimization of ferrite magnetic components in nuclear fusion research, enhancing their reliability and performance in high magnetic field environments.

Tentative

P1-005

Corrosive Effect of Stagnant Molten Lead on EUROFER97 in WLCB Breeding Blanket

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In fusion reactor breeding blankets, neutron multiplier materials play a key role in sustaining the fusion process by providing sufficient neutrons for tritium breeding, enabling energy extraction, and offering radiation shielding. Among all elements, only beryllium and lead are viable neutron multipliers. However, beryllium is toxic and rare, making it impractical for large-scale use.

A promising alternative is the use of stagnant liquid lead confined within EUROFER97, a reduced-activation ferritic/martensitic (RAFM) steel developed for fusion applications. EUROFER97 offers excellent mechanical properties, radiation resistance, and thermal stability, while liquid lead provides efficient neutron multiplication, radiation shielding, and heat removal. However, the interaction between stagnant molten lead and EUROFER97 under high temperature gradients and thermal cycling is not fully understood and presents a critical challenge.

This study focuses on corrosion mechanisms at the steel-lead interface, including material dissolution, protective oxide formation, and corrosion rates. Following exposure tests, the microstructure of EUROFER97 samples will be analyzed, and tensile tests will be conducted to evaluate mechanical degradation. Laboratory results will be used to extrapolate long-term performance under reactor-relevant conditions and to define thresholds for safe operation.

Another issue addressed is thermal cycling through the melting point of lead, which may lead to repeated melting/solidification inside the EUROFER97 container, inducing thermomechanical stresses. Possible damage from the mismatch in thermal expansion coefficients, combined with corrosion effects, will also be assessed.

If these challenges prove to be minor or manageable, the proposed approach could offer a practical and reliable neutron multiplication solution for breeding blanket designs.

Tentative

P1-006

Analysis of Radiation Hardening effect of Ferritic/Martensitic Steel Based on Bayesian Optimization

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Ferritic/martensitic steel is a candidate material for the fusion reactor structural materials, and its irradiation performance is a key performance that restricts its application in nuclear reactors. This article collects 2048 pieces of data on the composition and yield strength of F/M steel, and conducts correlation analysis and screening of important attributes on the data. Using Bayesian optimization BP neural network regression model, the correlation between key components such as Cr, Mo, Si, W and irradiation hardening of F/M steel is constructed, and the performance prediction of irradiation hardening for different components of F/M steel is obtained. Based on correlation analysis and modeling prediction analysis, it is concluded that the irradiation hardening of F/M steel decreases when the Cr content is about 9.0%, the Si addition amount is about 0.8%, and the Mo addition content is higher than 0.25%. At the same time, the addition of W and V has little effect on the irradiation hardening of F/M steel, while the addition of C will exacerbate its irradiation hardening. Subsequently, by analyzing the combined effect of two or three elements on the yield strength change, a design scheme for F/M steel composition with lower irradiation hardening under certain irradiation conditions can be obtained.

Tentative

P1-007

Effect of Irradiation Temperature on EUROFER97 Fracture Toughness after 20 dpa Neutron Irradiation

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EUROFER97 is the European reference structural material for the first wall and blanket of a DEMO fusion reactor. During fusion plant operation, neutron irradiation causes embrittlement and reduces the material's fracture toughness. For the DEMO design, it is important to assess the irradiation embrittlement of EUROFER97 while considering the effect of irradiation temperature to define an appropriate operation temperature window for the material. Additionally, it is crucial to study the material under non-optimal conditions, which may arise during the manufacturing of large components. In this work, we present the fracture toughness results of a technologically heat-treated EUROFER97 steel after 20 dpa neutron irradiation at controlled irradiation temperatures ranging from 207 °C to 345 °C. The neutron irradiation was conducted at the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). Unique miniature 4-notch specimens, with overall dimensions of 45 mm (length) x 3.3 mm (width) x 1.65 mm (thickness), were irradiated and tested. The testing and analysis followed the Master Curve method in accordance with the ASTM E1921 standard. The degree of embrittlement in EUROFER97, measured by the increase in the Master Curve reference temperature, T0, showed a strong dependence on irradiation temperature, with higher irradiation temperatures resulting in a smaller increase in T0. For example, the largest increase in T0, measured at 228 °C, was observed after irradiation at 207 °C. In contrast, the smallest increase in T0, estimated at 5 °C based on ongoing testing results, was observed after irradiation at 345°C.

Tentative

P1-008

Irradiation Hardening in Advanced Reduced Activation Ferritic-Martensitic Steels for Future Fusion Applications

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Reduced activation ferritic-martensitic (RAFM) steels are a recent class of radiation-resistant steels designed to be used as the structural components of power-producing fusion reactors. In this work an advanced RAFM steels has been developed with superior radiation resistance with respect to the EUROFER-97 upon which was based.

Self-ion irradiation campaigns up to 100 dpa at 350 °C show an increase in hardness of only 20%. The work hardening response, as determined from spherical nanoindentation, is unchanged between the as-received state and irradiation to 100 dpa, implying that the alloy should retain modest ductility under these conditions. Proton irradiations at 250 °C, 350 °C, and 400 °C demonstrate that the low temperature hardening embrittlement threshold is largely unaffected, increasing by only ~50°C with respect to EUROFER-97.

A refinement of alloy chemistry and a subsequent modification of the thermomechanical treatments to favour MX precipitates is therefore a very promising strategy for the further development of fusion steels.

Tentative

P1-009

Direct experimental measurement of the solubility of hydrogen isotopes in the Eurofer-97 RAFM

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Accurate knowledge of hydrogen isotopes transport properties in structural materials of interest for Breeding Blankets (BB) and Tritium Extraction and Recovery (TER) systems is fundamental for experiments, modelling, safety and design activities. This work aims at measuring these transport properties in Eurofer-97 at laboratory conditions, providing a reliable measure, at relevant operational conditions, on which a degree of uncertainty exists. Not only permeability (P) and diffusivity (D), but also solubility and trapping of hydrogen isotopes in fusion materials must be unambiguously defined in order to assess release through walls and breeding blankets and retention, which can also lead to hydrogen embrittlement.

The protium/deuterium solubility, expressed in terms of the effective Sieverts´ constant (KSeff), is usually calculated indirectly from the permeability and diffusivity parameters obtained from permeation experiments, assuming that the Sieverts´ law can be applied and that a Diffusion Limited Regime (DLR) takes place. This is the case for the main references for protium and deuterium solubility in Eurofer-97.

A possible way to validate values for the solubility of hydrogen isotopes is to avoid indirect measurements, as the permeation technique, and use methods that measure solubility directly. This is the case of the absorption-desorption technique, where the direct measurement is the solubility itself. Some measurements have already been published in the past using this technique with similar steels using the same facility, including analysis of trapping phenomena. This works shows the experimental results for the solubility of protium/deuterium in Eurofer-97, obtained during a direct absorption-desorption measurement campaign.

Tentative

P1-010

Progress of Engineering Qualification on CLAM and Welding Wire for ITER-TBM in China

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China Low Activated Martensite (CLAM) steel has been chosen as one of the primary candidate structural materials for ITER-TBM in China. Engineering Qualification (EQ) is a key step for the application of CLAM and welding wire, and its technical requirements were formulated referring to the EN/ISO standards. 3×5.2 ton ingots of CLAM were smelted by vacuum induction melting (VIM) and vacuum arc remelting for its EQ. The results of conventional mechanical properties meet the technical requirements, the creep tests accumulate over 120000 h at 400-600 °C and 1000000 cycle fatigue tests at RT are completed. The longer time creep for 400000 h and high temperature fatigue tests are ongoing. In addition, the CLAM wire for Tungsten Inert Gas Welding (TIG) was designed according to the matching requirements on low-activation composition, irradiation-resistance microstructure and mechanical properties of joints. 5×300 kg wire ingots were smelted by VIM and electro slag remelting. The manual and automatic welding wires with 2.4 mm and 1.2 mm diameters respectively were prepared by forging and drawing processes. Welding qualification of TIG include All-Weld Metal Test (AWMT) and standardized plate test were performed with the suitable welding procedure specification. The longitudinal tensile strength/elongation of AWMT were over 650 MPa/17% at RT and 400 MPa/20% at 550 °C, the impact energies at RT and -20 °C were above 230 J and 160 J respectively. The EQ testing results of CLAM and its wire meet the technical requirements, which provides important foundation for ITER-TBM application of CLAM in China.

Tentative

P1-011

Corrosion behavior and mechanical properties of low activation ferritic martensitic steel and its welded joints in liquid lithium lead

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This study presents the corrosion behavior and evolution of mechanical properties of China Low Activation Martensitic (CLAM) steel and its laser welded joints under high-temperature flowing liquid lithium lead (Pb-17Li) for up to 6000 h. The corrosion morphology and element distribution were analyzed by scanning electron microscopy (SEM) and energy dispersive spectroscopy (EDS). The corrosion products were analyzed by X-ray diffractometer (XRD). The mechanical properties of samples pre- and post-corrosion were evaluated using an electronic universal testing machine and a pendulum impact testing apparatus. The results indicate that the surface of CLAM steel samples begins to exhibit significant corrosion after varying durations of exposure to Pb-17Li flowing at 500 °C. This is primarily attributed to physical corrosion and mechanical erosion. In comparison to the base material, laser welded components of CLAM steel exhibit a greater reduction in plasticity and decreased impact toughness in liquid Pb-17Li. This is attributed to an increase in slab martensite within the weld zone and the precipitation of carbides at the grain boundaries, leading to alterations in internal stresses and lattice defects. Consequently, the region near the surface becomes more brittle, resulting in a degradation of mechanical properties.

Tentative

P1-013

Reduced activation 9Cr-ODS steels with EUROFER97-like Charpy properties and industrialization efforts

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With the advanced blanket of EU-DEMO, the goal is to demonstrate a long-living operation. To circumvent premature embrittlement of the structural steel by accumulated displacement damage and helium transmutation, an operating temperature of at least 650 °C and high resistance (100 dpa) to accumulated helium are required. These operational requirements rule out the use of RAFM steels due to softening and vulnerability to helium embrittlement. In the framework of the EUROfusion WPMAT program, therefore, oxide dispersion strengthened (ODS) 9Cr-steels were developed and produced. Industrialization of production has been initiated.

In the first part, we prove by means of Charpy tests the feasibility to produce 9Cr-ODS steel plates with a ductile-to-brittle transition temperature (-90°C) comparable to the non-ODS RAFM steel EUROFER97 and even slightly superior upper shelf energy. Fracture toughness tests show furthermore excellent K1c results. Progress in upscaling is described. In the second part, these encouraging results are discussed in the context of alterations of (1) mechanical alloying process parameters that were applied in the course of production of several powder batches and (2) the alloys' chemical compositions. Here we show that increasing the yttrium content beyond the commonly accepted limits significantly improves the impact properties. This counter-intuitive behavior is explained by a more effective stabilization of the grain structure. While ODS steels that are described in literature frequently suffer from strong bimodality, our alloys exhibit a homogeneous microstructure. Our hypothesis is that the suppression of extensive areas of nanoscale grains is a prerequisite for excellent impact properties in ODS steels.

Invitation

P1-014

High Throughput Analysis of Thermally Aged Reduced-Activation Ferritic-Martensitic Steels for Future Fusion Applications

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Reduced activation ferritic-martensitic (RAFM) steels are a class of radiation-resistant steels designed for use as the structural components of fusion reactors. RAFM steels are operationally limited to 550 °C; above this, they are susceptible to grain boundary unpinning, phase instability, and loss of creep strength. This is attributed to the coarsening of M23C6 precipitates at high temperature. Advanced RAFM (ARAFM) steels are therefore being developed to operate above 550°C to increase the thermodynamic efficiency of the power plant.

The key UK consortium in this area is the NEURONE (NEUtron iRradiatiOn of advaNced stEels) programme, which seeks to produce a microstructure strengthened by a fine dispersion of MX-type precipitates by optimising alloy chemistry and thermomechanical treatment (TMT). Tungsten is also utilised as a solution strengthener to increase high temperature strength.

In this work, we present data on ARAFM steels that have been thermally aged between 600-700 °C for up to one year. The microstructural characterisation highlights the complex interplay between initial tungsten content and the volume fraction of precipitating Laves phases. High throughput nanoindentation tests have been developed to screen alloys and identify changes in key mechanical properties as a function of ageing temperature and time. These findings have been used to provide feedback to the alloy design process to refine chemical and TMT adjustments, with the aim of establishing an optimal tungsten content to offer long-term thermodynamic stability and sufficient high temperature strength.

Invitation

P1-015

Experimental challenges for the determination of ductile fracture toughness of F/M steels

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The qualification of a material for use in a nuclear reactor requires several steps. According to the RRC MRX code, the first step is the preparation of a so-called "material file". For EUROFER97 steel, the preparation of the material file started in 2000 under the auspices of AFCEN. The Material file includes: physical properties; basic mechanical data; boundary conditions; mechanical properties where creep is significant and mechanical properties where irradiation is significant. Basic mechanical data deals with time-independent properties such as tensile, impact, fatigue and fracture. Ductile fracture toughness of EUROFER97 is already included in the RCC-MRx Probationary Phase Rule, but more data is needed, especially at high temperatures, to assess the performance of thermal blanket modules. In this paper, the J-R crack propagation resistance curves are determined for EUROFER97 and T91 steels by applying the ASTM E1820 test standard, from room temperature up to 550 °C. The experimental challenges associated with determining crack growth during high temperature testing and the comparison of different methods are discussed.

This work was partly supported by the European community within the FP7 project MATTER under Grant Agreement No. 269706 and within the EUROfusion Consortium Grant Agreement No 101052200

Tentative

P1-016

Corrosion resistance of High-Mn Oxide-Dispersion Strengthened (ODS) Alumina-Forming Austenitic (AFA) Steels in Liquid Metal Sn

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Liquid Metal tin (Sn) is one of the candidate coolants of liquid metal divertor concept in fusion reactors because of its excellent thermal properties such as low vapor pressure, low melting point, and high thermal conductivity. However, the chemical compatibility of liquid Sn with structural materials is a critical issue to be addressed compared to the other liquid-metal environments like Pb-Li alloys, because the liquid Sn is highly corrosive and causes liquid-metal embrittlement. Advanced structural materials with high corrosion resistance are essential for the liquid Sn environment. To meet these requirements, we have developed novel high-Mn oxide dispersion-strengthened (ODS) steels with additions of several wt.% of Al and Si, resulting in excellent high-temperature oxidation resistance and mechanical properties at room and high temperatures. In this study, we performed corrosion tests of high-Mn ODS steels in static liquid Sn at 500 °C. The results showed that pre-oxidized samples containing an alumina layer on the subsurface demonstrate dexcellent corrosion resistance than as-received samples. The pre-oxidized surface was still visible on pre-oxidized samples, the well-polished can not be observed on as-received samples. There was less Sn remained on the pre-oxidized sample surface, indicating less intermetallic compounds formation. The detailed microstructural analysis will be given.

Tentative

P1-017

Possibility of fabricating ODS high entropy alloys by using cold spray method

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Research on oxide dispersion strengthened high-entropy alloys (ODS-HEAs) is being conducted to develop next-generation reactors including fusion reactors. ODS-HEAs are typically fabricated using the mechanical alloying (MA) method; however, this process has drawbacks, including prolonged manufacturing time and microstructural changes due to repeated processing and heat treatments. To address these issues, this study explores the cold spray (CS) method, in which powder particles are deposited on a substrate by high-velocity impact. This impact may serve as a substitute for the enforced solid solution or amorphization of oxide particles observed in the MA method. The feasibility of CS as an alternative to MA was evaluated by analyzing microstructural changes before and after CS processing.

Gas-atomized FCC-type HEA powder ($Cr_{0.8}FeMn_{1.3}Ni_{1.3}$) with 0.3 wt% Y was deposited onto an SUS304 substrate using the CS method. Nitrogen was used as the process gas, with temperatures ranging from 550 to 1000 °C and pressures of 5-6 MPa. Subsequent heat treatment at 1000 °C was conducted to induce re-precipitation. XRD was performed to identify crystal structures and calculate dislocation densities, while TEM was used for microstructural evaluation.

The gas-atomized powder initially contained an FCC matrix and a secondary phase, later identified as $Mn_{0.5}Ni_{2.5}Y$ precipitates (~100 nm). After CS processing, secondary phase peaks disappeared, and dislocation density increased to a level comparable to MA-fabricated materials. This disappearance is attributed to dissolution or amorphization caused by plastic deformation upon impact. These findings suggest that CS has potential as an alternative fabrication method for ODS-HEAs.

Invitation

P1-018

Fabrication and Characterization of Oxide Dispersion Strengthened Alloys with directly introduced $Y_2Ti_2O_7$ Pyrochlore Oxides for Fusion Applications

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The core structural materials for advanced nuclear/fusion systems are considered to operate under higher temperatures and neutron irradiation than commercial nuclear systems. To realize these systems, it is essential to develop advanced structural materials having high creep and irradiation resistance. Oxide dispersion strengthening (ODS) is one of the most prospective ways to improve properties at elevated temperatures in many alloy systems. This is mostly attributed to uniformly distributed nano-oxide particles with a high number density. ODS alloys are normally produced by a process of powder metallurgy. The Y_2O_3 and Ti(or alloyed Ti) powders are milled together with a metallic powder by mechanical alloying. Subsequent powder densification process leads to the formation of nm-scaled precipitates, mainly of composition $Y_2Ti_2O_7$. However, in addition to the nanometric precipitation, these materials contain a population of micrometric precipitates of various compositions, in particular silicon or aluminum oxides. It has been proved that in the Si or Al-containing ODS alloys, Ti is unable to form fine Y-Ti-O complex oxide due to the formation of Si or Al oxide is easier than that of Ti oxide. These coarse precipitates are undesirable and lead to a drastic decrease in mechanical properties. In this study, various ODS alloys were fabricated and characterized using a direct introduction $Y_2Ti_2O_7$ complex oxides. $Y_2Ti_2O_7$ powders were synthesized by mechanical milling and subsequent heat-treatment. The objective of this work is to investigate the feasibility of fabricating ODS alloys with a direct introduction $Y_2Ti_2O_7$.

Invitation

P1-019

Impact of Mesoscale Heterogeneous Microstructure on Tensile Properties and Deformation Behavior of ODS Ferritic Alloys

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Iron-based ODS alloys are expected to be used as structural materials for fusion reactor blankets in the DEMO and beyond. Although ODS alloys exhibit excellent strength, their low ductility remains an issue to enhance their toughness. Therefore, this study examines a mesoscale heterogeneous microstructure to ODS ferritic alloys in order to achieve both high strength and ductility through the interaction between hard and soft grains.

Fe powder with and without 1 mass% Y_2O_3 powder were processed using a planetary ball mill (300 rpm, 48 h) to obtain fine Fe powder and Fe-1Y_2O_3 alloy powder (ODS powder), respectively. These two powders were mixed in different mass ratios to obtain five heterogeneous powder mixtures: ODSx% (x = 100, 75, 50 ,25, 0). These powders were sintered by pulse electric current sintering at 1050 °C and 50 MPa, followed by hot rolling at 1050 °C with a rolling ratio of 67%, and finally water-quenched after heat treatment at 1050 °C for 2 hours. Tensile specimens were fabricated using electrical discharge machining of SS-J type miniature specimens, and then tested at room temperature with a strain rate of 6.67×10^{-4} /s.

The tensile test results showed that the alloys produced from the mixed powders overcame the trade-off between ODS0% alloy and ODS100% alloy. In particular, the ODS25% alloy exhibited highest UTS due to its enhanced work-hardening behavior. The effect of heterogeneous distribution of Y_2O_3, which is observed by electron-probe micro-analyzer, on the deformation behavior will be discussed.

Tentative

P1-020

Understanding of the high temperature creep resistance of ODS low activation F/M steels with carbon free design

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From a viewpoint of engineering, the high temperature creep performance is important for a long-term safe service of ODS steels application for future fusion reactors. Although some reports have confirmed the excellent creep properties of ODS steel. There are still some challenges: (1) non-standard samples and testing methods are often employed due to limited materials, complicating to the comparison of experimental results from different reports; (2) detailed creep data at different temperatures are still limited, its relationship with composition design and microstructure evolution is still unclear.

This work investigates the creep behavior of ODS low activation steels with carbon free design prepared in hundred kg level by MA technology at temperature range between 650 °C to 950 °C using a standard round rod sample with a length of 74 mm. The microstructure evolution during creep test is characterized. The relationship between composition design and creep performance of ODS steels is discussed. The 9Cr2W ODS steel shows the most excellent creep resistance and a much longer lifetime compared to the other steels at all experimental temperature and applied stresses. For ODS steels, it is very important to determine the threshold stress for engineering design. The steady-state creep rate at 650 °C-150 MPa is as low as 1.5×10^{-7} h^{-1}. At 850 °C-100 MPa, the steady-state creep rate is just in the level of 10^{-6} h^{-1}. More ultra-fine fine oxide particles precipitate in the grains and some intermetallic compounds precipitates along the grain boundary which contribute to the creep property.

Invitation

P1-021

Nanoscale insights into 9Cr-ODS steels

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Achieving optimal performance in future fusion reactors depends on a well-designed configuration. Alongside the extensive use of structural steels like EUROFER97, oxide-dispersion strengthened (ODS) steels can be employed in components exposed to significant heat and neutron radiation, offering high resistance to neutron radiation, improved high-temperature strength, and superior creep properties. Large scale ODS production involves ball milling of large quantities of metal and oxide powders. The choice of the milling atmosphere might improve the ODS particle distribution and the contamination control. A batch of four powder samples was prepared for this study: two were conventionally milled in an argon (Ar) atmosphere, and two were milled in a hydrogen (H_2) atmosphere. All samples were subjected to hot isostatic pressing (HIP) at 1150 °C, followed by rolling at 980°C. Microstructural characterization was performed using scanning electron microscopy (SEM) and transmission electron microscopy (TEM). Specifically, electron backscatter diffraction (EBSD) maps were acquired via SEM to extract matrix grain sizes, which were used to contextualize the TEM results, while energy-dispersive X-ray spectroscopy (EDX) and electron energy-loss spectroscopy (EELS) were conducted using TEM to analyze ODS particle sizes, particle densities, and composition. High-resolution phase-contrast imaging was additionally employed to examine the structure of individual ODS particles. Furthermore, the particle sizes, densities, and composition of carbides and nanosized Ar gas bubbles were also investigated. The microstructural findings will be compared with tensile and Charpy impact test data to evaluate whether milling in a H_2 atmosphere offers advantages over milling in an Ar atmosphere.

Tentative

P1-022

Retardation of Phase Transformation by Oxide Particles in Oxide Dispersion Strengthened (ODS) Steels

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Oxide dispersion-strengthened (ODS) steel, with its excellent high-temperature strength and radiation resistance, is considered a promising candidate for fusion reactor structural materials. The finely dispersed oxide particles in ODS steel play a crucial role in phase transformation, particularly in delaying the α/γ phase transition. This study focuses on 11CrODS steel, quantitatively analyzing the pinning effect of oxide particles on phase boundary migration and systematically elucidating the mechanism behind the delayed α/γ phase transition in this alloy.

Five experimental samples of 11CrODS steel with different oxide particle spatial distributions were prepared through vacuum annealing at 1250 °C. High-temperature X-ray diffraction (XRD) was used to in-situ track the α/γ phase transition behavior of the samples. The phase equilibrium of the Fe-Cr-Ni-Ti-C system and the free energy difference between the α and γ phases were calculated by ThermoCalc software. Transmission electron microscopy (TEM) was applied for multi-scale characterization (the size distribution, spatial configuration, and other characteristics) of oxide particles.

TEM characterization shows that the Y-Ti-O nanoparticles are formed in the steel matrix. These highly stable oxide particles significantly suppress α/γ phase boundary migration through the Zener pinning effect, raising the phase transition start temperature by approximately 67 °C compared to the system without particles. However, as the annealing time exceeds 90 hours, the average particle size increases from the initial 4.1 nm to 7.1 nm due to Ostwald ripening, and the pinning effect on the α/γ phase boundary is reduced by approximately 75%. This decreases phase transition temperature accordingly.

Tentative

P1-023

Design of novel low activation intermetallic reinforced nano-structured steels for fusion

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Ferritic superalloys and high entropy alloys (HEA) are novel materials being developed for fusion and Gen-IV fission to allow for higher operating temperatures, due to intermetallic reinforcement, alongside enhanced radiation tolerance owing to high sink strength. In this work, three Fe-Ni-Al-Cr-Ti ferritic high entropy superalloys were designed with a focus to replace nickel with reduced activation elements, namely Mn and V, and so to assess the suitability of such alloys toward fusion applications. Alloys were produced by arc-melting and characterised using SEM, EBSD, XRD and hardness testing. In the vanadium-containing alloy a microstructure of disordered A2 BCC matrix with B2-FeAl precipitates was successfully produced. While the manganese-containing alloys showed cuboidal L21 (Fe_2AlTi) nano-precipitates, increased Mn content led to the formation of cubic-A13 (β-Mn) and χ-phase in samples aged at 700/850 °C. With this study, it was possible to achieve low activation A2 - B2/L21 ferritic superalloy microstructures within a compositionally complex (HEA) matrix, successfully removing Ni, showcasing their potential application in fusion energy.

Tentative

P1-024

Effect of Secondary Phases and Solute Additions on Bonding and Thermal Transport at W/Cu Interfaces

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Under ORNL's Fusion Foundational Materials Program, the design criteria dictating alloy development for fusion reactors have been broadened to include the effects that alloying elements and altered microstructures may have on the strength of crucial interfaces between components. For example, bond cohesion, resistance to cracking, and efficient heat transfer are crucial properties for the interfaces between heat exchanger and structural components of divertors in nuclear fusion reactors. However, limited work has been conducted on the effect of relevant alloying elements on the cohesion and characteristics of these interfaces. Several Cu alloys have been studied as candidates for fusion reactor heat exchanger components, with Cu-Cr-Zr (CCZ) chosen for the ITER research reactor; alloys such as Cu-Cr-Nb-Zr (CCNZ) have been developed to improve further upon CCZ's creep properties. Sintered samples of pure W layered with CCNZ were produced at Stony Brook University. Currently, full characterization of these samples is underway. The thermal transport across interfaces and micromechanical properties will be measured for these samples as well as pure W/Cu and W/Glidcop bilayer specimens produced by Nippon Tungsten. Diffusion bond width, phase formation, coherency relationships between secondary phases and W at the interface, and the related cracking behavior will be characterized through scanning and transmission electron microscopy. The results will improve our understanding of how the use of Cu alloys, in place of pure Cu, will affect the critical Cu/W interfaces in fusion divertor components.

Tentative

P1-025

Effect of helium bubbles on hydrogen retention in tungsten studied by STEM-EELS

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Tungsten is planned to be used as a plasma facing material and will be exposed to the burning plasma with high density hydrogen isotopes and helium simultaneously. It has been reported that helium atoms in tungsten precipitate as helium bubbles and act as deuterium trap site, although the direct evidence is still scarce. In this study, the effects of helium bubbles on hydrogen isotopes retention in tungsten were precisely evaluated with an aberration-corrected scanning transmission electron microscope combined with electron energy-loss spectrometer (STEM-EELS).

The material used for this study is powder metallurgy tungsten by Nilaco Co, which was processed into a disk shape (φ3mm), heat-treated, and thinned. The samples were pre-irradiated with helium ions at ~1523 K, followed by deuterium post-irradiation at room temperature. The positions and densities of deuterium and helium atoms in the vicinity of bubbles were investigated under isochronal annealing from R.T. to 673 K.

STEM-EELS analysis revealed the density changes of deuterium and helium atoms in the bubbles due to the irradiation and the annealing. While the helium density in the bubbles decreased after deuterium irradiation, it conversely increased under the subsequent annealing. This suggests that the helium atoms which were released from the bubble and trapped in the defects in the matrix were re-trapped in bubbles under the annealing. The correlation between the deuterium and helium densities in individual bubbles will be also presented in the conference.

Tentative

P1-026

High heat flux testing of wire-based laser metal deposition coated plasma-facing components

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The high particle fluxes impinging on plasma-facing components (PFCs) in future fusion power plants cause inevitable erosion of their armor. In situ repair of PFCs by local deposition of new material would open up the possibility of damage healing and compensation of erosion losses. The wire-based laser metal deposition (LMD-w) process fulfils the necessary requirements for use in the reactor vessel. Process development for the deposition of tungsten, which is the primary candidate as plasma-facing material (PFM), on tungsten substrate has already been done. Furthermore, tests on small scale W-blocks had proven that damage caused by transient heat loads (ELM-like) on the PFM can be healed this way.

Test components consisting of tungsten tiles with surface areas of 28×12 mm^(2) were brazed to copper cooling structures and used as demonstrators for PFCs. These tiles were coated using LMD-w with one respectively two stacked layers, each with heights of ~0.65 mm. Some coated surfaces were smoothed by laser re-melting. Afterwards, the components were exposed to up to 1000 stationary heat load cycles as well as combined stationary and transient thermal loads in the electron beam facility JUDITH 2. The chosen stationary thermal loads, calculated by FEM simulations, resulted in surface temperatures equivalent to heat fluxes of 10 MW m^(-2) and 15 MW m^(-2) on standard monoblock chains. Additionally, some layers were subjected to combined stationary and transient loads (0.13-0.55 GW m^(-2), 10^(3)-10^(5) pulses of 0.5 ms, 200-700 °C base-temperature). After these HHF experiments, the components were subjected to various examinations.

Tentative

P1-027

Trapping and Diffusion Profile of Hydrogen Isotopes for Damaged Tungsten-Tantalum alloy

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Plasma facing components, such as the divertor and first wall in fusion reactor, will be exposed to high-heat flux and energetic particles, which lead to tritium (T) accumulation and degradation of mechanical properties. Tungsten alloys, known for their excellent thermal and mechanical properties, are considered as one of the promising materials to endure such extreme environments. In this study, focusing on tungsten-tantalum (W-Ta) alloy, the trapping and diffusion processes of hydrogen isotopes were investigated.

In this experiment, polycrystalline W and W-1,3,5wt%Ta samples were prepared using powder metallurgy and hot rolling, with 6 mm in diameter and 0.5 mm in thickness. These samples were fabricated by A.L.M.T Co, Ltd. Surface damage was introduced by irradiating the samples with Fe^{2+} up to 1 dpa at R.T. at TIARA facility. Positron Annihilation Spectrometry (PAS) was employed to evaluate the size and density of irradiation-induced defects, such as vacancy-type defects. Deuterium (D) ion implantation and T gas exposure were performed to introduce hydrogen isotopes into the damaged samples. Results from Glow-Discharge Optical Emission Spectrometry following D implantation revealed that the implanted D diffused into the bulk was enhanced by the damage. Furthermore, the dissolution of hydrogen was suppressed when Ta concentration exceeded 3%. Imaging Plat analysis after T gas exposure showed the different T distribution, suggesting that the solute Ta migrated on the surface, leading to variations in Ta concentration, and change in T absorption behavior. The findings from PAS and additional analysis of hydrogen isotopes behavior will be further discussed in the presentation.

Tentative

P1-028

Development of potassium doped tungsten thick plates with optimized mechanical properties by microstructure control.

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Since longer time operation with neutron irradiation is expected in fusion reactors after ITER, tungsten materials with microstructural stability, improved mechanical properties, resistance to high heat flux, and tolerance against irradiation embrittlement have been developed making full use of the microstructure control technologies. It has been demonstrated in effectiveness of potassium doped tungsten. The rolled thick plates of potassium doped tungsten was manufactured by powder-metallurgy utilizing our manufacturing experience of ITER divertor monoblocks. The manufacturing procedure is as follows: sintering a potassium doped tungsten ingot, hot rolling, and heat treatment of stress relieving after the rolling. Evaluation of the mechanical properties and recrystallization behavior was conducted in case of as-rolled and after the heat treatment. From these results we confirmed that the developed material was improved by potassium doping in terms of recrystallization resistance or mechanical properties. We will continue to make further improvements. This presentation will show the effectiveness of developed material including details of cross-sectional observation, mechanical properties, and recrystallization behavior.

Tentative

P1-029

From Lab-Scale to Industry: Advanced CVI Strategies for Wf/W Composite Fabrication

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Tungsten fiber-reinforced tungsten (Wf/W) is a composite material with significant potential for application in nuclear fusion reactors. This potential stems from its enhanced damage tolerance and thermomechanical stability. While its effectiveness has been demonstrated at the laboratory scale, the transition to industrial production remains a critical challenge.

To date, component-relevant dimensions have only been achieved by powder metallurgy and layer-by-layer chemical vapor deposition (LBL-CVD). However, both approaches involve substantial technical and economic limitations.

This work proposes an advanced chemical vapor infiltration (CVI) method, building upon a comprehensive understanding of the reaction kinetics between precursor materials. The method was meticulously developed for advanced tungsten textiles and utilizes three-dimensional temperature gradients, a localized heat source, and precisely controlled flow conditions.

In contrast to LBL-CVD, which processes only one layer per day, the proposed CVI method enables the simultaneous deposition of over 200 layers in a single processing step. The resulting materials achieve a fiber volume fraction of 56% and a relative density of up to 97%.

The scalability of this novel method is constrained exclusively by the dimensions of the reactor chamber and the available surface area of the localized heat source. The resulting materials exhibit cyclic mechanical strength with extrapolatable fatigue behavior, thereby outperforming current industrial standards.

These results represent a significant milestone towards the industrial-scale production of Wf/W composites and introduce a scalable and potentially economically viable route to produce high performance materials in fusion applications. Future research will focus on optimizing infiltration times and evaluating performance under fusion-relevant conditions.

Tentative

P1-030

Development of Novel WO_3-doped ODS-Cu Alloys Towards Industrial-Scale Production

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Copper alloys are recognized as candidate for heat sink materials of divertor due to the high thermal conductivity, especially oxide dispersion strengthened copper (ODS-Cu) alloys which are known for irradiation resistance and great mechanical properties at high temperature. ODS-Cu alloys made by mechanical alloying (MA) have been proved to enhance the mechanical properties by forming refined Y-Zr complex oxide particles. Nevertheless, the fabrication of ODS-Cu meets challenges on the coarsening of MAed powders and nonuniform distribution of oxide particles, which is mainly caused by the high ductility of copper. Process control agent (PCA) has been widely used in powder metallurgy to obtain finer powders, while previous study has demonstrated that the addition of stearic acid as PCA leads to high amount of pores by thermal decomposition in sintering. In order to achieve mass-production of ODS-Cu alloys, WO_3 will be used as carbon-free PCA to fabricate ODS-Cu in this research.

Gas-atomized Cu-Y-Zr powder, gas-atomized Cu-Zr powder, pure copper powder and WO_3 powder were mechanically alloyed at argon atmosphere by ball milling machine Emax with water cooling system for 12 hours. The MAed powder was sintered at 880 °C with a load of 50 MPa by spark plasma sintering.

As for results, WO_3 addition dramatically enhances the production rate of fine MA powders up to nearly 100%. X-ray diffraction results indicate that WO_3 can decompose in MA process, providing oxygen source for the internal oxidation of Y and Zr. Mechanical characterization and microstructural observation of the WO_3-doped ODS-Cu alloys will be shown.

Tentative

P1-031

Comparative Study on Microstructural Response of ITER-Grade Tungsten Under Fusion-Relevant Irradiation Conditions

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Tungsten and tungsten alloys are the leading candidate materials for plasma-facing components in fusion technology due to their superior thermal properties. However, neutron irradiation induces defects that damage the microstructure and degrade mechanical properties. Understanding tungsten's response to neutron irradiation is crucial for assessing its performance under fusion-relevant conditions.

This study investigates the microstructural evolution and hardening mechanisms in ITER-grade tungsten exposed to various radiation conditions, including heavy-ion and neutron irradiation at low (25-300 °C) and elevated temperatures (600-1200 °C) to doses up to ~1 dpa. Transmission Electron Microscopy (TEM) analysis revealed the formation of dislocation loops, nanovoids, and transmutation products, whose characteristics were analyzed. The measured defect characteristics highlight the effects of (i) radiation type, (ii) radiation temperature, and (iii) radiation dose. Further examination using in-situ TEM is applied to compare the effects of annealing and high-temperature irradiation on the thermal stability and recovery mechanisms of radiation-induced defects. Microstructural characteristics are linked to radiation-induced hardening using the dispersed barrier model and compared with measured mechanical properties.

The findings provide an experimental dataset on the microstructural characteristics, defect behavior, and recovery mechanisms of tungsten irradiated under various fusion-relevant conditions. Additionally, the comparison of different irradiation conditions offers valuable insights for future tungsten studies.

Tentative

P1-032

Mesoscale Modeling of First Wall and Blanket Materials Under Fusion Reactor Conditions

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The structural integrity of first wall and blanket components in fusion reactors is critical for ensuring long-term operation under extreme thermomechanical conditions. These materials undergo complex microstructural evolution, including the formation of dislocation loops and interactions with precipitates, which significantly affect their mechanical response. To capture these effects, we perform mesoscale simulations using MoDELib (Mechanics of Defects Evolution Library), modeling defect dynamics in crystalline materials. These simulations provide insights into the evolution of key microstructural properties, such as glissile dislocation densities as a function of plastic strain rate, which are essential for understanding deformation behavior at this scale. The extracted parameters are used to calibrate the Kocks-Mecking hardening model, which is then integrated into MOOSE, an open-source finite element framework for component-scale multiphysics simulations. By linking mesoscale defect behavior with macroscopic material performance, this multiscale modeling approach enhances predictive capabilities for fusion reactor materials. The results contribute to a deeper understanding of deformation mechanisms under fusion-relevant conditions, aiding in the development of more resilient structural materials. This work supports the broader goal of improving the reliability and performance of fusion energy systems by providing a more accurate representation of material behavior across different length scales.

Tentative

P1-033

Wet chemical surface pre-activation of tungsten powders for additive manufacturing

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In recent years, laser powder bed fusion (LPBF) of pure tungsten has garnered significant attention for applications in thermonuclear fusion and other fields. However, the high ductile-brittle transition temperature of tungsten makes the formation of micro-cracks during LPBF processing seemingly inevitable. Recent studies have demonstrated that introducing nano-scale surface textures on metal powders can significantly enhance laser absorption efficiency, thereby providing a viable solution for achieving fully dense pure tungsten components via LPBF at low energy densities. In this study, we investigate the impact of powder surface pre-activation techniques on the formability of LPBF pure tungsten.High-purity tungsten powder (5-25 μm) was procured from Avimetal AM Tech Co., Ltd. The powder was subjected to surface activation using 40 wt.% HF solution and 30 wt.% H_2O_2 solution, respectively. Following treatments, the activated powders were characterized in terms of particle size distribution, flowability, and laser reflectivity using a laser particle size analyzer, a Hall flow meter, and a UV-Vis-NIR integrating sphere spectrophotometer. The surface activity of the powders was further evaluated through spark plasma sintering (SPS). The results indicate that the particle size of the original, acid-etched, and alkali-etched powders remained largely unchanged. However, the laser reflectivity at a wavelength of 1060 nm was significantly reduced, with values of 30.68%, 30.18%, and 27.64%, respectively. Additionally, the relative density of the activated samples increased notably. In this presentation, we discuss in detail the influence of powder pre-activation technology on formability and its potential to enhance the performance of LPBF pure tungsten components.

Tentative

P1-034

Enhancing Fracture Toughness of Tungsten via W-Cr Composite with Tungsten Heavy Alloy Structure for Plasma-Facing Components

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Tungsten is a promising material for plasma-facing components (PFCs) in nuclear fusion reactors due to its high hardness, high melting point, high thermal conductivity, and low sputter yield. However, its low fracture toughness at low temperatures makes it susceptible to brittle fracture, confining the tungsten only suitable to high temperature application. Ni-based tungsten heavy alloy has been widely adopted to overcome the tungsten's innate brittleness. However, utilizing the tungsten heavy alloy for fusion application is not desirable, because the Ni content in tungsten heavy alloy produces radioactive wastes with long half-life in the neutron-incident environment. In this study, Cr, which produces radioactive wastes with short half-life, was selected as a replacement of Ni in tungsten heavy alloy structure. In order to fabricate the structure, the W-Cr core-shell powder was synthesized via high-energy milling, and the powder was densified by spark plasma sintering (SPS). Liquid sintering method commonly used for the fabrication of conventional Ni-based tungsten heavy alloy was unavailable, since tungsten and chromium form solid solution at elevated temperatures destroying the structure of the composite. The fabricated W-Cr composite showed enhanced fracture toughness, which indicates the possibility that the W-Cr composite hinders the propagation of the crack initiated from the plasma-facing surface of the divertor monoblock. Additionally, various properties were evaluated across different temperatures to verify the material's feasibility as a PFC. The results demonstrate that the designed W-Cr composite is a strong candidate for fusion applications, especially for the parts exposed to the low temperature range.

Tentative

P1-035

A new crack self-healing mechanism in additive manufactured tungsten alloy for nuclear fusion

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Tungsten is a critical material for plasma-facing components in nuclear fusion reactors due to its excellent properties including high melting point, high thermal conductivity, low tritium(T) retention and low sputtering rate in extreme fusion environment. Research on tungsten and tungsten alloys via Laser Powder Bed Fusion (LPBF) has received widespread attention because of high-degree freedom and integrated forming. However, it remains an unresolved issue to fabricate crack-free tungsten and tungsten alloys by LPBF, due to the high residual stresses, the weak connection between high-angle grain boundaries, and the poor plastic deformation capacity. In this study, W-Ta-C alloys were fabricated by incorporating TaC into pure W via LPBF in situ alloying, to investigate the effects of TaC on microstructure evolution, crack suppression and crack healing. The addition of TaC induced eutectic reaction, promoted grain refinement and decreased the Peierls-Nabarro stress for dislocation motion, thereby suppressing crack nucleation and propagation. Furthermore, a strategy to reduce the crack density through a self-healing mechanism was developed, which is achieved by dissolution and reprecipitation of carbides during hot isostatic pressing (HIP). After HIP treatment, most cracks were healed, resulting in an 88% reduction in crack density compared to as-built samples, ultimately achieving nearly crack-free LPBF-fabricated W alloy.

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Tentative

P1-036

Designing irradiation-tolerant TaTiVW refractory high-entropy alloys with defects simulation and heavy ion irradiation

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Body-centered cubic (BCC) single-phase solid-solution Refractory high-entropy alloys (RHEAs), comprising Ta, Ti, V, and W, are proposed as candidate structural materials for nuclear fusion, subject to specific downselecting constraints. The primary focus of this research lies in investigating the defect recombination capability under the hypothesis that variations in the potential energy landscape (PEL) determine the irradiation tolerance of RHEAs. Molecular Dynamics (MD) will optimize the composition of RHEAs to achieve the most PEL heterogeneity. Heavy self-ion irradiation will then be employed to compare various samples and validate the simulation findings. To ensure consistent defect mobility, the irradiation temperature will be maintained at one-third of each sample's homologous temperature (Tm). Additionally, In situ ion irradiation transient grating spectroscopy (I3TGS) and Transmission Electron Microscopy (TEM) will be employed to evaluate thermal elastic properties and microstructures to assess radiation-induced damage.

Tentative

P1-037

Hydrogen Permeation Behavior during the Formation of Tungsten Deposition Layers under Simultaneous Irradiation with Hydrogen and Helium

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Invited

Understanding the hydrogen isotope permeation behavior through the first wall is crucial for tritium safety management and fuel cycle control in DT fusion reactors. The first wall is exposed to irradiation with not only energetic hydrogen isotopes but also neutron and helium, both of which are reaction products of the DT fusion reaction. The candidate material for the first wall is tungsten (W), and its surface is sputtered by incident particles, leading to the formation of W redeposited layers on the first wall. Previous studies have shown that helium irradiation induces nanobubbles formation in surface layer of bulk W, altering hydrogen diffusion and retention properties. However, W deposition layers inherently contain a high density of defects, and the effect of additional helium irradiation on hydrogen permeability through the layer is not sufficiently understood.

In this study, W deposition layers were formed on substrates by sputtering a W plate using a mixed hydrogen-helium plasma while simultaneously measuring the hydrogen permeation flux. Subsequently, gas-driven permeation measurements and microstructural analyses of the W deposition layers were performed. The results revealed that hydrogen permeation flux increased with higher helium irradiation. The increase in the permeation coefficient was particularly significant when the helium mixing ratio was 30 % (based on the flow rate ratio to hydrogen), with the permeability increasing by approximately 18 times at 400 °C. This significant enhancement in hydrogen permeation is attributed to helium irradiation increasing the porosity of the W deposition layer, thereby providing additional diffusion pathways for hydrogen.

P1-038

Crack Suppression and Microstructural Evolution in Binary Tungsten Alloy fabricated by Laser Powder Bed Fusion

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Tungsten (W) and its alloy are crucial for plasma-facing components in fusion reactors due to their excellent properties, such as high melting point, thermal conductivity, and low sputtering yield. Additive manufacturing (AM), particularly laser powder bed fusion (LPBF), enables to produce complex W components. Nevertheless, cracking of W and its alloys during LPBF remains a significant challenge, primarily attributed to the inherent brittleness and high ductile-to-brittle transition temperature (DBTT), as well as the high thermal stress. While the crack suppression mechanism of LPBF pure W is well studied, research on W alloys is limited. In this study, we employed binary tungsten alloy to investigate the effects of LPBF parameters and printing strategies on crack and microstructure. We achieved nearly crack-free binary tungsten alloy with high-density cellular structure, exhibiting W-enrichment in the intracellular area and the other alloying element segregation along the cell walls. Additionally, secondary-phase particle and high-density dislocation were observed at the cell walls. The high-density cell structure could effectively mitigate stress concentration and impede crack propagation during LPBF, providing a viable pathway for crack suppression and enhanced mechanical properties in LPBF W alloys. Our findings provide insights for designing crack-resistant W-based materials, contributing to the development of plasma-facing components for fusion reactors.

Key Words: tungsten alloy, laser powder bed fusion, crack suppression, microstructure evolution

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Tentative

P1-039

Examination of a WTaCrV high entropy alloy under single and dual-beam ion irradiation

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High entropy alloys (HEA) offer potential improvements in radiation resistance over conventional materials. This comes as a result of several effects including lattice distortion and enhanced diffusion of vacancies making the annealing of defects easier. Tungsten-based HEAs are a potential alternative plasma facing component material. In this study thin films of WTaCrV HEA are synthesized by magnetron sputtering and characterized by SEM and XRD confirming a single phase composition. They are then irradiated using the DiFU facility at Ruđer Bošković Institute to a dose of 1 dpa both in single and dual-beam mode at room temperature and 520 °C. The ions chosen for irradiation were 200 keV He and 8 MeV Cu in order to ensure He is implanted in a depth of 500 nm with a dose of 200 appm, while minimizing Cu implantation effects. XRD observes both a shift in the lattice parameter and a small relaxation in the microstrain seen as a reduction in the width of the peak. However, no phase separation is observed by TEM analysis. The synthesized alloy therefore exhibits phase stability and radiation resistance under irradiation. The observed relaxation of crystal lattice will be discussed in relation to the sputtering parameters applied during deposition of thin film by magnetron sputtering.

Tentative

P1-040

Effect of irradiation fluence on deuterium retention for damaged W-Mo alloy

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Tungsten (W) is a promising candidate for plasma-facing materials in fusion reactors. Recently, several W alloys have been proposed, and their mechanical properties have been extensively evaluated. Notably, the solubility of hydrogen isotope in molybdenum (Mo) is lower than that in W, prompting an investigation into the hydrogen isotope retention behavior of W-Mo alloys under fusion-relevant conditions. Furthermore, it has been reported that the hydrogen isotope retention in W varies depending on irradiation-induced defects and irradiation fluence. Thus, it is important to examine the effect of irradiation fluence on hydrogen isotope retention in damaged W-Mo alloy.

In this study, W and W-5.2%Mo alloy disks (6 mm^{φ}, 0.5 mm^{t}) purchased from A.L.M.T. Co. Ltd were used. The samples were irradiated with 6 MeV Fe^{2+} at room temperature (R. T.) to damage level of 0.1 and 1 dpa in TIARA facility at QST. Positron annihilation spectroscopy (PAS) was performed to evaluate the size of irradiation-induced defects. Subsequently, the samples were exposed to deuterium (D) plasma at 300 K with ion fluence of 1.0×10^{25} m^{-2} and 7.0×10^{25} m^{-2}. After D plasma exposure, thermal desorption spectroscopy (TDS) measurements were applied to evaluate the D retention behavior.

The TDS results showed a significant reduction in D retention for undamaged W-5.2%Mo. However, additional D desorption peak was found for damaged W-5.2%Mo exposed to higher ion fluence of 7.0×10^{25} m^{-2}. In the presentation, the correlation between irradiation-induced defects and D retention behavior in W-Mo alloy will be discussed in detail.

Tentative

P1-041

Neutron irradiation effect on the material properties and the evaluation of bonding strength with W of Cu alloys

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Cu alloys are good candidates for fusion reactor divertor heat sink materials due to their high thermal conductivity. Among them, dispersion strengthened Cu alloys (DS-Cu) and precipitation hardened Cu alloys (PH-Cu) are commercially available candidates due to their high strength and thermal conductivity. However, most studies on the materials were carried out almost 20 years ago and the detailed studies on materials by using newer testing and analysis methods are few.

In this research, several Cu-Al_2O_3 specimens were irradiated in High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory. The irradiation temperature was about 80 °C and specimens were irradiated up to 5 dpa. Vickers hardness test, electrical resistivity test, tensile strength test, the scanning electron microscope (SEM) observation and electron dispersive X-ray spectroscopy (EDS) on the fracture surface were conducted to evaluate the material properties of irradiated Cu-Al_2O_3 alloys.

On the other hand, as the divertor heat sink material, the bonding strength between W is another key issue for Cu alloys as well. Nevertheless, no effective method which is applicable to the small bonding specimens were reported. In this research, the high temperature laser microscopy, digital image correlation (DIC) analysis, and ASTM D4541 pull-off strength testing method were applied to observe the in-situ thermal fatigue and to measure the bonding strength of Cu alloys and W fabricated by several methods. The effect of thermal fatigue on the bonding strength as well as the applicability of the ASTM method on the measurement were discussed in the session.

Tentative

P1-042

Evaluation of deuterium retention behavior in potassium-doped tungsten with irradiation defects under high-temperature conditions

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Tungsten (W) is a leading candidate for plasma-facing materials (PFMs) in deuterium-tritium(D-T) fusion reactors. However, neutron-induced irradiation damage significantly degrades the material properties of W. To address this issue, Potassium(K)-doped W has been proposed, as K bubbles mitigate irradiation damage. Furthermore, several studies have reported that the K doping suppresses the embrittlement of W due to high thermal loads. Despite these advancements, the behavior of hydrogen retention in K-doped W under high-temperature irradiation conditions remains poorly understood. This study investigates the role of K doping in hydrogen isotope retention at high temperatures.

In this study, both pure W and K-doped W samples, fabricated by A.L.M.T Co., Ltd., were utilized. These samples were irradiated by 6 MeV Fe^{2+} at R.T. and 500 °C (1 dpa). PAS measurement was performed to evaluate the size and density of irradiation-induced defects. Subsequently, the samples were exposed to 3 keV D_2^+ irradiation with the ion flux of 1.0×10^{18} m^{-2} D^+ m^{-2} s^{-1}, up to the fluence of 1.0×10^{22} m^{-2} D^+ m^{-2} at R.T. After that, TDS measurement was performed from R.T. to 1173 K.

For R.T. irradiated samples, TDS results showed that K-doped W had less D retention compared to pure W. PAS results indicated that K-doped W had a lower long positron lifetime intensity than W, suggesting fewer large defects, such as voids in the K-doped W. These findings imply that the decreased D retention in K-doped W may be attributed to the reduced presence of large-scale defects.

Tentative

P1-044

Unveiling the Impact of Dopants on the Metallic Cu-W Interface for Fusion Reactor Applications

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Cu-W metallic multilayers are promising candidates for the first wall of future fusion nuclear reactors due to their high melting point, excellent thermal conductivity, and defect-annihilating properties. However, the accumulation of impurities generated during neutron irradiation can weaken the interface, leading to operational failure over time. Enhancing the interfacial properties of these multilayers is therefore critical. In this study, we employ density functional theory (DFT) to explore the effects of doping with potential elements on the interfacial properties of the Cu-W system. Our findings aim to provide insights into improving the durability and performance of Cu-W multilayers in fusion reactor environments.

Tentative

P1-045

Overaging of CuCrZr: Degradation and Limits to the Restoration of the Initial Properties

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The precipitation hardened CuCrZr alloy is the baseline heatsink material in EU-DEMO. In the ideal solution annealed, cold worked and subsequently aged condition (SAcwA), CuCrZr exhibits a balanced combination of high strength, toughness, and thermal conductivity. During the manufacturing of a divertor target, however, CuCrZr is heated to relevant temperatures to ensure reliable joining of heatsink and W armor. Overaging of the nanoscale Cr-rich precipitates is the consequence.

Here we assess the degradation of CuCrZr after simulated joining cycles and determine the limits to the restoration of the initial properties. Aging of SAcwA was carried out to represent the most relevant joining technologies: 980 °C /1 h /slow cooling, as widely used brazing parameters and 550°C /4h to simulate diffusion bonding. Additionally, sections of both overaged materials were subjected to a solution annealing, water quenching, and aging (SAA) treatment as an approach to restore the initial properties best possible.

Aging 550 °C /4h moderately reduces the strength of the initial SAcwA state to 70%, whereas tensile behavior of CuCrZr after 980 °C /1h is comparable with stress relieved pure Cu. The SAA treatment does not improve the yield strength of the 550 °C aged material, but the hardening behavior and thus ultimate tensile strength. The 980 °C aged material benefits significantly from the SAA treatment: The tensile behavior can be restored to a level that corresponds to the 550 °C aged material. However, the necessity of a SAA treatment after joining and the detection of abnormal grain growth impairs the 980 °C in favor of the 550 °C route.

Tentative

P1-046

Exploring the inhibitory effect of WTaVCr high-entropy alloys on hydrogen retention: From dissolution, diffusion to desorption

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Due to the excellent resistance to radiation damage, high-entropy alloys (HEAs) have been considered as important candidates for plasma-facing materials. However, to realize the great potential of HEAs in fusion energy, it is important to consider the hydrogen (H) behaviors, which remain to be elucidated. Here, we systematically investigate the dissolution, diffusion and desorption of H in equiatomic WTaVCr using the first-principles calculations. It is found that the H solution energy in WTaVCr is lower than that in pure W, which can be clarified by the H affinity of metallic elements and the available volume of interstitial sites. Accordingly, a possible diffusion path of interstitial H is selected, containing 26 metastable sites, and the corresponding energy barrier is 0.46 eV. Besides, the maximum trapping energy and accommodation number of H at a mono-vacancy in WTaVCr is only 0.30 eV and 3 ~ 5 H, respectively, which is much lower than that in pure W (1.28 eV and 12 H). More importantly, because of the low trapping energy and moderate diffusion energy barrier, the desorption temperature of H from a mono-vacancy is lower than the room temperature, implying that vacancies are not efficient trapping centers for interstitial H atoms in WTaVCr. Therefore, although the equilibrium interstitial H concentration in WTaVCr is higher than that in pure W, H retention at high temperatures and H-induced blistering are suppressed due to the weak H-defect interactions. Our results will provide a good reference for understanding the H behaviors in HEAs.

Tentative

P1-047

Development of an interatomic potential for the study of tungsten-based SMART alloys under the atomic cluster expansion formalism

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To address the growing demand for low-carbon energy, the United Kingdom has committed to building the world's first nuclear power plant by 2040, the Spherical Tokamak for Energy Production (STEP). One of the challenges faced by the STEP project concerns the oxidation of the reactor's tungsten-based first wall, which may occur during a loss of coolant accident or during remote maintenance handling. To mitigate the oxidation of tungsten, there has been a growing interest in the use of Selfpassivating Metallic Alloys with Reduced Thermo-oxidation ('SMART' alloys) in future tokamak fusion reactor designs. In particular, the ternary alloy W-11.4Cr-0.6Y has been identified as a potential candidate for fusion reactor first walls owing to its favourable oxidation resistance and self-passivating behaviour. However, the role of the alloy elements, Cr and Y, is not fully understood at the microscopic level. To address this, a new machine-learning interatomic potential (MLIP) under the atomic cluster expansion formalism has been developed to enable the study of the oxidation of tungsten-based SMART alloys at the molecular dynamics level of theory. We present our current progress on the development of this MLIP and demonstrate its applicability to the study of tungsten-based SMART alloys, comparing its performance with previous interatomic potentials and density functional theory calculations

Tentative

P1-048

Effects of neutron irradiation on hardness and electrical resistivity of tungsten and tungsten alloys after neutron irradiation in BR2

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Recently, tungsten (W) is considered to be candidate materials for plasma face components (PFCs), but rolled W material has issues such as recrystallization suppression and the low-temperature embrittlement behavior. So, dispersion strengthening (K bubbles or La_2O_3) or alloy addition (Re, Ta and etc.) are being attempted to develop W materials for PFCs. There are five types of alloys used in this study (pure tungsten, K-doped W, W-1%La_2O_3 and W-1, 3%Ta). Powder metallurgy and hot rolling followed by stress-relief heat treatment were conducted and supplied by A.L.M.T. Corp. Neutron irradiation was conducted by LIBERTY-13 irradiation campaign in BR-2 reactor (3 mm diameter, 0.25 mm thickness disk shape samples). The irradiation temperature was 290 °C (± 15 °C) and irradiation fluence was 5-8 E+23[neutron/m_2](En > 1 MeV). After the irradiation, Vickers hardness test was performed using indentation load of 1.96N and electric resistivity measurement was conducted by the 4-probe method at room temperature. Annealing was performed from 300℃ to 1000 °C in a vacuum after neutron irradiation. The aim of this study was to investigate the behavior of neutron irradiation hardening and electrical resistivity in the W materials of the dispersion strengthening alloys and Ta-added alloys. As a result, electric resistivity of most samples increased after neutron irradiation, except the W-3%Ta condition. The resistivity of Pure W was lowest and that of W-3%Ta was highest. Hardness of all of samples increased after irradiation, especially that of W-3%Ta was highest than other samples. And then, we investigated the effects of annealing temperature on irradiation hardening.

Tentative

P1-049

Vacancy Cluster Formation and Hydrogen Isotope Exposure Effects on Neutron Irradiated Tungsten

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Tungsten (W) is recognized as one of the leading candidates for plasma facing materials. Previous studies have shown that the retention of hydrogen isotopes in W increases significantly after neutron irradiation due to the trapping effect of defects (vacancies). However, the effect of temperature differences during hydrogen isotope plasma exposure on the change in vacancies has not yet been fully elucidated. In this study, in order to experimentally clarify the effect of the temperature difference during hydrogen isotope plasma exposure on the change in vacancies, a W sample that had been neutron irradiated was exposed to deuterium plasma.

The W samples were neutron irradiated in BR2 and then exposed to deuterium plasma in the Compact Divertor Plasma Simulator at the International Research Center for Nuclear Materials Science, Institute for Materials Research, Tohoku University. The sample temperature during plasma exposure was 563 K and 773 K. The size of vacancies before and after plasma irradiation was measured using the positron lifetime measurement.

In the sample that was plasma irradiated at 563 K, there was almost no change in the positron lifetime before and after plasma irradiation, but in the sample that was plasma irradiated at 773 K, the positron lifetime increased significantly. On the other hand, in the sample annealed at 773 K, there was almost no change in the positron lifetime. From the above results, it was found that the size of the vacancy cluster changes depending on the presence or absence of hydrogen isotopes and the plasma exposed temperature.

Tentative

P1-050

Effects of ageing on the service thermal and mechanical properties of additive manufactured CuCrZr alloys

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Tentative

The optimization of cooling channel geometries for advanced heat sink designs has led to considering metallic additive manufacturing as one of the primary manufacturing processes, especially for components with highly complex cooling internal structures such as the divertor. Among the different AM technologies, Laser Powder Bed Fusion (LPBF) is one of the most cost-effective methods, as it allows a high degree of freedom in the selection of the different process parameters.

Among the processable alloys by LPBF, CuCrZr is currently one of the leading candidates for the manufacturing of the cooling channels of ITER and DEMO divertors due to its combined high thermal conductivity and mechanical properties. However, as a precipitation-hardened alloy, its performance strongly depends on chemical composition and thermal history, leading to a wide variety of suitable post-built heat treatments.

Although thermophysical properties are well characterized for ITER-grade CuCrZr manufactured by powder metallurgy, data for AM-CuCrZr alloys are sparse and not often collected for analysis given the wide variety of manufacturing and thermal treatments.

In the present work, the effect of three heat treatments, based on solution annealing and ageing (SA+AH) and direct ageing at 580 °C for 10 and 60 minutes (DAH10 and DAH60, respectively), on microstructure, tensile properties, and thermal conductivity was investigated as a function of temperature from RT to 450 °C under high vacuum atmosphere. Results show that high-density CuCrZr alloys can be produced by the LPBF route. Furthermore, by direct ageing treatment, excellent mechanical properties can be achieved while maintaining good thermal conductivity.

P1-051

Performance study of tungsten fiber-reinforced tungsten composites for nuclear fusion reactor

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Tungsten is a leading plasma-facing material for fusion reactors due to its high melting point, but its brittleness at high temperatures remains a concern. Incorporating tungsten fibers enhances its toughness and strength while preserving high-temperature performance. This study produced fully dense tungsten fiber-reinforced tungsten (Wf/W) composites in large sizes (>60 mm in diameter) using an innovative multi-step high-temperature, high-pressure sintering process. These composites demonstrated superior thermal shock fatigue resistance (>0.55 GW/m^2) under transient heat conditions simulating edge-localized modes (ELMs), outperforming most tungsten grades.

Mechanical properties of the composites were evaluated through high-temperature tensile tests (up to 1000 °C) and three-point bending tests, revealing crack propagation along the fiber-matrix interface, enabling pseudo-ductility. Heavy ion bombardment (7.5 MeV W^{2+}) simulated neutron irradiation damage (up to 5 dpa), followed by deuterium plasma irradiation (flux: 10^{22} D m^{-2}s^{-1} @ 500 K, fluence: 2×10^{26} D m^{-2}). The sputtering resistance of the composites under displacement damage conditions and the retention of deuterium in the composites were analyzed using thermal desorption spectroscopy. Post-irradiation, crack thresholds were tested under high thermal loads to assess thermal shock resistance. Results showed the Wf/W composites had twice the fracture toughness of ITER-grade tungsten and five times less deuterium retention than CVD Wf/W composites due to their high density. Their exceptional performance under transient thermal loads is attributed to the pseudo-ductility mechanism. This study provides the latest research progress on Wf/Wm composites developed in China.

Invited

P1-052

Interaction between the dislocations and cavities in tungsten with in-situ TEM observation during tensile test at 973 K

Kouichi Tougou[1], Kaito Mizuno[2], Ken-ichi Fukumoto[3], Ryoya Ishigami[4]

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The irradiation-induced defects including cavities play a significant role in the resistance against the dislocation motion and correspond to the origin of irradiation hardening for the diverter materials in fusion reactor. The purpose of this study is to evaluate the irradiation hardening strength and the transition of interaction mechanism between dislocation and cavity in tungsten (W). It was investigated by in-situ transmission electron microscopy (TEM) observation during tensile test at 973 K. W samples were annealed at 1373 K for 10 hours in a vacuum for recrystallization. The 2 MeV helium ion irradiation was performed to generate the cavities in W samples with the WERC tandem accelerator. The damage level of the ion irradiation was 0.2 dpa at 973 K. The cavity diameter and number density were approximately 2 nm and 5 × 10^22 m^-3, respectively.

The obstacle barrier strength α was calculated by the equation of “cos(Φ/2)” using the critical bow-out angle Φ of dislocation while the mobile dislocation was pinned at the cavity. The average of α and the fraction of cross-slip and double cross-slip were determined about 0.4, 60 %, respectively. The fraction of cross-slip and double cross-slip of 2 nm size cavity at 973 K was larger than those at 293K and 723 K.

Tentative

P1-054

High heat flux test components for repaired plasma-facing materials

Daniel Dorow-Gerspach[1], Gerald Pintsuk[1], Jannik Tweer[1], Marius Wirtz[1], Christian Linsmeier[1]

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Extreme stationary and transient heat loads can be anticipated in the divertor region of a future reactor. Combined with heavy particle loads, the plasma facing materials (PFM) are heavily damaged and eroded during operation, limiting the lifetime of these components. Currently, the replacement of damaged components is foreseen, resulting in long downtimes. Using local deposition techniques or additive manufacturing like wire-based laser metal-deposition (LMD-w), in-vessel repair procedures are being developed [1].

The current state-of-the-art component design for the divertor is the monoblock (MB) chain using W-blocks and a CuCrZr pipe as heat sink. However, they are rather limited and expensive, making the development including parameter studies of such repair procedures very difficult and time consuming.

Therefore, we developed a simple flat tile type component design and test procedure to mimic the situation of an ITER-like MB-chain including surface sizes, armor thickness and gap between individual blocks. This development was backed up by FEM studies allowing successful realizations of the desired temperature loads.

Different repair procedures were applied on W-tiles and soldered on Cu-cooling structures and these components were than HHF loaded for 100 and 1000 cycles in such a way to mimic the surface temperatures and temperature profiles present in standard MB-chains at 10 MW/m^2 and 15 MW/m^2 stationary load. In addition, our standard transient loading procedures [2] were to the larger MB like tiles. Therefore, it is possible to investigate the influence of repair parameters, surface conditions or layer thicknesses under as realistic as possible heat load conditions.

Tentative

P1-055

High pulse number tests of pre-damaged plasma facing surfaces

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Very high power loads such as plasma disruptions and vertical displacement events result in the formation of recrystallized and molten layers on the plasma facing material (PFM) surface. These will have a strong influence on the material properties as well as the fatigue lifetime of the material.

Tungsten meeting the ITER material specifications was exposed to thermal transients with power densities above the melting threshold in the linear plasma device PSI-2 to generate surface layers of molten material with a depth of ~200 µm. Subsequently, these pre-damaged samples were simultaneously exposed to high pulse numbers of ELM-like thermal shock events using a high energy laser and to a mixed H/He background plasma.

All induced surface modifications such as melting, crack formation and roughening due to plastic deformation were analyzed by scanning electron microscopy, light microscopy, laser profilometry and metallographic means. The characterization of the damage behavior and the comparison of the already available threshold values of as-received and recrystallized material with the new data on re-solidified material will add information to the database for the impact of melt layer formation on damage formation and erosion of the PFM and subsequently the lifetime of PFMs after such intensive heat load events.

The results show that the damage evolution of the molten and solidified surface is much faster than on as-received samples due to the related softening. This could mean/could lead to a shorter lifetime and severe surface damage during operation of a fusion reactor.

Tentative

P1-056

Optimization of additive manufacturing process parameters in ITER-grade CuCrZr using rapid screening methods

Edgar Leon-Gutierrez[1], Francisco Canillas[1], Antonio Fernandez-Vina[2], Nerea Ordas[3], Carlos Bloem[4], Rebeca Hernandez[2]

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The feasibility of obtaining dense parts with good mechanical properties of the sink material for ITER and DEMO ITER-grade CuCrZr has been demonstrated through powder bed fusion electron beam (PBF-EBM). In certain instances, these properties have been further enhanced, highlighting the significant variability and dependency on manufacturing parameters and subsequent heat treatments. This variability underscores the necessity for systematic testing biacross a large matrix of additive manufacturing (AM) process parameters, to optimize material performance.

Tensile tests are commonly used during the qualification of AM processes, as well as to evaluate their stability, reproducibility and repeatability. However, the substantial material volume required, and the high machining costs make it impractical to systematically explore vast parameter matrices. Additionally, AM processes can lead to local microstructural variations due to the unique thermal histories occurring at different locations within a single component. And there is not always enough material available during the development phases.

In this study, small punch tests are employed as screening method to optimize the AM process. Multiple batches of CuCrZr were fabricated using different PBF-EBM parameters and heat treatment conditions. This screening approach enabled the identification of optimal conditions to produce parts with superior mechanical properties. Furthermore, the materials demonstrating the best mechanical performance were used to correlate small punch test results with standardized tensile tests. This correlation makes it possible to establish a direct relationship between the results of small punch tests and engineering mechanical properties and provides valuable data to design components of fusion reactors.

Invitation

P1-057

Overview of deuterium retention measurements in tungsten with impurity seeding in the linear plasma device PSI-2

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Hydrogen fuel retention is a crucial issue for the availability of a fusion reactor. We performed systematic studies of the influence of individual impurities helium, nitrogen, argon and neon as well as a mixture of them on the deuterium retention in tungsten for a range of surface temperatures Ts of 500-800 K and seeded impurity fractions of 3-10%. The incident ion flux and fluence were ~10^22$m^{-2}s^{-1}$ and ~10^26m^{-2}, respectively. We confirmed the prominent effect of helium reducing the D retention by a factor of 3-100, with a stronger reduction at higher Ts. The effects of argon and neon were similar when the incident ion energy was well above the sputtering threshold. At Ts=500 K, both impurities increased the D retention by a factor of 2-4, while at 770 K their effect was the opposite. Nitrogen had a strong impact increasing the retention by a factor of ~10 and ~100 for 500 and 770 K, respectively. The positive effect of He vanished when N was added (D+He+N plasma), with the D retention remaining at the level of the pure D exposure. Sputtering is presumably also the reason for a significant influence of argon on the effect of nitrogen. At Ts=800 K, while nitrogen increased the D retention by a factor of ~100, for D+N+Ar the rise of retention was 2-3 times less strong.

The contribution summarises implications for the impurity seeding in a reactor. E.g. with nitrogen the enhanced tritium implantation in tungsten could set a safety limit for ITER to ~1000 discharges.

Tentative

P1-058

Direct comparison of gas- and ion-driven deuterium permeation through fusion-relevant materials

Armin Manhard[1], Philipp Sand[1], Udo von Toussaint[1]

[1]Max Planck Institute for Plasma Physics

To estimate fuel losses into and through armour and structural materials for plasma-facing and blanket components of fusion reactors, it is essential to know their transport parameters for hydrogen isotopes, i.e., permeability and diffusivity. In the reactor, the materials will be exposed to neutral gas and, for plasma-facing components, also ion bombardment. At elevated temperatures, both can contribute to the total permeation flux.

In our permeation set-up TAPAS [1], we are able to charge samples with deuterium ion beams under comparable conditions (flux, energy and material temperature) as they are expected to occur at the first wall of a fusion reactor. Additionally, we can also apply gas pressures up to several Pa, which is also in the relevant parameter range. This allows to directly compare the two different charging mechanisms in-situ without switching the set-up.

We present the results of gas- and ion-driven permeation experiments for tungsten, Eurofer97 RAFM steel and the tungsten heavy alloy HPM1850 samples, measuring the permeating deuterium by a highly sensitive quadrupole mass spectrometer. We explain how gas- and ion driven permeation measurements complement each other, and discuss the importance of the different hydrogen isotope charging mechanisms under fusion reactor first wall conditions. In addition, we show the consideration of the effect of oxygen impurities may be in some cases essential for the evaluation of the permeation behaviour of the different materials.

[1] P. Sand et al, Rev. Sci. Instrum. 95, 123302 (2024)

Tentative

P1-059

Evaluation of residual thermal stress and temperature distributions in W-HEA-CuCrZr multilayer systems for fusion nuclear divertors.

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[1]Universidad Carlos III de Madrid / Instituto Tecnologico de Quimica y Materiales Alvaro Alonso Barba (IAAB), [2]Universidad Carlos III de Madrid

Current DEMO divertor designs use plasma faced W monoblocks coupled to the CuCrZr based components of the water-based heat extraction system. However, the optimum operating temperature for W is 800-1200 °C, while it is below 400 °C for CuCrZr. Another challenge is the bonding of these materials, as their expansion thermal coefficients, CTE, present a large mismatch, with CTE(Cu) ≈ 4CTE(W), resulting in thermal residual stresses at CuCrZr interface exceeding its yield strength, leading to potential interface failure. A proposed solution is to incorporate an interlayer acting as a thermomechanical coupling layer with reduced thermal conductivity and intermediate CTE value, which keeps the temperatures of the materials within their respective operating ranges and reduces the induced strains at the bond of the materials.

This study evaluates Cu5Cr35Fe35V20 (Mo, Ta, Ti or W)5 high entropy alloys (HEAs) as potential thermomechanical coupling interlayers. The thermal residual stress distributions across the four W/HEA/CuCrZr triple-layer systems have been calculated as a function of interlayer thickness using an analytical model and thermo-mechanical analysis based on the finite element method for divertor designs in accordance with those proposed for DEMO. For these analyses, the experimental values of thermal conductivity, Young's modulus and CTE have been used. The results show that a HEA interlayer of adequate thickness ensures that the residual stresses at the interfaces of the components, particularly in CuCrZr, are below their yield strength, which prevents plastic deformation at the interfaces during operation. Temperature distributions across the systems were also determined, confirming the thermal barrier effectiveness of the interlayer.

Invitation

P1-061

Non-isothermal high-temperature corrosion behavior of $CrBe_{12}$ in vapor-gas mediums with different isotopic compositions

Kuanysh Samarkahnov[1], Yerzhan Sapatayev[1], Timur Kulsartov[2], Vadim Bochkov[1], Inesh Kenzhina[3], Zhanna Zaurbekova[2], Alexander Yelishenkov[3], Sergey Udartsev[4], Mikhail Podoinikov[4]

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Template

Intermetallic beryllium compounds, such as $MeBe_{12}$ (Me = Ti, Cr, V), are promising materials for solid breeder blanket systems in future fusion reactors like ITER and DEMO due to their unique physico-mechanical properties. However, limited data on their behavior in vapor-gas environments with varying isotopic compositions hinder a full assessment of their operational characteristics and application in fusion technology. Studying the nature and mechanisms of corrosion processes in these materials is crucial for evaluating their suitability for use in nuclear fusion energy. In-depth research simulating real operational conditions is crucial for clarifying the interaction mechanisms between beryllides and chemically active gases, and for developing strategies to enhance their durability and resource efficiency.

This study focused on $CrBe_{12}$ beryllide samples in plate form, synthesized by "UMP" JSC (Ust-Kamenogorsk, Kazakhstan). High-temperature corrosion experiments with $CrBe_{12}$ samples were conducted using the TiGrA experimental complex, which enables thermogravimetric analysis (TGA), differential scanning calorimetry (DSC), and mass spectrometry measurements within a single experiment. The experiments were performed under non-isothermal heating conditions up to 1200 °C in a vapor-gas medium with varying isotopic compositions.

The experimental results revealed the main patterns of the corrosion processes in chromium beryllides. Temporal dependencies of mass changes in the samples and gas phase composition during heating were determined, along with temperature-dependent corrosion reaction rates and corrosion layer thickness. Comprehensive materials science analyses were performed to assess the impact of high-temperature corrosion on the structure and phase composition of the beryllides.

P1-062

The evaluation of deuterium retention properties of $Be_{12}Ti$ after irradiation with D ions at high temperature

Yutaka Sugimoto[1], Jae-Hwan Kim[1], Taehyun Hwang[1], Mitsutaka Miyamoto[2], Hiroyasu Tanigawa[1]

[1]Fusion Energy Research and Development Directorate, National Institutes for Quantum Science and Technology, [2]Department of Material Science, Shimane University

In a blanket system of DEMO (Demonstration) fusion reactors, beryllides (beryllium intermetallic compounds) are planned to be used as neutron multipliers. The proposed blanket system will adopt beryllide blocks instead of beryllide pebbles from viewpoints of higher thermal conductivity, lower tritium retention, and simplification for structure. To effectively generate tritium (T) as fuel, the neutron reaction between lithium and neutron (n), and $^{9}Be(n,2n)^{4}He$ of nuclear reaction is used in beryllide. Since other nuclear reactions between Be and neutrons result in tritium production. The tritium-retaining beryllides can affect the safety and operation of a fusion reactor. Therefore, it is essential to investigate the tritium retention properties due to the reduction in tritium retention and the expectation of tritium desorption behavior.

In this study, the deuterium retention properties of the beryllide ($Be_{12}Ti$) were evaluated using an deuterium ion gun and thermal desorption spectroscopy by comparison to the results in beryllium. Mechanically polished and electropolished $Be_{12}Ti$ samples were used. Under irradiation conditions, deuterium(D) ions were implanted with 3 keV, fluence was $1x10^{21}$ D/m^2, and irradiation temperatures were from R.T. to 873 K. The bubble behaviors in $Be_{12}Ti$ and Be were observed using a transmission electron microscope with an ion gun under irradiation and annealing. In addition, thermal desorption spectroscopy evaluated apparent desorption activation energies from the electropolished and mechanically polished $Be_{12}Ti$ samples after the irradiation at high temperatures.

Tentative

P1-063

Analysis of hydrogen isotope release from lithium ceramics with different pore sizes and distributions

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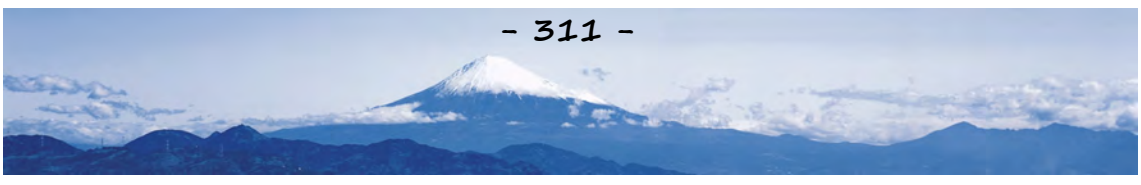
The effect of the pores on the tritium release from lithium ceramics requires elucidation. The pore size and distribution may affect the tritium release peculiarity, e.g., desorption temperature, desorption rate, and tritium retention in ceramics. In this study, the tritium release model, named as TRLC (Tritium Release from Lithium Ceramics), was improved based on the kinetics of tritium diffusion, trapping and de-trapping, and annihilation of defects. The tritium transportation in closed and open pores was incorporated into the model, which was analyzed based on Fick's second law and gas diffusion theory in porous media. It was found that both types of pores resulted in an additional desorption stage at higher temperature range near 820 K. The increase in the density and size of closed--pores led to the shift of desorption stage to a further higher temperature. The phenomenon was confirmed by the experimental data from the Li_2TiO_3 irradiated by neutrons. Further, the D_2 gas exposure and subsequent deuterium thermal desorption measurement for the ceramics with different pore sizes and distributions were conducted to modify the model. For the application of the lithium ceramics in the blanket, the calculation results show that the tritium retention in ceramic with large closed-pores was increased after the shut-down of the reactor. (208 words)

Tentative



P1-064

The impact of lithium deficiency and presence of helium on the performance of lithium metatitanate (Li2TiO3).

Charles Victor Thorogood[1], Derek Sinclair[2], Julian Dean[2], Eduardo Garciadiego-Ortega[3], Slava Kuksenko[3], Amy Gandy[2]

[1]University of Sheffield / UKAEA, [2]University of Sheffield, [3]UKAEA

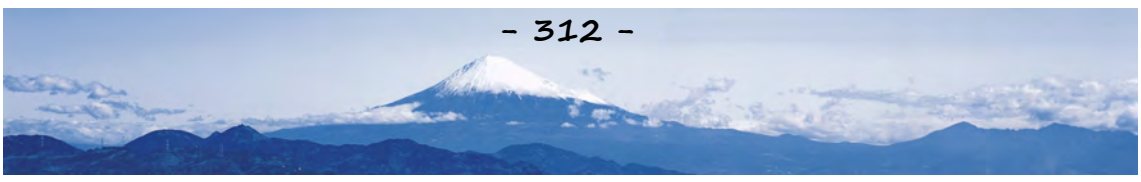
Tritium is required as a fuel for most current fusion reactor designs. It can be created by the neutron transmutation of lithium into tritium which also produces helium as a by-product. Many different lithium containing materials can be used for this process but lithium metatitanate (Li_2TiO_3) is a promising candidate due to its chemical stability and good mechanical properties. Understanding how the material will be affected by fusion like conditions will be key to making an informed assessment.

This project investigates different synthesis routes to maximize lithium density whilst retaining microstructure control and remaining industrially scalable. This is being investigated using a variety of different compositions, sintering conditions and sintering aids.

This project is also investigating the effects of exposure to extreme neutron fluxes. Specifically, these neutron fluxes will cause the stoichiometry to change due to lithium transmutations occurring. Understanding how lithium deficiency affects the materials crystal structure, microstructure and the ability of the tritium to escape is vital to assess the viability of this material.

Neutrons also cause damage cascades and the creation of interstitial helium. These will also likely have a profound effect on performance and have been investigated using ion implantation. The effects of these have been investigated using a variety of characterization techniques such as GIXRD, SEM and impedance spectroscopy such that the effects they have had on the material can be assessed.

Tentative



P1-065

CTE Measurement of Pebble Beds with Radial Expansion Correction of the Sample Holder

Harsh Patel[1], Maulik Panchal[1], Aroh Shrivastava[2], Paritosh Chaudhuri[2]

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Lithium ceramic pebbles, such as Li_2TiO_3, are considered promising tritium breeder materials for use in the blankets of fusion reactors. Under high-temperature operating conditions, these ceramic pebbles undergo volumetric expansion, which can induce mechanical stress and affect the integrity of surrounding structural components. Therefore, accurate measurement of the coefficient of thermal expansion (CTE) of pebble beds is essential for the reliable and safe design of fusion blanket systems.

To address this, an experimental setup was developed to measure the CTE of pebble beds within the typical operating temperature range of fusion blankets. Pebbles are filled into a cylindrical container and secured by a cylindrical block placed on top. Two precision quartz rods are connected to the container and the top block, respectively, and are used to transmit displacement to a Linear Variable Differential Transformer (LVDT), which captures the axial expansion of the bed. A three-zone split tube radiation furnace surrounds the assembly to ensure uniform heating.

To minimize measurement errors arising from the thermal expansion of surrounding components, two correction curves were applied. The first was obtained experimentally using a solid cylindrical quartz block to quantify axial expansion error, and the second was derived analytically and through finite element method (FEM) simulations to account for the container's radial expansion.

This setup enables accurate determination of the thermal expansion behaviour of Li_2TiO_3 pebble beds. The poster will present the experimental setup in detail, including its design, calibration methodology, and resulting CTE measurements.

Tentative

P1-066

Thermal Cycling Test of EU Reference Tritium Breeder Pebbles

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Advanced Ceramic Breeder (ACB) pebbles serve as the EU reference material for the ITER TBM and solid breeder blanket concepts in DEMO. Therefore, a comprehensive knowledge of the material properties and behaviour is essential. Although the ACB material has been extensively characterised in the past, also with regard to their long-term thermal stability, this work encompasses results from the first thermal cycling test. Thermal cycling was used to imitate high thermal stresses in the material.

Within the thermal cycling test, two batches of ACB pebbles consisting of nominal 70 mol% Li_4SiO_4 and 30 mol% Li_2TiO_3 were annealed in cycles comprising 12 h at 300 °C and 12 h at 800 °C. After 2, 14 and 30 cycles, samples were retrieved for their examination. A gas atmosphere of He + 0.1 vol% H_2 was applied during the experiment. Before and after the heat treatment, the ACB pebbles underwent an extensive characterisation. Information on the chemical and phase stability, the microstructure, porosity, and mechanical strength were obtained.

The first thermal cycling test on ACB pebbles and its results will be presented. In addition, a comparison to previous long-term annealing experiments will be drawn. This study will contribute to the knowledge on thermal stability of ACB pebbles for their use in future fusion power plants.

Tentative

P1-067

Study on the kilogram-scale preparation lithium lead titanate ceramic pebbles as a new neutron-tritium complex breeder for fusion reactor blanket

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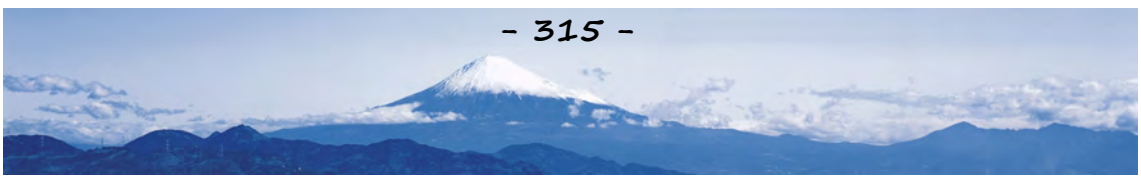
[1]Anhui University

Lithium metatitanate (Li_2TiO_3) is regarded as one of the most promising tritium breeder materials for fusion reactor blanket because of its outstanding comprehensive properties. Based on the similarity of Pb and Ti in chemical bonding, a neutron-tritium complex breeder exhibiting physical-mechanical properties akin to Li_2TiO_3 can be synthesized by substituting a portion of Ti with Pb. In this study, lead dioxide, titanium dioxide and lithium hydroxide were used as raw materials to prepare lithium lead titanate ($Li_2Pb_xTi_{1-x}O_3$) by solid phase method. Building upon the conventional indirect wet process, the polymer-assisted sedimentation technique was employed to generate spherical droplets by gradually introducing ceramic slurry into a viscous heated medium. Throughout the sedimentation process, internal polymerization within the droplets ultimately resulted in the formation of precursor pebbles. The impact of lead content and sintering parameters on the physical characteristics of the pebbles was examined, facilitating the kilogram-scale fabrication of lithium lead titanate ceramic pebbles. The results indicate that the optimized synthesis protocol is as follows: a Pb:Ti (molar ratio) of 2:3, a polymerization monomer addition of 7.1 wt%, sintering at 850 °C for 6 hours, yielding ceramic pebbles with diameters of 1.0-1.2 mm, a crushing load of 32 N, a sphericity of 1.02, and a relative density of 87.2%.

Tentative



P1-068

TGA study of high-temperature corrosion behavior of neutron-irradiated $Be_{12}Ti$ in the presence of water vapor

Saulet Askerbekov[1], Asset Shaimerdenov[2], Zhanar Bugybay[2], Assyl Akhanov[2], Magzhan Aitkulov[2], Yevgen Chikhray[3], Timur Kulsartov[3], Inesh Kenzhina[1], Zhanna Zaurbekova[3], Manarbek Kylyshkanov[4]

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Tentative

This paper studies the high-temperature oxidation of titanium beryllide ($Be_{12}Ti$) in an inert gas atmosphere with an admixture of water vapor. The objects of the study are both unirradiated and irradiated to different doses samples: 1.9 dpa and 2.9 dpa. Irradiation was carried out at low temperature in the WWR-K reactor (Almaty) with the following neutron flux parameters:

Thermal neutrons: 5.7×10^{13} n/cm^2 s;

Fast neutrons: 2.4×10^{13} n/cm^2 s.

Hot isostatic pressing (HIP) manufactured the studied material at Ulba Metallurgical Plant JSC (Ust-Kamenogorsk, Kazakhstan). Corrosion experiments were carried out using a TGA-DSC 3+ synchronous thermal analyzer. Separate analysis of heat flux (Differential Scanning Calorimetry) and mass change (Thermogravimetry) under the influence of heating and water vapor atmosphere allowed to quantify the thermal effects of chemical reactions and establish their relationship with the change in sample mass.

The applied thermal analysis methods allowed us to estimate the effect of neutron irradiation doses on the corrosion resistance of $Be_{12}Ti$ and determine the kinetic parameters of its interaction with water vapor. We also established temperature dependencies of the rate constants of chemical reactions for samples subjected to different irradiation levels.

The obtained experimental results emphasize the effect of neutron irradiation on the corrosion behavior of $Be_{12}Ti$, which is of decisive importance for predicting its reliability and durability in the breeder blanket of a thermonuclear reactor.

The research has been funded by the Science Committee of the Ministry of Science and Higher Education of the Republic of Kazakhstan with Program BR24993119.

P1-069

Liquid Breeder Materials qualification in IFMIF-DONES Facility

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The IFMIF-DONES facility aims to test and validate qualify materials for nuclear fusion development, serving as a cornerstone for advancing fusion technology.

With anticipated delays in ITER's deuterium-tritium phase, initial results from the Test Breeder Module (TBM) are expected to be postponed. This highlights the urgent need for experiments replicating fusion irradiation conditions to validate breeder blanket technologies. In response, IFMIF-DONES has revitalized its experimental program. Building on the achievements of earlier phases, such as IFMIF-EVEDA, the project has progressed the conceptual and preliminary designs of additional modules to qualify breeder blanket technology, including among them the Liquid Breeder Validation Module (LBVM).

The LBVM is specifically designed to test materials for future liquid breeder blankets under irradiation conditions comparable to those in DEMO. Neutron transport calculations have been conducted to evaluate radiation damage rates on coatings materials for tritium anti-permeation, anti-corrosion, and mitigation of magnetohydrodynamic (MHD) effects. Results have been compared to breeder zone conditions in DEMO concepts like DCLL and WCLL, showing that IFMIF-DONES can provide valuable information about the behavior of coatings under a fusion reactor environment.

Tentative

P1-070

What is the Optimal Size of Beryllium Pebbles as a Neutron Multiplier?

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The correlation between the size of neutron multipliers (ranging from 1 mm to 10 μm in diameter) and helium recoil release was studied using the PHITS code to mitigate swelling actively. The results indicated that smaller neutron multipliers enhance helium release. However, smaller spheres are more susceptible to helium irradiation from adjacent spheres. To address this issue, a 10 μm beryllium sphere filled with lead is proposed as a potential candidate for advanced tritium breeding blankets operating at high temperatures. Low-melting-point lead is expected to improve helium removal from the system. Furthermore, enhancements in the tritium breeding ratio and shielding performance of fusion blankets are also anticipated.

Tentative

P1-071

Study on Powder Synthesis with Core-shell Structure for Tritium Breeder

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Lithium orthosilicate (Li_4SiO_4) and Lithium metatitanate (Li_2TiO_3) have been studied as promising tritium breeder for fusion energy. Li_4SiO_4 has a lithium density of 0.51 g/cm^3, which is higher than the lithium density of Li_2TiO_3 (0.43 g/cm^3), resulting in higher tritium production efficiency. However, Li_4SiO_4 is prone to decomposition reaction at high temperature, resulting in low mechanical-chemical stability at the operation temperature. Recently, core-shell type Li_4SiO_4-Li_2TiO_3 pebble, coated with Li_2TiO_3 on the surface of Li_4SiO_4 pebble, has been developed to enhance the low mechanical-chemical stability. (Core: Li_4SiO_4, Shell: Li_2TiO_3) It is usually prepared in two steps: fabrication of Li_4SiO_4 pebbles and coating with Li_2TiO_3. This Li_4SiO_4-Li_2TiO_3 pebble fabrication method is not suitable from the point of view of mass production for breeding blanket. In addition, the difference in shrinkage rates between Li_4SiO_4 and Li_2TiO_3 provides a potential for separation of the interface between core and coating during fabrication and operation.

In this study, we aim to synthesize Li_4SiO_4@Li_2TiO_3 powders with a core-shell structure to improve the low thermal stability of Li_4SiO_4 and simultaneously prepare pebbles that can be mass-produced. (Core: Li_4SiO_4, Shell: Li_2TiO_3) Li_4SiO_4@Li_2TiO_3 powders were synthesized at nanoscale using hydrolysis and syn-lithiation methods. The detailed preparation process and the properties of synthesized Li_4SiO_4@Li_2TiO_3 powder will be described in this presentation.

Tentative

P1-072

Novel hybrid Li-Be ceramics, combining both functions of tritium (T) breeder (Li) and neutron multiplier (Be)

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Invalid

Tritium fuel generation in fusion reactors will most likely happen by neutron induced lithium (Li) atoms transmutation. The breeding part of the fusion reactor blanket is expected to be combined with neutron multiplier. The development of hybrid Li-Be ceramics may reduce thermal gradients while maintaining acceptable neutron multiplication and T breeding.

$Li_2Be_2O_3$ was prepared by reacting the stoichiometric mixture of Li_2CO_3 and BeO powders at high temperature. For the first time the crystal structure of Li2Be2O3 was established. The structural information allows us to utilize a first principles-based approach to investigate stability and thermal transport properties. It was demonstrated that the compound demonstrates stability and thermal conductivity comparable to previously discussed hybrid ceramics $BeLi_2O_4Ge$ and $BeLi_2O_4Si$. However, the higher density of Li and Be atoms in $Li_2Be_2O_3$ compound makes it a preferable candidate to solid breeder material.

A neutronic assessment of the new $Li_2Be_2O_3$ ceramic using OpenMC will also be presented and compared to a Li-octoplumbate ceramic. The use of Be offers greater advantages to that of Pb due to the lower neutron multiplication threshold energy of Be-9 resulting in greater tritium generation.

These results indicate that $Li_2Be_2O_3$ ceramics offer a potential route towards better matching of thermal properties with minimal functional property degradation, thereby offering better overall fuel cycle performance.

Research sponsored by the Office of Fusion Energy Sciences, U.S. Department of Energy, under contract DE-AC05-00OR22725 with UT-Battelle LLC. PDE acknowledges support from the UKAEA's Tritium Cycle Division.

P1-073

Investigation of the possibility of obtaining beryllides of rare-earth metals

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Metal beryllides are considered as promising materials for use in extreme conditions of fusion power plants, such as breeder zones of reactors, targets for the tritium cycle, and high-temperature heat dissipation elements. In this context, the study of the possibility of obtaining rare earth metal beryllides is of interest from the point of view of developing new radiation-resistant, heat-resistant and corrosion-resistant materials. Metal beryllides are considered as promising materials for use in extreme conditions of fusion power plants, such as breeder zones of reactors, targets for the tritium cycle, and high-temperature heat dissipation elements. In this context, the study of the possibility of obtaining rare earth metal beryllides is of interest from the point of view of developing new radiation-resistant, heat-resistant and corrosion-resistant materials. At the same time, compounds with some metals, in particular with elements of the lanthanide group, are studied extremely insufficiently. And the properties of some binary pairs, in particular with such elements as holmium, erbium, ytterbium, etc. are absent at all. This work presents the results of investigation of the possibility of obtaining beryllide samples with holmium and erbium by synthesis from powders. Characterization of initial powders, investigation of thermal effects of material interaction, temperature-time parameters of beryllide formation from initial materials and properties of obtained beryllide powders and samples are given.

Tentative

P1-074

Li-Pb corrosion effects on deuterium permeation in functional coatings using yttrium oxide and zirconium oxide

Kosuke Okubo[1], Wataru Inami[1], Yoshimasa Kawata[1], Takumi Chikada[1]

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In fusion reactor liquid blankets, corrosion of the functional coatings by liquid tritium breeders such as lithium-lead (Li-Pb) is an unavoidable concern. Our previous studies revealed deuterium permeation behavior and Li-Pb compatibility of several kinds of ceramic coatings; however, the effects of coating composition on corrosion and hydrogen isotope permeation behavior have not been clarified. In this study, Li-Pb exposure tests and deuterium permeation tests under Li-Pb exposure were conducted for yttrium oxide (Y_2O_3), zirconium oxide (ZrO_2), and 4 mol% yttria-stabilized zirconia (YSZ) coatings to investigate the effects of compositions on Li-Pb corrosion and hydrogen isotope permeation behaviors.

Three kinds of the coatings were fabricated by radio-frequency magnetron sputtering on reduced activation ferric/martensitic steel F82H plate substrates. The static Li-Pb exposure tests were conducted using Li-Pb eutectic alloy at 550 °C for 500 h. Deuterium permeation measurements under static Li-Pb exposure were performed using a gas-driven deuterium permeation setup with the coating exposed to Li-Pb at the deuterium-loading side at 300-600 °C.

No cracking and delamination were observed on the coating surfaces after the static Li-Pb exposure tests. Additionally, the sample surfaces were similar before and after the tests, suggesting that the thin-layer corrosion product formed on the coating surface. The YSZ coating showed five orders of magnitude lower deuterium permeation flux than the uncoated substrate, while the Y_2O_3 and ZrO_2 coatings showed two orders of magnitude lower fluxes. These results suggest that the YSZ coating enhances hydrogen isotope permeation reduction performance under Li-Pb exposure.

Tentative

P1-075

Corrosion mechanisms of ceramic-iron joint coatings in lithium-lead under thermal cycling and gamma-ray irradiation

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Functional coatings have been investigated to suppress tritium permeation, electrical conductivity, and corrosion by high-temperature tritium breeders. Recently, ceramic-iron joint coating exhibited high deuterium permeation reduction as well as certain corrosion resistance for the Li-Pb under gamma-ray irradiation. In this study, thermal durability and Li-Pb compatibility of the joint coatings were examined through thermal cycling and gamma-ray irradiation under Li-Pb exposure.

ZrO_2 and Fe_2O_3 coatings were fabricated on 316L-type stainless steel tubes by metal organic decomposition. The coated tubes were joined with iron-tube using a hot-isostatic pressing machine at 25 MPa and 550 °C. The thermal cycle test was carried out inside an infrared gold furnace with Ar and H_2 gas flowing. The temperature was within 326-550 °C, with the heating/cooling rate of 10 °C/min. The number of the thermal cycling was 10 times. Some of the samples were irradiated using a ^{60}Co gamma-ray source for 200 h at 550 °C. Then, specimens were exposure to Li-Pb exposure at 550 °C for 500 h.

The thermal-cycled specimens of the ceramic-iron joint coating showed no significant changes and cracks or delamination in surface morphology and coating color, while the single-layer ZrO_2 coating formed small cracks. The results suggest that the ceramic-iron joint coating have resistance to the thermal cycles more than the single-layer coating. The results of Li-Pb exposure following thermal cycle as well as gamma-ray irradiation behaviors of the joint coating will be discussed in the presentation.

Tentative

P1-076

Ion irradiation and implantation effects on electrochemical properties of functional ceramic coatings

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In fusion reactor liquid metal blankets, MHD pressure drop is a critical issue due to reduction of fuel efficiency and cooling capacity. Functional coatings have developed using ceramics to reduce tritium permeation and corrosion by liquid metals. During the reactor operation, the structural and functional materials will be displaced by high-energy neutrons as recoil particles and then implanted into the functional coating. Implanted ions may cause a change in the coating properties, such as electrical insulation performance. Therefore, in this study, we investigated irradiation and implantation effects on the electrical conductivity of the ceramic coatings.

Zirconium oxide coatings were fabricated by metal-organic decomposition with a dip-coating technique on reduced activation ferric/martensitic steel F82H substrates. 8.0-MeV Au ions were irradiated to the samples with displacement damage of up to 10 dpa, and 1.0-MeV Fe ions were implanted with atomic concentrations of up to 2000 appm. Thereafter, we performed electrochemical impedance spectroscopy in the frequency range of 1?105 Hz at temperatures between room temperature and 550 oC. The electrical conductivity of the samples irradiated with displacement damages of 1.0 dpa and 10 dpa and implanted 2000 appm of Fe ions showed five orders of magnitude higher than the unirradiated sample at room temperature. On the other hand, the sample with a damage concentration of 0.1 dpa and an atomic concentration of 2000 appm of Fe ions shows no decrease in electrical conductivity. These results suggest that irradiated and implanted ions increase the electrical conductivity by introducing conduction bands.

Tentative

P1-077

Development of Tritium Permeation Barriers for Non-plasma-facing Applications for Nuclear Fusion

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Commercialising fusion reactors requires effective tritium management, as tritium permeation through structural materials poses safety risks, threatens component integrity, and impacts the tritium economy. Coatings are being explored to reduce permeation while maintaining performance, with promising materials including alumina (Al_2O_3) and erbia (Er_2O_3). Er_2O_3 coatings have shown up to a five-order reduction in deuterium permeation through steel. However, neutron irradiation, pressure, and temperature induce structural transformations that hinder their performance in fusion environments. The cubic phase (A-Er_2O_3) provides excellent hydrogen barrier properties but transitions to monoclinic (B-Er_2O_3) under these conditions. While B-Er_2O_3 is more resistant to neutron damage, its hydrogen permeation resistance is lower, potentially due to defects from phase transformation rather than an inherent limitation. Stabilising the monoclinic phase in as-grown coatings is crucial for accurately assessing its permeation resistance.This study explores Er_2O_3 coatings via sputtering and advances in bulk doping strategies. Initial material development focuses on bulk samples, incorporating dopants such as tantalum and titanium to stabilise the monoclinic phase. Doped samples will undergo proton irradiation followed by thermal desorption spectroscopy to evaluate tritium retention.Results will assess dopant efficacy, the impact of proton irradiation, and tritium retention in irradiated and unirradiated samples. These findings will provide key insights into the viability of monoclinic Er_2O_3 as a tritium barrier material, guiding future coating development for fusion applications.

Tentative

P1-078

Measurement of nitrogen diffusivity in Fe-Ti alloy films by glow discharge optical emission spectroscopy

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Liquid lithium is a promising candidate for tritium breeding in fusion blankets and a target material for d-Li stripping type neutron sources, such as A-FNS (Advanced Fusion Neutron Source) and IFMIF (International Fusion Materials Irradiation Facility). However, lithium has high reactivity and strong solubility for nitrogen, and the dissolved nitrogen negatively affects its compatibility with structural materials, necessitating its removal. The chemical trapping (hot trapping) method using Fe-Ti alloy, where 5-10% titanium is dissolved in iron, is considered a practical approach. Predicting the long-term behavior of nitrogen trapping requires an understanding of its distribution within the alloy. However, due to limitations in analytical techniques, this aspect remains unclear. In this study, glow discharge optical emission spectroscopy (GD-OES) was employed to analyze the nitrogen distribution in Fe-Ti alloy thin films. A sample prepared by physical vapor deposition (PVD), in which an Fe-Ti alloy (Ti concentration $\leq$5%) was deposited, was immersed in liquid lithium containing 1000 w-ppm nitrogen at 823 K for 10 hours. By fitting the observed nitrogen distribution to the diffusion equation, the apparent diffusion coefficient was determined to be 9.3×10^{-16} m^2/s. These experimental results clarify the nitrogen distribution within the Fe-Ti alloy thin films and contribute to understanding its diffusion behavior based on the obtained apparent diffusion coefficient.

Tentative

P1-079

Study on irradiation damage and temperature monitoring applying photoluminescence properties of Er2O3 under Gamma-ray environment

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Erbium oxide (Er_2O_3) emits red and green-colored photoluminescence lights by excitation with an ultraviolet light at 365 nm. Since the intensity ratio of these peaks changes depending on the crystallinity and temperature of the Er_2O_3 materials, we are proposing to apply these optical properties to material damage and temperature monitoring system in fusion reactors.

The proposed system consists of Er_2O_3 disks attached in a reactor, optical fibers for transmission of excitation and luminescence lights, and spectrometer and UV light source placed outside a reactor. It is considered that the monitoring using the system could be performed during the maintenance period of a reactor. However, in the optical fibers exposed to induced gamma-rays in the reactor, Cherenkov lights will be produced as noises and might disturb the optical measurement.

The intensities of Cherenkov light induced in the optical fibers during the maintenance period in a DEMO reactor were estimated by neutron, gamma-ray and electron transport calculation with the PHITS code. The result showed that the intensities of Cherenkov light could be approximately 10,000 times greater than the photoluminescence intensities emitted from Er_2O_3 discs in our present study using UV LED excitation lights.

For mitigation of the influence of Cherenkov light, enhancement of the photoluminescence intensities by using Er2O3 doped Y_2O_3 discs could be effective since concentration quenching occurred in pure Er_2O_3 discs is considered suppressed. The magnitude of the intensity enhancement and responses of the discs to the crystallinity and temperature will be reported.

Tentative

P1-080

Permeation and corrosion barrier capabilities of amorphous SiC coatings under PbLi exposure

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The development of multifunctional coatings that act as Tritium (T) permeation and Lead-Lithium (PbLi) corrosion barriers is necessary to ensure the optimal operation of nuclear fusion reactors with PbLi-based breeding blankets (BB). The main role of the protective coating in these designs is to prevent T permeation into the coolant and to protect the EUROFER steel from liquid metal embrittlement.

Amorphous Silicon Carbide (a-SiC) has been proposed as a candidate because of its properties (high thermal stress resistance, low reactivity with Li and low activation). So far, its T permeation barrier ability has been demonstrated, obtaining a permeation reduction factor (PRF) of 104, while withstanding e-irradiation and thermal load effects. Our current research is focused on investigating the performance of a-SiC coatings deposited by RF sputtering under PbLi exposure.

These experiments are performed in CICLO (CIEMAT corrosion loop) at LNF-CIEMAT, where two recipes of a-SiC coatings are subjected to PbLi flowing at 0.6 m/s for up to 6000h at a temperature of 450°C. After that, we characterize the morphology, microstructure and elemental composition as a function of depth of the coatings, with special interest in the possible Li diffusion inside the coatings. Finally, the optimized coating is submitted to a Deuterium (D) permeation experiment under exposure to PbLi at Shizuoka University; which provides an updated PRF value under relevant BB conditions.

Tentative

P1-081

Deuterium permeation and Li-Pb corrosion behaviors of single- and multi-layer coatings using Y-Al-O system

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Yttria- and alumina-based coatings are widely explored for their potential to reduce deuterium permeation and corrosion in liquid metal environments, making them promising candidates for fusion reactor applications. This study investigates deuterium permeation and Li-Pb corrosion behaviors of single- and multi-layer coatings using Y-Al-O system with the same total thickness of 500 nm but different layer structures. The coatings were fabricated using physical vapor deposition on reduced-activation ferritic/martensitic steel F82H substrates. Deuterium permeation experiments revealed that multi-layer coatings with the highest number of layers exhibited higher activation energy compared to the single-layer coatings, indicating that adding more layers can effectively mitigate hydrogen isotope permeation. The multi-layer coatings applied to both sides of the substrate exhibited five orders of magnitude smaller deuterium permeation flux than uncoated substrate as well as higher activation energy of permeation among the all coatings. Corrosion studies in liquid Li-Pb demonstrated that the YAlO coating exhibited enhanced stability due to its amorphous nature under prolonged exposure. The multilayer structure effectively suppressed Li-Pb infiltration, with minimal cracking and peeling of the layers, suggesting that interfacial engineering plays a key role in optimizing barrier performance. These findings provide valuable insights into coating design strategies for advanced fusion reactor systems, balancing hydrogen isotope control, and structural integrity in extreme environments.

Tentative

P1-082

One-step fabrication of AlPO4 nanosheet reinforced α-Al2O3 coatings with improved fracture toughness and hydrogen resistance

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Oxide coatings have great potential in protecting vital metal structural components in hydrogen energy and nuclear fusion reactors. Hydrogen permeation resistance and fracture toughness are two key characteristics that determine the performance and life span of coatings. However, there is still no strategy to simultaneously enhance the hydrogen permeation resistance and fracture toughness of coatings. Herein, AlPO4 nanosheets are in situ formed in α-Al_2O_3-based oxide coatings to simultaneously realize enhanced hydrogen permeation resistance and fracture toughness. The in-situ formation method guarantees considerable quantities of homogeneously distributed AlPO4 nanosheets with coherent interfaces with α-Al_2O_3. Owing to the positive reinforcing effect of AlPO4 nanosheets, the fracture toughness of the coating increases by 3 times, achieving an ultrahigh KIC value of 7.1 MPa/m^2. Meanwhile, the hydrogen permeation resistance of the coating increases by 78%, reaching as high as 3440 times that of the steel substrate. Additionally, the Cr-P bonds formed between the coating and substrate ensure their good bonding, with their bonding strength increasing up to 36 MPa. The combination of high hydrogen permeation resistance, high fracture toughness, and high bonding strength makes the AlPO4 nanosheet-reinforced coating a promising alternative for reducing hydrogen permeation in various hydrogen-related fields. This study also provides critical insights and practical guidelines for constructing high-performance functional coatings via nanosheet reinforcement.

Tentative

P1-083

HIP diffusion bonding of FeCrAl and RAFM steel for preparation of tritium permeation barrier

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Existing techniques for preparing tritium barrier layers face significant challenges in achieving engineering-scale fabrication within complex flow channels. To address this, we propose an innovative process strategy that first joins FeCrAl alloy with reduced activation ferritic/martensitic (RAFM) steel, followed by in situ oxidation of the FeCrAl surface to form the tritium barrier functional layer. We systematically investigate the hot isostatic pressing (HIP) welding process for FeCrAl and RAFM steel, revealing that direct bonding of FeCrAl with CLF-1 steel results in the formation of large-sized AlN precipitates and interfacial pores, severely compromising joint mechanical properties. By introducing a Ni interlayer, we effectively suppress AlN growth and promote fine, dispersed NiAl and Ni3Al strengthening phases near the interface, significantly enhancing joint performance. Additionally, by selecting IPFs steel with low nitrogen content, we eliminate interfacial defects and facilitate the formation of columnar grains with high Schmid factors, improving joint ductility. Leveraging these insights, we successfully fabricate a modular structure, achieving integrated joining of tungsten, steel, and tritium barrier materials. The FeCrAl/RAFM steel interface demonstrates excellent performance under simulated service conditions and high heat load tests in the EAST tokamak, exhibiting no cracking, while the modular structure exhibits significant tritium barrier effectiveness. This work advances the engineering application of tritium barrier layers in complex flow channels for fusion reactor systems.

Invitation

P1-084

Development of ceramic coatings as protective layers for structural vanadium-based alloys

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Any materials that are to be considered for the construction of future fusion energy reactors will need to sustain an extremely challenging environment. The materials constituting the breeding blanket structure in a tokamak design, specifically, need to have electrically insulating characteristics to minimise magneto-hydrodynamic effects, compatibility with high-temperature liquid metal coolants, and effective barrier properties against the permeation of tritium, to prevent fuel losses. These multiple requirements justify the need of applying functional coatings, able to confer the desired properties to the structural material of choice.

This work focusses on the functionalisation, via the application of coatings, of the V-4Cr-4Ti (V44) alloy, which is a candidate structural material for the blanket due to its relative resistance to irradiation, high thermal conductivity and good mechanical properties. Ceramic coatings, which are sought after due to their potential barrier properties against hydrogen isotopes, were deposited via reactive magnetron sputtering on coupons made of a titanium-based alloy, used as a proxy for the V44 material.

The initial screening of the coatings consisted in the study of their adhesion strength and the analysis of their crystalline structures, both in their as-deposited form and after an in-vacuum thermal treatment. The potential of the best performing coatings was further assessed, once deposited on V44 coupons, both via evaluating their tritium barrier and retention properties, using tritium gas soaking and subsequent thermal desorption spectroscopy, and via determining their compatibility with high-temperature liquid metals, by means of a direct corrosion test.

Tentative

P1-085

A mesoscale phase-field model of intergranular liquid lithium corrosion of ferritic/martensitic steels

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Liquid lithium (Li) is being envisaged as a breeder material attributed to its beneficial qualities, including absence of radiation damage and superior thermal properties. However, Li yields severe corrosive environments for structural materials leading to intergranular corrosion (IGC) via solubility-driven dissolution. This results in weight loss, thus heightening the risk of accumulated activated material, and deterioration of the material's structural integrity through weakening of grain boundaries (GBs). Phase-field modeling (PFM), a thermodynamic consistent computational technique, is utilized to simulate the IGC process. Through a diffuse interfacial region, PFM tracks the interface between two phases implicitly using an order parameter thereby eliminating the need for restrictive interfacial boundary conditions. Further, the diffuse interface allows for the exploitation of physical material properties that ultimately governs the diffusion-dependent intergranular penetration of Li. The resultant dissolution of alloying elements is tracked to evaluate the severity of corrosion. GBs and grains are distinguished via an additional order parameter, solved independently to the primary kinematic variables, thereby decreasing computational expense. The model is validated against experimental measurements for ferritic/martensitic steel exposed to static liquid Li. Thereafter, the microstructural influence on the corrosion process was surveyed. Near-surface GB density showed to play a non-trivial contribution to the material's performance. Yet, it was evident the grain size, which dictates the bulk GB density, ultimately governs the susceptibility to IGC. Subsequently, the incorporation of saturation shifted the behavior of the material whereby the microstructural influence on corrosion depth became distinct in turn dampening the correlation with weight loss.

Invitation

P1-086

Oxidation behavior of neutron-irradiated tungsten and its dependence on irradiation temperature

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In the event of a loss-of-vacuum accident in a fusion reactor, the reaction of high temperature tungsten (W) with oxygen or water vapor occurs. Oxides of W and rhenium (Re), a transmutation product from W, have high vapor pressure, and there is a concern about the release of radioactive materials during the accident. Therefore, the oxidation of non-irradiated W, W-Re alloy simulating transmutation and neutron-irradiated W were investigated by thermogravimetry under the Japan-US FRONTIER program.

The samples used were discs of non-irradiated W, non-irradiated W-10%Re alloy and neutron-irradiated W. Neutron irradiation was performed with thermal-neutron shielding in the High Flux Isotope Reactor (HFIR) at ORNL at 633 °C, 836 °C and 1121 °C to 0.44, 0.74 and 0.65 dpa, respectively. The oxidation was carried out under flowing Ar-20%O_2 gas at 600-900 °C in a thermogravimetry device.

The W sample irradiated with neutrons at 633 °C showed comparable weight change with non-irradiated W, indicating defects induced by the irradiation had minimal effects on the oxidation rate. The neutron irradiation at the higher temperatures led to slightly larger (factor of 2) weight gain at high oxidation temperatures (850 °C and 900 °C). However, clear difference was observed in the morphology of oxide layer; black dense oxide layer was formed on the irradiated W while the non-irradiated W had yellow pulverized oxide. The W-10%Re alloy had black dense oxide layer, as observed for the irradiated W. These results indicate the effects of transmutation Re on the oxide morphology.

Tentative

P1-087

Reaction products at the interface between supercritical carbon dioxide and stainless steel

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Supercritical carbon dioxide (S-CO_2) is expected to be used as a coolant in fusion power plants, because S-CO_2 is non-toxic and stable as a working fluid, and turbines using S-CO_2 have the advantages of compactness and high-power generation efficiency as a system. S-CO_2 is considered as a secondary coolant in the power generation system with a molten salt blanket because of its superior power generation efficiency at the operating temperature of the molten salt. However, since the molten salt has low hydrogen solubility, tritium permeation from the primary system to the secondary system through the heat exchanger is not avoided. In this study, to understand tritium behavior permeated into the S-CO_2 as the secondary coolant, mass transfer and chemical reactions at the interface between S-CO_2 and stainless steel were investigated.

CO_2 gas including H_2 with low concentration was supplied into a SUS316 container and heating up to 400°C and then the pressure increased to 11 MPa, which is a supercritical condition. A portion of the CO_2 gas was sampled periodically, and gaseous products were investigated using gas chromatography.

CO and CH_4 were detected as gaseous products in addition to the initially enclosed CO_2 and H_2; the CO is considered to be generated by the reduction reaction of CO_2 on the metal surface. CH_4 is considered to be formed by the reaction between C deposited by the reduction reaction of CO_2 on the metal surface and the enclosed H_2. The chemical reaction on the metal surface will be discussed.

Tentative

P1-088

Mobilization behavior of Re oxide sublimated from W-Re alloys during high-temperature oxidation

Takahiro Aoki[1], Teppei Otsuka[1]

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W is transmuted into Re, Os, Ta, etc., and alloyed by neutron irradiation in fusion reactors. In a synergistic accident of loss of vacuum and coolant, W alloy could be heated and oxidized with air due to decay heat. During oxidation, mobilization of radioactive Re oxides were concerned due to its low sublimation point of 523 K. This study investigated the release and deposition behavior of Re oxides sublimated from W alloys with Re 3 at% and 25 at% during high-temperature oxidation.

The W-Re alloy was placed at a center in a long quartz glass tube covered with an electric resistant furnace. Deposition collectors of small quartz glass plates were placed from the heater outlet to downstream. The experiments were conducted in an Ar-20%O_2 gas flow with 100 sccm at 1273 K, resulting in a temperature gradient to downstream. After the oxidation, the deposition amount was evaluated by measuring the weight gain of the glass plates.

All depositions were confirmed to be Re oxide by using SEM-EDS and ICP-OES. The weight measurement gave a semiquantitative distribution; no deposition was found on the glass plate above 523 K. The deposition amount increased from 2 cm and reached a peak at 5 cm from the heater outlet. The small weight gain continued far beyond 15 cm, indicating long-range mobilization of the Re oxide.

In the presentation, we discuss the influences of gas flow conditions, temperatures, and moisture in the gas to establish predictive models of mobilization of the Re oxide.

Tentative

P1-089

Evaluation of the integrity of CVD-SiC coatings after exposure to high temperature and high pressure water.

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Silicon Carbide (SiC), known for its excellent mechanical properties and chemical stability at high temperatures, is expected to be used as a structural material in reactors, such as accident-tolerant fuel cladding for light water reactors and the dual coolant lead lithium blanket module for fusion reactors. However, SiC reacts with high-temperature, high- pressure coolant water or liquid metal coolants, leading to corrosion under irradiation. Therefore, this necessitates the development of corrosion control technologies. In this study, we compared the integrity of chemical vapor deposited(CVD)-SiC coatings, a candidate for corrosion-resistant coatings for SiC, after exposure to high-temperature, high-pressure water. CVD-SiC coatings, which had been deposited on graphite substrates at temperatures of 1050°C, 1150°C, and 1250°C by using a CVD system equipped with a high- output laser, were used as test specimens. Corrosion tests were conducted at 320°C and 14-15 MPa under saturated dissolved oxygen for 3 and 24 hours, and the samples were examined by SEM after the tests. The results showed that higher deposition temperatures improved the corrosion resistance of CVD-SiC coatings. Especially, the sample deposited at 1250°C, which showed minimal corrosion pitting and slight surface change, retained its structural integrity relatively well after testing. These results suggest that high-temperature deposition enhances crystallinity and reduces defects, thereby suppressing grain boundary corrosion. The corrosion mechanisms and considerations for improving coating integrity are discussed based on TEM observations.

Tentative

P1-090

Compatibility of composite insulators with high temperature liquid Lithium

Claude De LamaterBrotherton[1], Marie Romedenne[1], Sergey Smolentsev[1], Roger Miller[1]

[1]Oak Ridge National Laboratory

Open-surface Li flow divertor designs have been proposed to protect divertor structural steel structures from high heat fluxes (10 MW/m2) expected during fusion reactor operation. To accommodate magnetohydrodynamic (MHD) pressure drop and compatibility challenges produced by flowing high temperature Li, steel-insulator-steel composite structures are being developed. The thickness of the steel must be optimized to minimize MHD pressure losses and to maximize high temperature Li compatibility. Composite structures of alumina felt welded between 410 stainless steel foils are proposed. The 410 steel was selected as a surrogate for reduced activation ferritic martensitic (RAFM) steel in fabricating experimental sample composites. Composite structures and SS3-type steel specimens of various thicknesses (63, 100, 500 μm) were exposed in static Li for 1,000 h at 500 °C. Change in electrical resistivity before and after Li exposure was evaluated for exposure temperatures between 400 and 600 °C using a high vacuum furnace to prevent oxidation of the foils. Room temperature tensile properties and microstructural characterization were performed to study the impact of high temperature Li and foil thickness on strength, ductility, nitridation and dissolution behavior. The results will be discussed in the context of understanding the composite-Li system and the potential use of these composites in fusion reactor service.

Tentative

P1-091

Impact of high temperature liquid metal exposure on design of structural first wall and blanket materials

Marie Michelle Romedenne[1], Claude De LamaterBrotherton[1], Yi-Feng Su[1], Nick Russell[1], Masatoshi Kondo[2], Naoko Oono[3], Yutai Katoh[1], Yuji Hatano[4], Aaron Washington[5], Sandeep Irukuvarghula[6], Samara Levine[6], Jim Pickles[6], Bruce A. Pint[1]

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Liquid metals (e.g. Li, PbLi, Sn) are proposed in fusion devices for their ability to remove the extreme heat of first wall and structural components and to breed tritium. Decades of research on heat and mass transfer modeling has enabled the fusion industry to optimize reactor design. However, the selection of suitable materials and components that is usually based on neutron activation and irradiation resistance has started to rely heavily on liquid metal testing under various reactor simulated conditions. Through targeted experiments, key parameters governing alloy degradation mechanisms in high temperature liquid metals can be elucidated to accelerate material selection and qualification. Previous research conducted between 400-600°C identified increased mass loss kinetics with coolant velocity and non-metallic impurities in the liquid metals. Dissolution rates were mostly parabolic and greater for ferritic than austenitic steels and for PbLi than Li. More recently, static and convective flow liquid metal experiments, for up to 2,000 h and maximum temperature up to 700 °C, identified differential material mass transfer between functional SiC and structural steels and promising coatings. In addition, preferential dissolution based on precipitate nature and alloy chemistry was determined. The interplay between magnetic field, liquid metal chemistry and alloy composition was evaluated in liquid metals containing various impurities. Finally, combined stress-corrosion and irradiation-corrosion effects in Li, PbLi and Sn are investigated using multiscale characterization tools and thermodynamic modeling. Results of these various studies will be discussed in terms of selecting suitable materials, and the impact on first wall and blanket component design

Tentative

P1-092

Ultra-efficient hydrogen isotope oxidation by Pt nanoclusters confined in Silicalite-1 zeolite

Xingwen Feng[1], Jiangfeng Song[1], Yan Shi[1], Linsen Zhou[1], Xianglin Wang[1]

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Catalytic oxidation plays a crucial role in the efficient treatment of hydrogen isotopes, with the key technical challenge being the development of high-performance catalysts to enhance isotope removal efficiency, thereby reducing environmental pollution and ensuring public radiation safety. Herein, we report the strategic control of platinum nanoclusters confined within silicalite-1 zeolites to enhance the high removal efficiency for hydrogen isotope gases. Incorporating a single lanthanum oxide (LaOx) into Pt nanoclusters adapts the local charge of adjacent Pt atoms, significantly altering their electronic structure. A comprehensive study of the correlation between this specific local charge adaptation and catalytic performance was conducted through experimental and theoretical analysis. The existence of individual LaOx species was confirmed by X-ray absorption spectroscopy and aberration-corrected electron microscopy, and the performance enhancement mechanism was supported by theoretical calculations. This embedded single LaOx both facilitates local charge adjustment and supplies abundant active oxygen species, leading to performance enhancement by reducing the energy barrier of rate-determining step (O* + H → OH*), in accordance with the Mars-van Krevelen mechanism. Consequently, during a surprisingly long-term restart performance test lasting up to ~267 d, the PtLaOx@S-1 catalyst maintained a high conversion rate of over 99% at 50 °C and a space velocity of 48,000 mL・g^{-1}・h^{-1}. This study highlights the potential of individual LaOx for enhancing hydrogen isotope oxidation while offering new insights into the reaction mechanisms of related catalysts.

Invitation

P1-093

Degradation behaviour of Vanadium-Lithium Tritium Breeder Blanket System under Fusion-relevant Neutrons

Joven J.H Lim[1]

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Understanding the effects of neutron irradiation on materials used in fusion reactors is critical for designing robust and efficient breeder blanket systems. The lithium/vanadium (Li/V) self-cooled breeder blanket concept offers significant advantages, including high-temperature operation for improved power conversion efficiency and high tritium yield from pure lithium. However, key challenges remain in assessing the impact of neutron irradiation on vanadium alloys, particularly in liquid lithium environments, where irradiation, corrosion, microstructural evolution, and tritium retention may affect long-term performance.

To address these challenges, a series of controlled irradiation experiments has been conducted using the Tunable Micro-Breeder Neutron Emission Device (TuNED), developed by the UK Atomic Energy Authority (UKAEA) in collaboration with the University of Birmingham. TuNED provides a fusion-relevant neutron field in a controlled environment, enabling direct investigations into irradiation-induced defects, tritium transport, and material degradation in vanadium alloys exposed to liquid lithium at elevated temperatures. These experiments provide essential data on how neutron irradiation influences tritium solubility, diffusivity, and retention.

Complementary modelling and simulation techniques are employed to predict material behaviour under prolonged neutron exposure. Atomistic simulations using machine-learning interatomic potentials are performed to interpret experimental findings of neutron-induced damage and its effect on tritium retention. This integrated approach enhances understanding of material performance, informing the development of vanadium alloys optimized for tritium management and structural integrity in fusion breeder blankets. The outcomes of this work will be discussed further.

Tentative

P1-094 "Presenting Author: Takaaki Koyanagi"

Corrosion of monolithic and composite SiC by molten Be-bearing salt

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Commercialization efforts for fusion reactors require tritium self-sufficiency and efficient heat transfer to power generation systems. High temperature molten salts such as LiF-2BeF_2 (FLiBe) are attractive both as the breeder materials and as coolant. Understanding material compatibility between breeder liquid and various materials (plasma facing components, channel flow inserts, coolant loop piping) is fundamental in developing breeder technology readiness level. The interactions between FLiBe and SiC materials were examined following static molten salt exposure for 1000 h at up to 750°C. Materials tested included single-crystal 4H-SiC, three polycrystalline SiC grades grown by chemical vapor deposition, and two composites using Tyranno SA3 or Hi-Nicalon type-S SiC fibers and chemical vapor infiltration SiC matrix. Overall, minimal weight loss and surface etching were observed regardless of the material type. Localized pitting and nonuniform Si removal from surface regions were associated with trace metal impurities (i.e., Fe and Ni) and oxygen. Cross-sectional observation revealed minimal subsurface changes in the microstructure. Composite samples containing trace aluminum appeared to be more corrosion resistant than Al-free samples. The presence of pyrolytic carbon interphase material between the matrix and fibers in composite samples appeared to limit the expansion of corrosion pitting. Based on the results of experimental and thermodynamic analysis, a corrosion mechanism for SiC materials is proposed where the types of impurities within both the salt and SiC material play a key role in the phase stability of the structural material - specifically, differentiating between reactive impurities (e.g., Fe, Ni, O) and stabilizing impurities (e.g., Al).

Tentative

P1-096

Tritiated water treatment of fusion reactor enhanced by metal-organic frameworks

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In the operation of fusion reactors, radioactive tritiated wastewater will be produced, which will pose potential threats to people and the environment. Water distillation (WD) is the most widely used treatment processes of tritiated water. However, the low separation factor and high separation energy consumption of WD restrict their application in tritiated water treatment. It is urgent to improve the separation performance of tritiated water by improving the separation factor and strengthening the gas-liquid two-phase exchange of tritiated water.

Herein, we successfully prepared the modified interface of metal-organic frameworks (MOFs) with superhydrophilicity and tritiated water enrichment performance by solution method to optimize the separation process of tritiated water. The separation experiment of tritiated water, quantitative calculation and computational fluid dynamics (CFD) theoretical simulation were used to study the improvement of the separation performance of tritiated water by the modified interface and the structure-activity relationship. The superhydrophilicity of the modified interface comes from the microporous hydrophilicity of the MOFs, and the superhydrophilicity and superaerophobicity enhance the phase exchange between gas reactants, which lead to the better droplets spreading and phase transfer. The distillation experiment of tritiated water shows that the HETP of unmodified packing is 5.0 cm and the minimum HETP of MOFs modified packing is 1.5 cm. The research results show potential in the treatment of tritiated wastewater from nuclear power plants and other tritium treatment systems.

[1] Microporous and Mesoporous Materials, 2023, 348, 112387.

[2] Separation and Purification Technology, 2023, 324, 124611.

[3] Separation and Purification Technology, 2024, 334, 126025.

Tentative

P1-097

Hydrogen Capacity Evaluation of the ZrCo Tritium Storage Bed in Fusion Power Plant and Verification of Its Repeated Recovery Effect

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Metal hydride storage beds will store the tritium for use in long-term operations of future Fusion Pilot Plant (FPP). Zirconium-cobalt (ZrCo) is a candidate material for tritium storage beds because its hydrogen equilibrium pressure is appropriate for FPP operations. The lattice structure changes of ZrCo during hydrogen loading and unloading lead to capacity degradation of ZrCo via disproportionation. It was reported that the decreased capacity can be recovered by regeneration by evacuation to 10-4 Pa under 500 °C. Regeneration is necessary for an FPP to maintain the performance of ZrCo. However, the level of capacity recovery during repeated regeneration under cycle operations has not been extensively investigated.

In this study, experiments were conducted to evaluate hydrogen storage capacity of ZrCo by exposing samples to hydrogen gas under controlled pressure and temperature. The regeneration was performed in moderate conditions after vacuum pumping down to 10-1 Pa. The loss and recovery capacities of ZrCo were analysed by measuring the pressure change. The long-term stability of the ZrCo tritium storage bed was assessed by repeating cycles of capacity loss and recovery. The results show that hydrogen capacity of ZrCo recovered to 70 % of its initial value after the first absorption and remained constant in every subsequent regeneration. It indicates that ZrCo tritium storage bed can sustain long-term operation in FPPs requiring years-long operation with predictable performance and without significant capacity degradation. Results of the composition and crystal structures of ZrCo after each treatment analyzed by X-Ray Diffraction will be presented.

Tentative

P1-100

Development and De-risking of Tritium Extraction Processes by Flow-Loop Integration

Luke Olson[1], Brenda Garcia-Diaz[1], Christopher Dandeneau[1], Adam Gootgeld[1], Marie Romedenne[2], Yury Osetskiy[2], German Samolyuk[2], Chase Taylor[3], Thomas Fuerst[3], Dilpuneet Aidhy[4], Adrien Couet[5]

[1]Savannah River National Laboratory, [2]Oak Ridge National Laboratory, [3]Idaho National Laboratory, [4]Clemson University, [5]University of Wisconsin-Madison

Savannah River National Laboratory (SRNL) is leading a team to advance materials and improve the technical readiness of Li, PbLi, and LiF-BeF2 (FLiBe) blanket tritium removal technologies to the point of insertion in a small-scale flow loop. Current materials and methods will be optimized for insertion into flow loops operated at Idaho National Laboratory (INL). SRNL is investigating tritium extraction from Li and PbLi via direct electrolysis, where LiT is electrochemically split while in the molten metal, thereby eliminating a molten salt contacting step proposed in alternative techniques such as the Maroni process. While the direct electrolysis process can potentially result in very low tritium levels in the blanket, lithium compatible solid electrolytes are needed to ensure reliable functionality. Simultaneous to efforts at SRNL to create an integrated system that is more easily integrated into a loop, Oak Ridge National Laboratory (ORNL) will work to evaluate and improve the chemical durability of solid electrolyte materials in Li using high throughput methods, with final products being tested at SRNL. Clemson will also provide theoretical guidance for solid electrolyte formulations through use of computational methods. The most durable electrolytes from testing will ultimately be sent to INL for PbLi flow loop integration. The SRNL molten salt tritium extraction component will use a Combined Redox Control and Extraction (CoRExt) process. The CoRExt method could potentially enable very low tritium levels while simultaneously conditioning the salt to be less corrosive. The general processes and results to date will be discussed.

Tentative

P1-101

Effect of hydrogen peroxide on corrosion properties of fusion reactor materials

Motoki Nakajima[1], Takashi Nozawa[1]

[1]QST

The use of reduced activation ferritic/martensitic (RAFM) steel and CuCrZr alloy is planned as water cooling channel materials for in-vessel components of fusion reactors. There is a need to provide a corrosion database of fusion reactor materials in high-temperature, high-pressure water. On the other hand, it is expected that the environment will change due to the water radiolysis by fusion neutrons. In particular, the effect of hydrogen peroxide produced by water radiolysis on the corrosion characteristics of fusion reactor materials is not yet clear. In this study, corrosion tests were conducted using RAFM F82H and CuCrZr alloy in high-temperature, high-pressure water with hydrogen peroxide injection, and the effect of hydrogen peroxide on corrosion characteristics was evaluated. In the static corrosion of F82H, the corrosion was suppressed with the increase of hydrogen peroxide injection up to 10 ppm/h. On the other hand, it was found that the amount of corrosion increased with the addition of hydrogen peroxide in the copper alloy. In the case of the F82H, the surface oxide formed densely under the hydrogen peroxide addition conditions, and it was thought that this acted as a protective film and suppressed the amount of corrosion. However, in the case of the copper alloy, it was thought that the surface corrosion organisms did not act protectively, and that this accelerated the corrosion. In this presentation, we will discuss the degree of influence of hydrogen peroxide on corrosion characteristics, considering factors such as the effect of water flow.

Tentative

P1-102

Microstructural analysis of oxidised Al-base coated EUROFER97 for corrosion protection

Liberato Volpe[1], Anastacia Wright[2], Benedict Keates[2], Zhexin Cui[2], Eric Prestat[1], Sergio Lozano-Perez[2], Joven Lim[1]

[1]UK Atomic Energy Authority, [2]University of Oxford

The reduced activation ferritic/martensitic (RAF/M) Eurofer97 steel is the candidate material for the water-cooled lithium lead breeder blanket (WC-LL BB) section of the European DEMO reactor, however the synergistic effect of the high temperature flowing coolant, tritium embrittlement and ionising radiation can drastically reduce the lifetime of the Eurofer97. Therefore, it is necessary to develop self-passivating coatings that can positively interact with the harsh environment and extend the lifetime of the Eurofer97 during in-service operation. In this study, Eurofer97 was firstly coated with Al-base coatings using either a chemical vapour deposition (CVD at 700 °C and 750 °C) or electroplating process (room temperature); and then oxidised in a 2-step process using the same tempering (980 °C) and normalisation (760 °C) temperature used for Eurofer97. The oxidation temperatures were used to obtain a stable and protective Al-base oxide layer without altering the microstructure of the Eurofer97. Advanced microscopy characterizations revealed the formation of a complex near surface region microstructure formed by an outer α-Al_2O_3 layer over an intermetallic Al-Fe-rich interdiffusion microstructure formed by columnar grains (60 - 90 μm). However, the interdiffusion layer was highly decorated with voids formed during the oxidation process, that might act as high stress intensity points during in-service operations. Therefore, further optimisations on the oxidation process are still needed to improve the resulting oxidised microstructure of Eurofer97.

Tentative

P1-103

Experimental Design for Assessing the Performance of Structural Materials and Components in Liquid Lithium Environments

Julia Navas Gomez[1], Elisabetta Carella[1], Alfonso Ros[2], victor Villamayor[2], Raul Nunez[2], Joaquin Molla[1]

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Tentative

Liquid lithium is crucial in the fusion field due to its properties, such as the ability to breed tritium, enhance plasma confinement, high-energy neutrons generator, and minimize irradiation damage to reactor components. However, lithium reactivity is more pronounced in its liquid state at high temperatures. Developing lithium-based technologies for fusion requires materials capable of withstanding these extreme conditions.

To address this challenge, the Metal Liquids Laboratory at LNF-CIEMAT has developed the ELIOS facility (Experimental facility on LIthium cOmpatibility with materialS). This facility is designed to evaluate the behavior of various materials after exposure to molten lithium at high temperatures and under controlled pressure conditions for extended periods. Following exposure, mechanical and electrical properties, microstructure, and chemical composition of these materials will be analyzed. Additionally, ELIOS enables the study of impurities influence on the corrosive behavior of lithium previously contaminated.

To perform these experiments, the samples will be placed in sealed containers filled with lithium, which will be heated in small ovens under an inert atmosphere to ensure safety in case of lithium leakage. A dedicated control and data acquisition system has been implemented to regulate the test exposure parameters. Furthermore, the containers can be sealed with gaskets, which can also be assessed by monitoring pressure fluctuations that act as indicators of structural failure, allowing for estimating the components' lifetime.

This work will provide critical data to support the development of more resilient materials and components for liquid lithium systems in fusion applications, contributing to the advancement of future fusion reactors.

P1-105

The use of Yttria-stabilized zirconia for the manufacturing of an electrochemical hydrogen sensor for liquid metals

Julian Patino[1], Belit Garcinuno[1], Montserrat Martin[1], David Barras[2], David Rapisarda[1]

[1]CIEMAT, [2]CIEMAT / UCM

Different liquid metals (lithium, lead-lithium alloy, sodium) are commonly used within the fusion and fission fields for various purposes -e.g. blanket, coolant, neutron multiplying, radiation moderator, breeder, etc.-. Additionally, hydrogen isotopes (protium, deuterium, and tritium) are produced in both types of reactors and due to their physicochemical properties, they can diffuse throughout the reactor materials, leading to several issues such as embrittlement of structural materials, loss of efficiency or fuel, concerns regarding radiation protection and nuclear safety, or risk of explosion. For this reason, on-line monitoring of hydrogen in liquid metals is a critical and challenging aspect in both fusion and fission reactors.Electrochemical sensors are widely extended as on-line sensors due to their fast measurement capabilities, accuracy, and sensitivity. In the Liquid Metals Laboratory (LML) at CIEMAT, an ElectroChemical Hydrogen Sensor (ECHS) for liquid metals has been designed, manufactured, and tested. This sensor consists of two electrodes -one serving as the reference one- an electrolyte, and a semipermeable membrane. A peculiarity of the ECHS is that its electrolyte and membrane are combined into a single component, a solid electrolyte. The design presented in this work employs an internal reference system of Li-LiH and a solid electrolyte made of Yttria-stabilized Zirconia (ZrO_2 · Y_2O_3, YSZ), a well-known ceramic material with hydrogen migration properties and good chemical compatibility with lithium, lead-lithium eutectic alloy, or sodium. The work carried out at CIEMAT includes three main stages: the design, a preliminary evaluation of materials, and the final validation of the manufactured device under fusion-relevant conditions.

Tentative

P1-106

Thermogravimetry analysis for the reaction of tungsten and tungsten-rhenium alloy under oxygen atmosphere in 873-1173 K

Makoto Kobyashi[1], Teppei Otsuka[2], Yasuhisa Oya[3], Azusa Matsumoto[4], Yuji Hatano[4], Masashi Shimada[5], Hanns Gielt[5], Larry Loftus[5]

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Tungsten (W) materials, to be used in the first wall of fusion reactors, will be exposed to neutrons. Over time, this exposure will lead to radioactivation due to transmutation elements such as rhenium (Re), osmium, and tantalum isotopes. Under high-temperature and highly oxidative environments, such as in a loss-of-vacuum accident, W and its transmutation elements may form volatile oxides to be released. This study focuses on the effects of Re, the primary transmutation element in tungsten, on its oxidation behavior using thermogravimetric analysis (TGA).

Disc-shaped Polycrystalline W and W containing 10% Re (W-10%Re), with a diameter of 6 mm and a thickness of 0.5 mm, were used as test samples. Each sample was placed in a TGA device, and weight changes were measured under an Ar-20% O_2 atmosphere within a temperature range of 873-1173 K for 150 min. The surface morphology of the samples after TGA analysis was observed using scanning electron microscopy.

The results showed that the sample weight increased linearly with time across the tested temperature range. The rate of weight gain for W-10%Re was lower than that of pure W, likely due to the release of volatile Re oxides formed during TGA analysis. The metallic and reflective surface of the specimens transformed into a yellowed and rough texture for W due to the formation of WO3. In contrast, the surface of W-10%Re also became rough but exhibited a black coloration.

In this presentation, we will discuss the oxidation kinetics of W and Re in detail.

Tentative

P1-107

Corrosion Behavior and Characterization of Ni- and Ti-Based Alloys in Molten Salt and Liquid Metal Environments

Hongfei Liu[1]

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The corrosion behavior of nickel- (Ni) and titanium-based (Ti) alloys in extreme environments, such as molten salts and liquid metals, is a critical factor in determining their suitability for advanced energy and industrial applications. This study investigates the corrosion mechanisms, phase stability, and microstructural evolution of Ni- and Ti-based alloys when exposed to molten salts and liquid metal environments. Using advanced characterization techniques, including scanning electron microscopy (SEM), energy-dispersive X-ray spectroscopy (EDS), X-ray diffraction (XRD), electron backscattering diffraction (EBSD), and transmission electron microscopy (TEM) we analyze the surface degradation, oxide formation, and elemental diffusion patterns. The findings highlight the role of alloy composition, temperature, and exposure time in influencing corrosion resistance. Comparative analysis of corrosion rates and degradation products provides insights into material performance and potential mitigation strategies. These results contribute to the development of corrosion-resistant alloys for applications in nuclear reactors, concentrated solar power systems, and other high-temperature technologies.

Tentative

P1-108

Reuse of yttrium as a getter material to retain hydrogen isotopes in an impurity trap present in the ICS of IFMIF-DONES

Maria Sanchez Arenillas[1], Joaquin Molla[1], Maika Martinez Guareno[2], Sebastian Hendricks[1], Manuel Perez Mendoza[2], Fernando Sanchez[1], Marcelo Roldan[1], Belit Garcinuno[1]

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The development of effective hydrogen isotopes-trapping techniques is crucial to ensure the purity and performance of molten lithium loops. One of the key components of these circuits is the hydrogen trap, which contains a reactive metal capable of forming thermodynamically stable compounds with hydrogen isotopes. A promising solution is to use yttrium, a metal that forms hydrides more stable than those of lithium, thereby enabling the adsorption of hydrogen isotopes present in liquid lithium. This process can be reversed using thermal desorption spectroscopy (TDS) to characterize and analyze hydrogen release. From a technological perspective, hydrogen thermal desorption could be employed as a method to recover yttrium as a getter material in the trap. If, after thermal treatment, the yttrium releases the trapped hydrogen, replacing the material might no longer be necessary. To assess the viability of this process, the amount of hydrogen released has been measured through thermal treatments at moderately high temperatures. Subsequently, a series of cyclic experiments have been carried out, involving deuterium loading in yttrium, thermal desorption, and repeated loading and desorption. The results obtained have allowed us to determine how much deuterium can be desorbed from yttrium samples and to evaluate the recycling capacity of yttrium.

Tentative

P1-109

Microstructural analysis of oxidised Al-base coated EUROFER97 for corrosion protection

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The reduced activation ferritic/martensitic (RAF/M) Eurofer97 steel is the candidate material for the water-cooled lithium lead breeder blanket (WC-LL BB) section of the European DEMO reactor, however the synergistic effect of the high temperature flowing coolant, tritium embrittlement and ionising radiation can drastically reduce the lifetime of the Eurofer97. Therefore, it is necessary to develop self-passivating coatings that can positively interact with the harsh environment and extend the lifetime of the Eurofer97 during in-service operation. In this study, Eurofer97 was firstly coated with Al-base coatings using either a chemical vapour deposition (CVD at 700 °C and 750 °C) or electroplating process (room temperature); and then oxidised in a 2-step process using the same tempering (980 °C) and normalisation (760 °C) temperature used for Eurofer97. The oxidation temperatures were used to obtain a stable and protective Al-base oxide layer without altering the microstructure of the Eurofer97. Advanced microscopy characterizations revealed the formation of a complex near surface region microstructure formed by an outer α-Al_2O_3 layer over an intermetallic Al-Fe-rich interdiffusion microstructure formed by columnar grains (60 - 90 μm). However, the interdiffusion layer was highly decorated with voids formed during the oxidation process, that might act as high stress intensity points during in-service operations. Therefore, further optimisations on the oxidation process are still needed to improve the resulting oxidised microstructure of Eurofer97.

Tentative

P1-110

Investigating the decomposition of lithium hydride isotopologues for fusion fuel breeder blanket development

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Lithium based materials have been prime candidates for use within breeder blankets as neutron reactions with lithium leads to a fission reaction producing tritium, an isotope of hydrogen necessary for the D-T fusion reaction. In this work we investigate the application of lithium hydride isotopologues, using a atomic scale modelling methods to assess the material's thermal stability. Analysing four of the lithium hydride isotopologues (^{6}LiH, ^{6}LiD, ^{7}LiH and ^{7}LiD) provides the opportunity to determine the differences between the theoretical thermochemical decomposition temperatures. This research identifies a 40 K difference between the theoretical decomposition temperatures of the hydrogen and deuterium containing compositions, with the hydrogen containing isotopologues thermochemically decomposing at 959 K. It is also observed that there is no notable effect on the decomposition temperature caused by the lithium isotope within the composition. With this research we have created a base from which are continuing to develop our understanding of the materials thermal and physical properties, and therefore exploring the viability of lithium hydrides for breeder blankets.

Tentative

P1-111

Building a high temperature liquid lithium flow loop: Lessons for liquid metal engineering

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Liquid lithium as the tritium breeding material for commercial power plants presents significant benefits compared with other designs and is being pursued by a multitude of fusion power developers worldwide, but presents challenges due to its corrosive effects on structural materials.

To substantiate liquid metal blanket designs, the compatibility of materials must be understood under representative conditions. These conditions include, but are not limited to, variables such as temperature, impurity level, and flow rate. Material compatibility studies in liquid-sodium-cooled fission reactors may be used as a benchmark for lithium compatibility investigations in fusion.

Oxford Sigma has been designing and assembling a high temperature, pumped, liquid lithium flow loop for liquid metal corrosion testing of candidate materials for blanket components, called LiFTOFF. The objective of this facility is to be used by the fusion community to generate quality data which is relevant to the engineering design, substantiation, and operation of future facilities and fusion power plants.

Throughout this project, many lessons have been learned in the process of designing and building a liquid lithium loop with the requirement of generating quality corrosion data. Results of materials and assembly testing, undertaken for preliminary assessment of the materials and for facility derisking, have informed the loop design process and will be presented and discussed.

Issues encountered, solutions devised, and lessons learned will be shared for the benefit of the wider fusion community, accelerating future developments in this area for the commercialisation of fusion energy. Facility access and collaborative development opportunities will be discussed.

Tentative

P1-112

Investigation of Hydrogen Isotope Effect at Different Atomic Sites in Nano-Crystalline ZrCo

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ZrCo alloy exhibits a thermodynamic and kinetic difference when storing hydrogen isotopes. It would be a challenge for providing accurate ratio of deuterium and tritium fuel in fusion device,. To investigate the mechanisms underlying the hydrogen isotope effect of ZrCo, this study employed a quadrupole mass spectrometer and thermal desorption spectroscopy (QMS-TDS) combined with peak fitting analysis. The hydrogen isotope separation coefficients for the 3c, 12i, 4c2, 8f1, and 8e sites in Nano-Crystailline ZrCo hydrides with increasing the hydrogen ratio were examined, as demonstrated in figure 1. Experimental results revealed that the hydrogen-deuterium separation coefficient exhibits an exponential correlation with the binding energy. Among these, the 4c2 site displays the lowest hydrogen isotope effect. This would be attributed to the minimal energy difference between hydrogen and deuterium atoms during their transition from the gas phase to the 4c2 site in the alloy. This work provides an insight into the binding energy perspective to understand the mechanisms of the hydrogen isotope effect in ZrCo alloys and holds significant value for the future modification of hydrogen storage materials.

Tentative

P1-113

Studies of high-temperature interaction of Pb16Li eutectic with chemically active gases and water vapor

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The use of lithium alloys with such low-melting metals as lead, tin and gallium in energy systems is a promising direction for the development of modern energy. These lithium-containing alloys have a number of unique properties that make them attractive for use in power plants. In most cases, they are chemically stable and practically do not react with structural materials. However, in case of depressurization there is a risk of atmospheric gases and water vapor entering the systems of the thermonuclear reactor, which results in a high probability of high-temperature interaction of chemically active gases with liquid lithium-containing materials. Thus, studies conducted to identify the mechanisms and describe the processes of this interaction are very important. One of the lithium-containing materials proposed for use in thermonuclear installations is lead-lithium eutectic Pb16Li. The report presents experimental data obtained as a result of high-temperature corrosion tests of a sample of lead-lithium eutectic when it is purged with an inert gas with various percentages of hydrogen, oxygen, and water vapor at a temperature of Pb16Li from 100°C to 800°C. The studies were carried out using a complex method of thermal analysis and mass spectrometry implemented using a TGA/DSC3+ thermogravimetric analysis device connected to a quadrupole mass spectrometer. The experimental data were analyzed, as a result of which thermal effects characterizing the corrosion processes of lead-lithium eutectic during interaction with chemically active gases and water vapor were identified and described.

Tentative

P1-116

Characterizing the mechanical behaviour of different heat treated EUROFER97 materials in non-irradiated and irradiated state

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This study presents the results of mechanical testing on EUROFER97 specimens subjected to different heat treatment conditions, using Charpy impact and tensile tests. A technologically relevant application window is defined by two boundary conditions: the "highest possible ductility" treatment (980°C/30 min + 780°C/2h) and the "highest possible strength" treatment (1100°C/30 min + 700°C/2h). Additionally, it is demonstrated that electro-beam and diffusion-bonded welds subjected to the standard heat treatment (980°C/30 min + 760°C/2h) exhibit mechanical properties within this defined range, aligning well with those of a technological heat-treated EUROFER97.

The mechanical properties of the non-irradiated base material are compared with those of specimens irradiated in the KIT-BOR-60 irradiation program up to 20 dpa at 325°C and 550°C. At 325°C, irradiation causes an upward shift of approximately 200°C in the ductile-to-brittle transition temperature (DBTT) and a reduction in upper shelf energy by 6 J. At 550°C, these shifts are reduced to 20°C and 2 J, respectively. The data obtained from the irradiated base material is further used to predict the behaviour of electron-beam and diffusion-bonded welds under neutron irradiation. The findings are complemented by microstructural analyses, providing a comprehensive understanding of the material's performance.

Tentative

P1-117

Advancements in WEBAM for Fabrication of HCPB Breeding Blanket Components

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The Helium-Cooled Pebble Bed (HCPB) blanket concept is one of the two primary driver blanket candidates for the EU DEMOnstration Fusion Power Plant (DEMO) under the EUROfusion framework. The HCPB breeding blanket is a critical multi-functional structure, however, the fabrication of components of such a structure presents significant challenges due to their complex geometries and space constraints. In recent years, wire electron beam additive manufacturing (WEBAM) has emerged as a promising technology for the fabrication of complex components. WEBAM offers distinct advantages, such as high deposition rates, efficient material utilization, and the ability to produce defect-free components with intricate geometries and minimal material wastage. The present study explores recent advancements in the application of WEBAM for manufacturing HCPB breeding blanket components. The investigation emphasizes on the benefits of WEBAM, including its potential to meet requirements of precision and reliability for fusion reactor components. Additionally, the study examines the effects of material deposition on a previously performed electron beam (EB) full penetration butt welded joint. Mechanical testing and material characterization of fabricated samples are also conducted to validate and ensure integrity and performance.

Tentative

P1-118

Impact of the use of a dual laser system on the microstructure and mechanical properties of PBF-LB/M tungsten

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Tungsten is a desirable material for nuclear fusion due to its high melting point, strength and low sputtering rate. Tungsten's brittle nature greatly hinders its manufacturability but the development of additive manufacturing techniques, such as laser powder-bed fusion (PBF-LB/M) could overcome the challenges posed by traditional manufacturing methods. Previous work has successfully produced PBF-LB/M tungsten samples but crack formation is commonly observed [1]. Crack formation is usually mitigated by preheating the material above the ductile-to-brittle transition temperature (DBTT) of tungsten (200-400 °C) to minimise thermal shock. This work looks to reduce the likelihood of cracking further by post-heating samples with a secondary laser beam, reducing the risk of thermal shock as the material cools. Samples will be characterised using a suite of microscopy techniques and mechanical testing in order to better understand the conditions leading to crack formation in LPBF tungsten components. Samples will be characterised using a suite of microscopy techniques and mechanical testing in order to better understand the conditions leading to crack formation in LPBF tungsten components.

[1] Bey Vrancken, Rishi K. Ganeriwala, and Manyalibo J. Matthews. “Analysis of Laser-Induced Microcracking in Tungsten under Additive Manufacturing Conditions: Experiment and Simulation”. In: Acta Materialia 194 (Aug. 2020), pp. 464-472. ISSN: 13596454. DOI: 10.1016/j.actamat.2020.04.060.

Tentative

P1-119

Phase Evolution and Electrochemical Performance of CoCrFeNiM (M = Ta, Nb, V) High-Entropy Alloys Synthesized by Powder Metallurgy

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High-entropy alloys (HEAs) have attracted extensive interest due to their excellent mechanical, corrosion, and irradiation resistance properties derived from their unique microstructures. However, conventional melting methods often lead to elemental segregation and coarse grains, which limit alloy performance. In this study, CoCrFeNiM (M = Ta, Nb, V) HEAs were prepared via mechanical alloying and spark plasma sintering (SPS), and their microstructures, mechanical behavior, and electrochemical corrosion properties were systematically investigated. After 30 h of ball milling, CoCrFeNiNb and CoCrFeNiV powders exhibited single-phase FCC structures, while CoCrFeNiTa powders showed both FCC and Laves phases. The bulk CoCrFeNiTa alloy sintered at 1100°C consisted of an FCC matrix, Laves phase, and nanoscale Ta_2[Co, Cr, Fe] precipitates, with a relative density of 92.2%, compressive strength of 2177?MPa, and Vickers hardness of 9.37 GPa. CoCrFeNiNb and CoCrFeNiV alloys sintered at 1000 °C were composed of FCC and Laves phases, displaying high compressive strength (2201 and 1835 MPa, respectively) and hardness (5.86 and 6.89 GPa), but brittle fracture characteristics. Electrochemical corrosion tests in 3.5 wt.% NaCl solution at 25 °C revealed that CoCrFeNiNb exhibited the lowest corrosion current density, indicating superior corrosion resistance. XPS analysis showed that passive films enriched in Nb_2O_5, Ta_2O_5, and V_2O_5 formed on the alloy surfaces, effectively improving corrosion resistance. This work provides valuable insights into the phase formation, strengthening mechanisms, and corrosion behavior of HEAs fabricated via powder metallurgy.

Tentative

P1-120

Performance of Plasma-Sprayed Tungsten Coatings Under Plasma Exposure

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Template

Due to its high sputtering resistance, a high melting point, comparatively low tritium retention, and benign activity, tungsten (W) has unique properties as plasma-facing material (PFM) for fusion devices such as ITER and DEMO. However, the plasma conditions can cause damage to the material and components, resulting in reduced device lifetime. Repair of the plasma-facing components (PFC) is resource-intensive and time-consuming. Therefore, developing efficient in-situ repair methods is desirable. Low-pressure plasma spraying (LPPS) allows a fast and modest repair of damaged surfaces and/or restoration of PFM to adequate thickness. This is of interest for fusion devices operating with e.g., Eurofer steel as main structural material or those with Carbon Fiber Composite (CFC) PFCs.

This study used a statistical approach to reduce the number of coating experiments using tungsten powder and identify factors affecting the coating quality on various substrates such as Eurofer steel, tungsten, or CFC. The coatings were evaluated for various quality parameters including porosity, defects, surface roughness, thickness, and deposition efficiency. Fusion-relevant tests were conducted on 500 µm coatings with porosities below 1 % using the linear plasma device PSI-2. A variation of the plasma gases neon (Ne) and deuterium (D) provided data on sputter yield and deuterium retention. A comparative evaluation of the performance of the plasma-sprayed coatings in relation to bulk tungsten will be presented.

P1-121

Multi-Length Scale Characterisation of the role of Ti(CON) precipitates on radiation damage in V-based, reduced activation, metallic systems

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Structural materials in fusion reactors experience a uniquely challenging combination of operating conditions. To accelerate material selection and deployment, a mechanistic understanding of radiation damage and resistance is crucial to optimise existing candidate materials, such as V-4Cr-4Ti, and aid in the design of novel fusion-focused metallic systems, such as reduced-activation, compositionally complex alloys (CCAs).

Microstructural development plays a crucial role in engineering radiation resistance in metallic systems. Precipitation of secondary phases can influence mechanical, physical and radiation-resistance properties. Ti(CON) impurity precipitates have been specifically reported as having a complex role in the radiation behaviour of V-based alloys.

This work presents a multi-length scale design and characterisation approach, including CALPHAD, XRF, (GI)XRD, SEM/EDX, APT and (S)TEM, to further understanding of the development and radiation response of these precipitates.

The role of thermomechanical processing on Ti(CON) in V-4Cr-4Ti has been further elucidated and following heavy-ion implantation to 15 dpa, the radiation response of Ti(CON) in a novel, reduced-activation, V-based CCA system has been characterised.

In these CCAs it is observed that small additions of Fe & Cr drive different phase and radiation behaviours in TiCON. In TaVTiW5Fe5 microscale Ti(CON) precipitates preferentially forms at grain boundaries and undergo minimal radiation damage with their boundaries acting as Ti sinks. In TaVTiW5Cr5 , APT reveals nanoscale Ti(CON) grains evolving in size and morphology as a function of damage.

These results highlight the complexity of the role of Ti(CON) and suggest potential avenues for advanced microstructural design to optimise radiation response in V-based alloys.

Tentative

P1-122

The Irradiation Effect Research of the Spallation Target

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The target materials and the structural materials used in China Spallation Neutron Sources(CSNS) endure the high energy neutrons and protons bombardment, alternating stress fields, and corrosion. Meanwhile, abundant helium atoms are generated by high-energy nuclear reaction. Hence, the development of advanced radiation tolerance and corrosion resistant materials is important. We find the combination of amorphous structure and hetero-interfaces present excellent swelling resistance under irradiation. In order to improve the corrosion resistant of tungsten target, a Zirconium alloy coated tungsten (W) targets were prepared by hot isostatic pressure (HIP) diffusion welding. We use a sandwich structure to avoid the generation of the brittle phase. We also will introduce the progress of the post irradiation examination for CSNS target.

Tentative

P1-123

New concrete mix design for Nuclear Application

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The main components of nuclear power plants are realized by concrete structures. Indeed, primary containment building, shielding wall, secondary and internal containment, and cooling towers are realized by concrete members. The shielding wall is the only structure that should receive radiation from the plant and is designed very close to the reactor vessel to absorb radiation, act as a load-bearing structure that supports the reactor vessel and provide structural support for the nuclear steam system and various equipment in nuclear power plants. On the other hand, in fusion power plants, concrete shielding helps to prevent radiation releases and to mitigate radiation (gamma, neutron and other radiation). Although concrete is a widespread structural material throughout the world, the inclusion of other components can alter its original mechanical behavior. In this context, the authors have developed an intense research work to create new concrete mix design solutions against neutron emissions. For this reason, concrete elements mixed with steel fibers and plastic materials were designed to resist the penetrability of neutron emission. Specimens with different volumetric percentages of components were tested experimentally to characterize the mechanical properties and ability to shielding neutron particles. In more detail, neutron measurements were carried out at the Frascati Neutron Generator (FNG), operating at the ENEA Frascati Research Center and mechanical performance was analyzed at Strength Laboratory of the University of Salerno. The experimental results were analyzed and compared with those obtained using sample of normal concrete.

Tentative

P1-124

Enhancing Hydrogen Isotope Permeation Resistance: The Significance of Plasma - Induced - Oxide - Interlayer

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The high permeability of tritium within structural materials presents substantial challenges to the operation of fusion reactors. In response to this pressing problem, this study breaks new ground by enhancing the overall performance of tritium permeation barrier coatings. For the first time, an intermediate layer is introduced to modify the substrate's wettability. Specifically, a plasma oxidation treatment was carried out on the surface of T91 steel, leading to the formation of an (Fe, Cr)O oxide intermediate layer. Subsequently, an $\alpha - Al_2O_3$ coating was applied via the brushing method. The experimental findings show that the sample featuring the intermediate layer displays excellent hydrogen permeation resistance. Additionally, its corrosion resistance is enhanced after plasma oxidation, and its thermal shock performance is improved.

Tentative

P1-125

Development of Fabrication Technologies for the Breeder Unit Cooling Plate in the HCCP Breeding Blanket

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The breeder unit, a core component of the breeding blanket, is responsible for neutron multiplication and tritium breeding. During DT operation, nuclear heat is generated due to interactions between neutrons and functional materials such as neutron multiplier material and tritium breeding materials. The cooling plate of the breeder unit is essential for removing the nuclear heat. This study focuses on the development of fabrication technology for the breeder unit cooling plate. The cooling plate contains multiple rectangular cooling channels, each measuring 4.5 mm × 2.6 mm. Previously, these channels were fabricated using Electrical Discharge Machining (EDM), which led to significant time and cost. An alternative approach utilizing diffusion bonding with a hot press was proposed to overcome this limitation. A comprehensive test matrix was established and implemented to determine the optimal bonding and post-heat treatment conditions. In addition to cooling channel fabrication, bending of the cooling plate is a critical aspect that requires further technological development. Finite element method analyses were carried out to investigate bending behavior and identify key parameters influencing the bending process. Based on the findings, a dedicated bending tool was developed, and various test conditions were evaluated to determine the optimal bending conditions. This study presents the fabrication process of the large-scale mock-up and the results of performance validation tests. The findings are expected to contribute to advancing the fabrication technology of the breeder unit cooling plate.

Tentative

P1-127

Development of irradiation capsule design and plan for neutron irradiation of chromium beryllide samples in the WWR-K reactor

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Metallic beryllium has long been considered as a candidate neutron multiplier material in the solid-state breeder blanket of a fusion reactor. However, beryllium intermetallic compounds (beryllides) are currently preferred due to lower swelling and higher stability at high temperatures. One of the promising and less-studied beryllium compounds is chromium beryllide ($CrBe_{12}$), which is considered a candidate material for the neutron multiplier in the modular blanket of the future DEMO fusion reactor. Experimental data on the behavior of $CrBe_{12}$ in a neutron field are insufficient to conclude that it is suitable for a fusion reactor. In this regard, work was started at the WWR-K reactor to study the corrosion stability of $CrBe_{12}$, manufactured at the Ulba Metallurgical Plant, under reactor irradiation conditions. This paper presents the developed design of an irradiation capsule that forms conditions close to operational conditions in fusion reactors. The results of comprehensive neutron and thermophysical calculations are presented, justifying the irradiation conditions and their predicted values in the WWR-K reactor. The reactor irradiation plan and methodology for the corrosion resistance study of $CrBe_{12}$ samples are described.

The research has been funded by the Science Committee of the Ministry of Science and Higher Education of the Republic of Kazakhstan with Program BR24993119.

Tentative

P1-128

Scale-able and Tune-able Compact Accelerator Neutron Source for Materials Testing under Fusion Environments

Joven J.H Lim[1], Tony Turner[1], Prashanth Srinivasan[1], Louis Butt[2], Ben Phoenix[2], Duc Nguyen[1], Mark Gilbert[1], Yu-Lung Chiu[2], Lee Packer[1], Martin Freer[2], uBB project contributors[3]

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A scalable and tunable neutron source utilizing Neutron Field Engineering Technology (NFET) has been developed at the UK Atomic Energy Authority (UKAEA) in collaboration with the University of Birmingham to advance the qualification of materials performance under neutron irradiation. The Tunable Micro-Breeder Neutron Emission Device (TuNED) provides a fusion-relevant neutron field for strategic testing, enabling studies on the synergistic effects of irradiation with temperature, mechanical stress, magnetic fields, corrosion, and tritium-breeding environments-critical factors influencing material degradation in fusion reactors.

The ability to precisely tailor neutron spectra and flux levels makes TuNED a versatile platform for controlled material irradiation experiments, offering insights into displacement damage, transmutation effects, and tritium interactions. These studies are essential for understanding long-term structural integrity, thermal stability, and mechanical resilience of fusion materials. Additionally, TuNED enables targeted testing of tritium breeding materials, supporting the development of breeder blankets and tritium accountancy methods.

While complementary to volumetric neutron source/s currently being considered in fusion roadmap, TuNED provides a cost-effective, high-throughput alternative for iterative material qualification and early-stage screening. Data from these experiments will enhance simulation and modelling capabilities, improving predictive assessments of material lifetimes and failure mechanisms in fusion environments. By addressing critical gaps in irradiation testing, TuNED plays a vital role in accelerating the development of durable materials for next-generation fusion reactors

Template

P1-129

Advancements in Multi-Ion Beam Simulations for Fusion Material Testing at Elevated Temperatures

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Structural materials in fusion environments face unique challenges, such as extreme temperatures, transmutation effects, and neutron damage levels not found in fission reactors. Developing and rapidly screening novel materials to endure these harsh conditions is crucial. The Michigan Ion Beam Laboratory (MIBL) tackles this by using innovative multi-ion beam irradiation techniques. By accelerating various ion species, we rapidly simulate damage that reflects fusion reactor conditions. We will present two recent advances for simulating fusion environments at MIBL. First, we introduce a new high-temperature stage (>1200°C) with low thermal mass and PID control, allowing for rapid heating and cooling. This system facilitates steady-state and pulsed triple-ion irradiations. Second, we present simulations and commissioning tests for simultaneous irradiation with helium (He), hydrogen (H), and heavy ions vanadium (V), tungsten (W), rhenium (Re), and osmium (Os). This setup enables a wide range of simulated transmutation and displacement damage combinations in high-temperature refractory alloys. Our estimates for 12 MeV W-W^{+3} self-ion irradiations show that we can achieve ion fluences of 10^{16} W/cm^2, 10^{15} Re/cm^2, 10^{10} Os/cm^2, and 10^{7} He/cm^2 in 30 hours, yielding 100 dpa and appm/dpa rates of ~100 for Re and ~10 for He at an implantation depth of 300 nm. When combined with a system of foil degraders and moving shutters, this setup enables continuous gradients or discrete damage regions on single or multiple samples, facilitating accelerated screening. We conclude with preliminary results on W-rich alloy microstructural evolution under DEMO-like conditions, using dynamic Re implantation to emulate synergistic damage-transmutation effects.

Tentative

P1-130

Introducing the MIT PSFC LMNT Facility: Implementing high throughput, near prototypic irradiation of bulk fusion materials with 10-30 MeV protons

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Fusion power plants will expose materials to energetic neutron fluxes of up to 1e14 to 1e15 per cm^2 per s, equivalent to ~15 displacements per atom (dpa) per year and inducing (n,α) transmutation reactions resulting in bulk helium up to 150 atomic parts per million per year. The uncertainties around how such changes will impact the structural, thermal, and electrical property evolution of fusion materials must be minimized to provide confidence in the design and operation of first-of-a-kind systems being proposed for the 2030s. To overcome the technical limitations of traditional irradiation techniques on the timescale required, the direct irradiation of bulk material specimens by an intense beam of 10-30 MeV protons has been recently explored for near-prototypic irradiation of fusion materials. Advantages include high damage rates (~few dpa/day), fusion-relevant helium transmutation (up to ~10 appm/dpa), and the use of macroscopic (>200 um) specimens that enable extraction of engineering properties. This talk will provide an overview of a new purpose-built facility for fusion materials irradiation - the Laboratory for Materials in Nuclear Technologies (LMNT) - that is now under construction at the MIT Plasma Science and Fusion Center. LMNT will employ a 10-30 MeV, 800 uA, dual-extraction proton cyclotron with external beam lines connecting to four independent and modular endstations. The endstations contain beam target specimens capable of high power handling and can be configured to impose a wide range of environmental conditions on materials during irradiation including contact with high-temperature liquid coolants and even exposure to fusion-grade plasmas.

Tentative

P1-131

ECR Ion Source Upgrade for Fe Ion Beam Supply at KAHIF for Nuclear Materials Irradiation Tests

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An upgrade of the Korea Atomic Energy Research Institute Heavy Ion Irradiation Facility (KAHIF) is presented to enable the operation of an Fe ion beam using an Electron Cyclotron Resonance (ECR) ion source. For nuclear fusion materials irradiation tests, a beam that not only matches the target's elemental composition but also induces high displacements per atom (dpa) is required. While KAHIF previously operated with gaseous ion beams such as He, Ar, and Xe, the increasing demand for Fe-based structural materials has driven the need for metal ion beams, necessitating the conversion of a metal element into a plasma state. To achieve this, a Metal Ions from Volatile Compounds (MIVOC) system was introduced, employing ferrocene as a precursor for generating Fe gas. Preliminary tests were conducted to determine the optimal operating conditions for Fe gas generation. Subsequently, the ECR ion source was used to produce an Fe plasma, and mass spectroscopy was performed using a bending magnet to confirm the generation of Fe ions. By optimizing the extraction voltage and einzel lens voltage of the ECR ion source, the optimum ion species, Fe $^{13+}$, was selected for beam supply. The upgraded system successfully delivered an Fe 13+ beam at a current of 1 μA, corresponding to a fluence of 4.5×10^14 ions/cm^2. Detailed results of the ion beam supply experiments, the SRIM simulation outcomes, and the validation experiments on pure Mo specimens will be presented.

Tentative

P1-132

The experimental sodium cooled fast reactor Joyo as powerful neutron irradiation facility for developing fusion reactor materials

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The experimental fast reactor Joyo of the Japan Atomic Energy Agency (JAEA) has a large potential for not only the sodium-cooled fast reactor development but also fusion reactor material development. Joyo will re-start operation from the middle of 2026 Japan Fiscal Year. Its thermal output is 100 MWt in order to conduct irradiation tests up to high neutron irradiation doses. The maximum dose rate is 7 dpa/cycle (1 cycle = 60EFPD). This high flux and high energy neutron irradiation capability and large irradiation volume are advantageous especially for fusion reactor material irradiation up to several ten dpa. The temperature of irradiation specimens can be set around from 400 to 1,000°C according to required conditions. In addition, Joyo has on-line irradiation techniques developed for irradiation tests under the highly precisely controlled temperature within the range of ±4°C. Furthermore, post irradiation examination facilities are located adjacent to Joyo. Various post irradiation examinations can be performed there immediately after the irradiation in Joyo. More than 70,000 hours of operations were successfully carried out, and over 40,000 external users' specimens, have been irradiated in Joyo until now. We have abundant experience in irradiating fusion reactor materials such as ferritic steel, W, V, and alloys using them, as well as SiC. Joyo which has the capability of high fast neutron and high-precision irradiation, and its related post irradiation examination facilities will be opened to external users not only in Japan but also overseas after the re-start.

Tentative

P1-133

A unified approach to active sample handling and plasma exposure at the forthcoming nuclear linear plasma device JULE-PSI

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Fuel retention and wall material erosion are two critical operational lifetime limiting factors for fusion reactors. While linear plasma devices such as PISCES, PSI, MAGNUM and TPE have measured retention and erosion for candidate materials, the addition of neutron damage is known to further augment the fuel retention and wall-erosion. TPE has measured and established increased retention in low activation neutron irradiated samples, however to quantify the retention and qualify materials for future fusion power plants, a dedicated linear plasma device for measurement of the lifetime retention on highly irradiated samples is essential.

JULE-PSI is a forthcoming linear plasma device which is to be installed in a hot cell. The hot cell enclosure allows highly radioactive samples (~ TBq range) to be handled and exposed to fusion relevant first-wall plasma scenarios. Such experiments and exposures require active sample handling and transport, which forms a major challenge to the experiment. Furthermore, JULE-PSI is planned with in-situ laser-based diagnostics which are crucial to quantify, in-vacuo, the post exposure retention. Such a device requires a unified approach to active sample-handling, plasma-exposure and post-exposure-retention measurement.

In this contribution, a sample handling and transport concept for the linear plasma device will be detailed alongside a status update on the test-stand plasma commissioning of JULE-PSI. Furthermore, recent results from the in-situ laser-based diagnostics to be incorporated into JULE-PSI will be presented.

Tentative

P1-134

Operation and upgrade progress of KAHIF for fusion material irradiation testing

Sangbeen Lee[1], Seunghyun Lee[1], Dae-Sik Chang[1], Kihyun Lee[1], Dong Won Lee[1]

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KAHIF (KAERI Heavy Ion Irradiation Facility) is RF linear accelerator-based heavy ion beam irradiation test facility in Korea. The facility accelerates a wide spectrum of ions, ranging from He to heavy ions such as Fe, with an acceleration energy of 0.178 MeV per nucleon. The beam particle flux provided by KAHIF are 1.68 x 10^{13}[#/cm2/s] for He ion beam and 4.02 x 10^{12} [#/cm2/s] for Ar ion beam. Recently, Fe ion beams with 0.5 ~ 1 uA of the beam current have been extracted successfully. KAHIF has been serviced the irradiation tests with He and Ar ions for the fusion material research groups in Korea. Now, Fe ion beam is ready to irradiate the testing samples. With these ion beam condition, the facility enables research on evaluate the effect of neutron irradiation on material properties especially, fusion reactor structural materials and nuclear fuel materials.

In order to provide more various test conditions, the facility is planning to upgrade the beam line. The new target chamber is designed more bigger size to provide the increased temperature (up to 900℃) and the modulation of incident beam angles. The various port of the chamber will be utilized for monitoring the status of the irradiating sample. The new chamber for beam diagnostics is designed for providing the beam profile information. The detailed of the upgrades of KAHIF is presented in this paper.

Tentative

P1-135

The MIT PSFC LMNT facility: near-term research plans to rapidly advance fusion materials science and support fusion commercialization

Sara Elizabeth Ferry[1], Lou Wainwright[1], Matthew Baldwin[2], Sophie Blondel[3], Lianyi Chen[4], Michael Short[5], David Sprouster[6], Kevin Woller[1], Zachary Seth Hartwig[1]

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The U.S. is lacking in near-term experimental facilities capable of assessing radiation damage in fusion-prototypic conditions, despite the urgent need for this data due to the rapid progress and aggressive timelines of the global commercial fusion sector. A new multi-institutional project led by the MIT PSFC, funded as part of the U.S.-DOE FIRE collaborative program, aims to address this gap by leveraging a new 10-30 MeV proton beam facility, the MIT PSFC Laboratory for Materials in Nuclear Technologies. The LMNT is under construction at the PSFC with planned operations in 2026. We will present the facility's near-term research plans, with an emphasis on the technical goals of the FIRE collaborative, which includes the generation of data needed to improve models of radiation damage in fusion materials, effects of fusion-relevant transmutation in RAFM and copper-based alloys, radiation damage in tungsten alloys, materials discovery, and simultaneous irradiation/corrosion effects in liquid breeder materials. We will also discuss the broader goals of this new cyclotron facility, including: 1) how the facility supports the recent U.S. fusion material community roadmap; 2) the complementarity of the facility to planned and proposed fusion prototypic neutron source facilities; and 3) the role of the cyclotron facility as a powerful research platform that the MIT PSFC team plans to make accessible to both private fusion company partners and to publicly-funded nuclear materials researchers.

Tentative

P1-136

The MIT PSFC LMNT facility: Target design, testing methods, and experimental techniques for 10-30 MeV proton irradiation

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Existing methods for assessing radiation damage on the bulk properties of fusion materials have limitations. Fission reactor irradiation is limited by dose rates of a few dpa/year and inability to emulate fusion-relevant solid and gaseous transmutation while high costs, multi-year cycles, and a small and aging fleet of reactors prevent further constraints. Low energy proton and self-ion beam irradiation damage is limited to the near-surface (~0.1 to ~10 μm), restricting analysis to microstructure imaging and techniques like nano- and micro-hardness testing to infer structural properties, while multiple accelerators are required to emulate fusion-induced composition changes. LMNT - a new fusion materials irradiation facility at the MIT Plasma Science and Fusion Center - will employ an alternative technique using 10-30 MeV proton beams to combine the high damage rates of ion beams with the uniformly irradiated thick samples of neutron irradiation. This talk will present target designs for irradiating sets of over 10 samples simultaneously, including thermal modeling and strategies for controlling sample temperature during irradiation, as well as the design and simulation of samples optimized for tensile testing with minimum irradiated mass and activation. Simulations of the solid and gaseous transmutation show a wide range of achievable transmutation-to-dpa ratios through tuning of the proton energy, allowing emulation of fusion-relevant conditions for near-prototypic conditions and transmutation-free damage creation. Additionally, dose rates can be precisely controlled to study the effect of damage rate on defect evolution. These capabilities are expected to provide the fusion community with exciting new avenues for materials research.

Tentative

P1-137

Facility for the Evaluation of Thermal and Elastic Material Properties During Plasma or Ion Irradiation

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Of critical interest to the fusion materials community are the thermal and mechanical properties of reactor components. A facility for the investigation of both radiation- and plasma-materials interaction is demonstrated with tungsten. The facility couples a 1.7 MV tandem ion-accelerator with a linear helicon plasma chamber and a device for the production of on-sample transient gratings for in-situ thermal diffusivity and surface acoustic wave analysis. To study the effect of plasma-material interaction and irradiation damage, initial data using tungsten are presented, exhibiting fuzz growth under helium plasma exposure and changes in thermal properties under 10.2 MeV self-ion irradiation up to 7.9 dpa.

Tentative

P1-138

Overview of a KAERI Heavy-ion Irradiation Facility (KAHIF) for nuclear fusion/fission material research

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In the fields of nuclear fission and fusion, a neutron irradiation damage assessment is essential. Traditionally, neutron irradiation research has been conducted at large-scale facilities, such as research reactors or accelerator-based high-energy neutron sources. To provide a more cost-effective and time-efficient alternative, ion irradiation testing has been adopted.

Korea has addressed these needs by establishing the KAERI Heavy-Ion Irradiation Facility (KAHIF). Since its inception in 2019, KAHIF has completed over 450 hours of beam irradiation and undergone significant upgrades to enhance its capabilities. These upgrades have expanded the facility's capability to generate iron (Fe) ion beams, along with a wider range of gas ions, including helium (He), argon (Ar), and xenon (Xe). Additionally, the facility includes a high-temperature test chamber, capable of reaching temperatures up to 450°C. These advancements have enabled a broad range of tests on various nuclear fission and fusion materials, addressing domestic research needs.

KAHIF supports research on Reduced Activation Ferritic-Martensitic (RAFM) steel, known as Advanced Reduced Activation Alloy (ARAA), which has been developed by KAERI for use in fusion reactors. In addition, our research focuses on cladding materials for nuclear fuel, neutron absorbers for spent fuel storage, and novel nuclear fuel materials for small modular reactors (SMRs). This presentation will highlight the progress and current status of experiments in these studies.

Tentative

P1-139

Application of EPICS in KAHIF and KoHLT-EB System for Fusion Material Irradiation and Plasma Facing Component Testing

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Heavy ion irradiation facility and high heat load test facility using electron beams as device for developing and testing fusion materials and plasma facing components (PFC) have been operated in Korea. These facilities are referred to as the KAERI Heavy Ion Irradiation Facility (KAHIF) and Korea Heat Load Test Facility using Electron Beam (KoHLT-EB). The KAHIF was originally the Tokai Radioactive Ion Accelerator Complex (TRIAC) and, transferred to KAERI. Then, it has been used as ion irradiation facility for fusion material with He, Ar, and Fe ions. The control system was upgraded using the Experimental Physics and Industrial Control System (EPICS) to reconfigure the facility's operating environment and add a system to collect data for the creation of irradiation transcripts. The KoHLT-EB used a proven solution based on Programmable Logic Controller (PLC) from Siemens to control and monitor the electron beam and implement safety interlocks, but also used EPICS to monitor the vacuum system and cooling system, configured a Field Programmable Gate Array (FPGA) based data acquisition system to speed up data processing while using software filtering for noise processing of analog signals such as temperature sensors. More detail on how EPICS was applied to these facilities will be discussed in the presentation.

Tentative

P1-140

Evaluation of Deuterium Retention Properties on Pure Tungsten with Various Cold-Working Ratio

Shuka Tsuda[1], Masayuki Tokitani[2], Shun Shimabukuro[3], Mitsutaka Miyamoto[4], Yukinori Hamaji[2], Hiroyuki Noto[2], Suguru Masuzaki[2], Hiroshi Idei[3]

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Tentative

Tungsten is the most suitable candidate as a plasma facing material for fusion reactors, in which its intrinsic brittle feature can be controlled by subsequent work hardening processes, e.g., cold-working, etc. The cold-working process modifies the microstructure of tungsten matrix such as grain refinement and the introduction of dislocations. Those modified microstructures would lead to changes in retention property of the hydrogen isotopes in tungsten. This study focused on evaluation of deuterium (D) retention properties on pure tungsten with various cold-working ratio.

Mechanically polished tungsten specimens with two cold-working types, high-(H-W) and low-(L-W) cold-working ratio were prepared, with size of 10×9×0.23 mm^3 and 10×10×0.82 mm^3, respectively. Grain size and shape in both top and cross-sectional surface were observed by scanning electron microscope (SEM). In addition, potentially included residual-stress during manufacturing processes in both surfaces were visualized by a Kernel Average Misorientation (KAM) map function in Electron Backscatter Diffraction (EBSD) technique. The specimens were irradiated with 1.5 keV-D^+ at room temperature up to a dose of 3×10^21 D/m^2. Then, thermal desorption spectroscopy (TDS) experiments to D_2 desorption property were conducted.

High amount of deuterium retention in H-W was confirmed compared with L-W. Especially, very intense desorption peaks in low (near room temperature) and high (450-560K) temperature region were observed in H-W. SEM and KAM map results showed that H-W grains have a higher aspect ratio, and include higher residual-stress, compared with L-W. Those microstructural features in H-W including high residual-stress may act as an effective trapping site of hydrogen isotopes.

P1-141

Modeling the Effect of He/dpa Ratio on Prismatic Loop Evolution in Irradiated Ferritic-Martensitic Alloys

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Ferritic-martensitic (FM) steels are considered the primary candidate for structural materials in fusion reactors due to their resistance to void swelling and corrosion, along with their strong mechanical properties. While significant research has been conducted on their use in fission environments, fusion environments present distinct challenges. A key concern is the impact of helium produced by 14 MeV neutrons, which can influence microstructural properties, such as the size and density of a/2 and a prismatic dislocation loops. Here we present results using molecular dynamics simulations of the behavior of these loops in the presence of helium to build an atomistic database to inform mesoscale predictions of loop evolution in irradiated Fe-Cr based alloys. Xolotl, a continuum-scale cluster dynamics code, solves coupled partial differential equations to track defect microstructure evolution under irradiation conditions. Currently, Xolotl is capable of predicting the evolution of species like a/2 and a dislocation loops, interstitials, and voids. This work seeks to integrate the effect of defect interactions with helium into Xolotl framework to study the effect of helium on microstructure evolution. The results will be benchmarked against available experimental data.

Tentative

P1-142

Machine-learning force fields for hydrogen and vacancy complex in tungsten

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Tungsten (W) is used for plasma-facing components in nuclear fusion reactors, such as ITER and DEMO, due to its low tritium retention as well as its high melting point and low sputtering yields by hydrogen ions. However, experimental findings have shown that ambient deuterium (D) atoms in W can stabilize irradiation-induced defects thereby enhancing D retention. These results suggest that hydrogen-vacancy complexes may play a role in the stabilization of irradiation-induced defects. In this study, to understand hydrogen-vacancy (H-V) interaction microscopically, we developed a machine-learning force field (MLFF) for the H-V complexes in W based on density functional theory (DFT) calculations. In this MLFF, structural data (total energy, forces, and the stress tensor) were obtained from the DFT calculations using VASP, and local atomic environments were characterized by the Smooth Overlap of Atomic Positions (SOAP) descriptor. Using Bayesian linear regression, unknown structural properties were predicted with quantified uncertainties. Then, the on-the-fly machine-learning approach was employed to efficiently sample training data to improve the MLFF. Using the MLFF, we calculated the binding energy of hydrogen atoms to a monovacancy and compared it with previous calculations of the DFT and model potentials. For the vacancies containing one to eleven hydrogen atoms, the difference between DFT values and the predicted values by the present MLFF was 0.47 eV at most, with the root mean squared error of 0.24 eV. The agreement is significantly improved in comparison with the model potentials, allowing MLFF-based MD simulations with the accuracy comparable to DFT.

Invited

P1-143

SIMS and TEM characterization of Eurofer97 irradiated with helium ions using energy degrader and dose gradient

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This study investigates the simulation of neutron damage and transmutation effects in Eurofer97. Since there is currently no available neutron source capable of replicating fusion conditions, ion irradiation is utilized to emulate the damage and transmutation products, with helium being a primary transmutation product through the (n,α) reaction. At the Ruđer Bošković Institute, the DiFU dual-beam facility [1] has been developed for irradiation of fusion-related materials' samples using heavy ions irradiation for damage production and light ions (p and He) implantation for emulation of transmutation products.

To achieve a more homogeneous distribution of implanted helium ions, an oscillating energy degrader has been developed. This system uses a thin aluminum foil that oscillates in front of the ion beam, altering the effective penetration depth and spreading the final range in the target, ensuring helium is implanted homogeneously in depth. The depth profile of helium ions in irradiated Eurofer97 samples is confirmed via SIMS and TEM, showing a uniform, over a micrometer deep distribution, as opposed to the profile retrieved from samples irradiated without the degrader.

Additionally, a dose gradient apparatus enables irradiation with varying doses across a single sample by scanning the beam in a non-uniform pattern, thereby creating a dose gradient across the scanned area, confirmed by SIMS and TEM.

These techniques provide a comprehensive approach to simulating fusion reactor conditions and evaluating material responses to fusion-related radiation.

[1] Tadić T, Dunatov T, Fazinić S, et.al.. Development of the Dual-Beam Ion Irradiation Facility for Fusion Materials (DiFU). Materials. 2023; 16(3):1144.

Tentative

P1-144

The effect of microstructure and defect concentration of tungsten on surface blistering and deuterium retention

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The interaction between tungsten material and hydrogen isotope plasma has been extensively studied for decades. Blistering and deuterium retention in tungsten are sensitive to its microstructure and defect concentration. Two hot-rolled polycrystalline tungsten materials produced by different manufacturers (ATW and PLW), two columnar crystal tungsten materials with different grain diameters (CVDW-L and CVDW-S) produced by chemical vapor deposition (CVD), a cold-rolled tungsten foil material (WF) and a recrystallized tungsten material (WR) were used in this study. All samples except WR were divided into two groups: one annealed at 1273 K for one hour to reduce defect concentration and release residual stress, while the other remained as received for reference. These samples were exposed to deuterium plasma with a fluence of 4×10^24 D・m^(-2). Significant blistering was observed on the as-received ATW, PLW and WF, moderate blistering on WR, while only a few blisters were observed on as-received CVDW-S and CVDW-L. The annealed samples had fewer blisters than as-received ones, with almost no blisters observed on annealed CVDW-S and CVDW-L. Deuterium retention behaviors were analyzed by thermal desorption spectroscopy (TDS) and the results suggest that samples with lower defect concentration have lower deuterium retention. The mechanism of interaction between deuterium and defects in tungsten was discussed, explaining the effects of microstructure and defect concentrations on surface blistering and deuterium retention.

In Review

P1-145

Diffusion of helium in tungsten-based refractory alloys by molecular dynamics

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Understanding helium (He) behavior in structural materials is critical for the development of fusion reactors, as He accumulation can lead to material degradation and failure. High-entropy alloys (HEAs) have emerged as promising candidates for fusion applications due to their unique properties, such as enhanced radiation resistance. We present a newly developed interatomic potential to model the interaction between He atoms and the HEA WTaVMoNb. The potential parameters for the interactions of Mo, Nb, Ta, V, and W with He are determined by fitting results from first principles calculations. Key fitting targets include the formation energies of a single He atom at interstitial sites, the binding energies of He-vacancy in pure metals, and the migration energy of a single He atom in metals. Then we calculate the binding energies of He clusters and He clusters with a single vacancy in five metals (Mo, Nb, Ta, V, and W) and seven alloys (WMo, WTa, WV, WTaMo, WTaV, WTaVMo, and WTaVMoNb). Additionally, we study the average migration energies of a He atom from pure W to the W-based HEA WTaVMoNb. Finally, we simulate the nucleation and diffusion of He bubbles using the developed potential. Our results indicate that only a small percentage of He atoms form bubbles in V-containing materials.

Tentative

P1-146

Investigation hydrogen-helium synergistic damage mechanisms using multibeam ion irradiation

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Transmutation products, Hydrogen(H) and helium(He) synergize with displacement damage to severe degradation of structural materials performance and threaten the safety of the fusion reactor. Since there is a lack of available fusion neutron sources, multi-beam irradiation was used to investigate the role of H and He in the evolution of irradiation defects evolution. It was observed that the influence of H on the synergistic effects is minimal. When H coexists with displacement defects alone, it does not show obvious effects. Only when He is present simultaneously can H slightly promote cavity nucleation in triple-beam irradiation. Also, the evolution of irradiation defects varies with the hydrogen-helium ratio. A rise in the H/He ratio under triple-beam conditions enhances defect nucleation and increases the number density of dislocation loops or cavities, leading to reduced swelling and irradiation hardening. For the same dose rate variation, the presence of H, He has a more significant effect on defect evolution than single beam irradiation, as evidenced by a faster decrease in the average defect size and a significant increase in the number density. Temperature and dose are important parameters affecting the evolution of defects. The irradiation swelling is most significant at 450℃, and there is a trend of saturation of irradiation cavity number density with dose, and cavity growth is the main factor leading to increased swelling.

Tentative

P1-147

The role of H and He in synergistic effects on cavity evolution in pure Ni at multiple temperatures

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In fusion reactors, the D-T fusion reaction produces 14 MeV neutrons that induce displacement defects and transmutation H/He gas atoms by (n,p) and (n,α) reactions. Early low-dose (0.02 dpa) fusion neutron experiments revealed H-He-displacement defect synergism in pure Ni, accelerating material degradation. Due to limited fusion neutron facilities, multiple ion beam (MIBI) irradiation is essential for studying radiation effects. To explore the impact of H and He in synergistic effects, systematic experiments were conducted at Xiamen Multiple Ion Beam in situ Analysis TEM Facility. Pure nickel was irradiated to ~12 dpa with different gas concentrations at temperatures ranging from 300 to 600 °C. The multi-ion beams consist of 400 keV Ni^+, 40 keV H_2^+ & He^+. The H-He gas-dose ratios were 0-0; 0-10; 40-10; 50-0 appm/dpa. Ion-irradiation induced cavities were characterized using transmission electron microscopy (TEM). The results indicated that hydrogen's role was not significant at lower temperatures; however, at 600 ℃, Ni+H dual beam irradiation with 50 appm/dpa H produced smaller and denser cavities with a 26.8% decrease in diameter and a 114.38% increase in density compared to Ni ion-irradiation. This suggests that H could promote the nucleation and dispersion of cavities at higher temperatures. Further comparison between Ni+He and Ni+He+H ion irradiation revealed that the presence of helium slightly enhanced the influence of hydrogen on cavity evolution, while the impact of helium was found to be more pronounced. Overall, this work provides a comprehensive understanding of fusion neutron irradiation damage on blanket structural materials.

Tentative

P1-148

Modeling experimental dependencies of deuterium penetration through SS316IG stainless steel under reactor irradiation conditions

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SS316IG (ITER Grade) stainless steel is one of the main materials used in the design of vacuum chambers, heat exchangers, breeder blankets and divertor components of fusion devices, such as ITER and DEMO.

Under high temperature and intense neutron irradiation conditions, SS316IG steel can become permeable to deuterium and tritium, which can lead to their leaks from the reactor, creating risks of radioactive contamination, changes in the mechanical properties of steel due to the softening effect, hydrogen brittleness and radiation modification of the structure.

Thus, the development of predictive models for the transfer of hydrogen isotopes in structural steels of nuclear reactors is an urgent task that allows assessing the safe and efficient operation of fusion reactors.

In the presented work, the analysis of the results of reactor experiments on the permeability of deuterium in SS316IG steel performed at temperatures of steel samples of 350-550 °C at an deuterium input pressure of ~500 Pa and a flux of thermal neutrons of ~1014 n/cm^2・s is carried out. Based on the diffusion model in the presence of traps, a change in the nature of the permeability curves for temperatures of the test sample below 450 °C was described.

It was shown that the observed effects of a decrease in the deuterium flow through steel sample, as well as an increase in the time to achieve a steady-state deuterium flow through the sample, can be caused by traps.

Tentative

P1-149

Atomistic Modeling of Hydrogen Permeation in Vanadium Structural Alloys for FLiBe Blankets

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One of the key material science challenges in developing fusion energy is the design of structural and functional materials for the tritium breeding blanket. One of the leading blanket designs incorporates a lithium fluoride/beryllium fluoride (FLiBe) molten salt breeding blanket with a vanadium alloy (V-4Cr-4Ti) as the structural material. While V-4Cr-4Ti has favorable properties like low activation, high thermal stress factor, and good compatibility with liquid lithium, it has a high tritium solubility which will result in tritium permeation into the structural alloy requiring a tritium permeation barrier (TPB). Furthermore, the blanket will be operating in a neutron environment which will alter the microstructure of both the vanadium alloy and TPB and the mechanisms of tritium permeation in these materials.

In this work, we will report on the effectiveness of TPB materials using density functional theory (DFT) and the development of a V-Ti-Cr-H machine learned interatomic potential (ML-IAP) to study radiation damage and tritium permeation. Candidate TPB barriers materials will be assessed for tritium solubility, migration barriers, and adsorption using DFT to determine materials with favorable properties. In parallel, a V-Ti-Cr-H ML-IAP will be trained to reproduce key properties for studying radiation damage and tritium permeation such as defect and tritium formation energies and migration barriers. Complimentary hydrogen permeation and ion beam experiments to assess these materials will also be reported. SNL is managed and operated by NTESS under DOE NNSA contract DE-NA0003525

Tentative

P1-150

Atomistic simulations of tritium across a vanadium-lithium interface using machine-learning interatomic potentials

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Lithium-based liquids, ceramics and molten salts are candidate tritium breeder materials in future fusion reactors. For the case of solid and liquid lithium, vanadium and vanadium-alloys can act as the surrounding structural material due to their compatibility. The structure of the lithium-vanadium interface, particularly under irradiated conditions, becomes critical in the transport of tritium through the breeder blanket. There is lack of understanding in literature on the interaction of tritium at material interfaces in the reactor. Atomistic simulation techniques such as molecular dynamics (MD) can help in fundamentally analyzing the problem at the nano-scale, albeit with an accurate interatomic potential. In this work, we develop a machine-learning intertomic potential (MLIP) for Li-V-H. Specifically, we train an atomic cluster expansion (ACE) potential owing to it's ability in systematically capturing complex local atomic environments. The potential is trained not only on 0K density functional theory (DFT), but also finite temperature ab initio molecular dynamics (AIMD) configurations of tritium in solid V, liquid Li and at the interface. The potential shows excellent agreement in the calculated energies and forces, and predicts accurate tritium-vanadium and tritium-lithium binding and migration properties in comparison to DFT. MD simulations are performed using the ACE potential to analyze the stability and microstructure of the irradiated vanadium-lithium interface. Following that, simulations are carried out to study the temperature-dependent trapping and diffusion of tritium through the interface as a function of tritium content. The predicted effective diffusivity of tritium through the interface can be used in improving larger-scale transport codes.

Tentative

P1-151

Hardening of ion-irradiated and electron-irradiated single-crystal tungsten induced by hydrogen charging

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Tentative

The divertor of fusion reactors is exposed to high heat loads and neutron irradiation, which introduces defects and gas atoms. The gas atoms interact with the defects, and this leads to the change in the mechanical properties of the material. We investigated change in hardness of ion-irradiated and electron-irradiated single-crystal tungsten induced by hydrogen charging.

Discs with diameters of 3 mm and thickness of 0.5 mm were cut in single-crystal tungsten (99.999%). 6.4MeV Fe^{3+} ions were irradiated at the dual-ion irradiation experimental test facility, Kyoto University (approximately 5 dpa in the defect peak region). 7.5MeV electrons were also irradiated at electron linear accelerator of Institute for Integrated Radiation and Nuclear Science, Kyoto University (approximately 1.4×10^{-3} dpa). The irradiated samples were exposed to hydrogen gas at 543 K. Nanoindentation hardness of ion-irradiated samples and micro-Vickers hardness of electron-irradiated samples were obtained.

The micro-Vickers hardness of electron-irradiated samples increased after the irradiation. From positron annihilation lifetime measurements, vacancy clusters were not detected after the hydrogen charging at 543 K in electron-irradiated samples. Therefore, the hardness did not change after the hydrogen charging. Meanwhile, the bulk equivalent hardness of tungsten irradiated with ions at 573 K increased upon the hydrogen charging at 543 K. Because of the high irradiation doses in ion-irradiated samples, the formation of vacancy clusters and the trapping of hydrogen atoms to the vacancy clusters during the hydrogen charging led to the hardening. In the conference, we will discuss the hardening mechanism in more detail.

P1-152

Helium induced swelling and blister formation in fusion candidate materials

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The extreme conditions of the plasma-facing wall in a fusion reactor induce significant changes in material properties. One of the primary challenges is that exposure to intense helium bombardment, reported to cause bubble formation, fuzz growth, peeling and blistering-key phenomena that contribute to rapid material degradation and surface erosion.

In this work, we employ a Helium Ion Beam Microscope (HIM) as a rapid screening tool to induce localized helium damage in materials and to investigate the onset of blistering, swelling, and associated mechanical property changes.

We conduct helium implantation in tungsten (W), the leading candidate for fusion reactor materials, and analyze local blister formation as a function of helium dose. These results are then compared with other materials, including refractory high-entropy alloys (RHEAs) which contain W as a principle constituting element and vanadium (V). Our study identifies distinct material responses to localized high-dose irradiation and tracks the initiation and progression of blistering under intense conditions.

We have observed that for W, V, and RHEAs, helium bubbles coalesce, ultimately leading to crack formation and subsequent surface-near blistering. This process is accompanied by changes in mechanical properties changes, which we further examine using micromechanical testing and detailed transmission electron microscopy (TEM) analysis.

The study highlights distinct material responses to high-dose irradiation and advances understanding of helium-induced damage mechanisms in fusion environments

Tentative

P1-153

Multi-scale modelling of hydrogen isotopes retention and diffusion in highly irradiated materials

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A fundamental description of gas transport and retention in plasma-facing materials is crucial for tritium inventory modelling. During future reactor operations, the material microstructure is expected to evolve and reveal defects which act as “traps” for diffusing gas atoms, thereby compromising material performance. The classical formalism for gas diffusion and trapping is given by the McNabb-Foster equation [1] and Oriani equilibrium condition [2] to describe the reduction in diffusivity in the presence of traps. This “effective” diffusivity was initially proposed for single-occupancy traps under a dynamic equilibrium, where trapping and detrapping rates are equal.

In this work, we have developed a multi-gas, multi-trap, multi-occupancy and multi-dynamic transport code ‘Palioxis’ to bridge the gap between microscopic first principles and macroscopic partial differential equation approaches for modelling the diffusivity under irradiated conditions. We calculate the effective diffusivities of hydrogen isotopes in vanadium, tungsten and iron, as a function of homogenous vacancy concentration and gas concentration. In tungsten, the diffusivity monotonically increases with increasing hydrogen/gas content. However, a small drop in diffusivity is identified in vanadium and iron when gas and vacancy concentrations are comparable. The difference in behaviour between the elements is traced back to the difference in incremental binding energy of the nth hydrogen to an (n-1) vacancy complex. We validate the results from Palioxis by directly comparing effective diffusivities to those from molecular dynamics simulations performed with classical and machine-learning interatomic potentials.

[1] A. McNabb, P. Foster Trans. Metall. Soc. AIME 227 (1963)

[2] R. Oriani Acta Metall. 18 (1970)

Tentative

P1-155

Atomistic Response of Tungsten-Containing Refractory High-Entropy Alloys to Helium Ion Irradiation

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It is well known that the leading fusion material, W, suffers from severe surface modification under the bombardment of low-energy He flux in the fusion environment. We explore the response of refractory high-entropy alloys (HEAs) composed of Mo-Nb-Ta-V-W under He implantation and compare the results with pure W. To this end, we have parameterized the interatomic interactions between all the constitutive metals and He. With the developed empirical potential, we show that the formation and binding energies in HEAs are lower than those in W, suggesting that large He bubbles are thermodynamically unstable in HEAs. Simulations of He implantation further reveal that injected He distribution only weakly depends on the surface orientations in HEAs. In addition, the He bubbles in HEAs are much smaller and discretely positioned. We further show that the trap-mutation process in HEAs is ineffective for the growth of He bubbles, as interstitial defect diffusion is highly localized in HEA. Therefore, the sluggish motion of interstitial defects can help delay surface nanofuzz growth, considering that trap mutation and surface adatom formation mechanism dominate the fuzz growth. These results suggest superior resistance of HEAs to surface modifications under the impact of energetic He ion flux, rendering them promising candidate materials for fusion reactors.

Tentative

P1-156

Helium behavior of FeCrVTix medium-entropy alloys for advanced nuclear applications

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The service behavior of materials in extreme environment is one of the major bottlenecks restricting the utilization and development of fusion reactors. Compared with traditional steels such as T91 and SS316, some high/medium-entropy alloys (HEAs/MEAs) have obvious advantages in strength, toughness, high-temperature stability, irradiation resistance and corrosion resistance, etc., and have a wide range of the potential applications as nuclear structural materials.

The helium (He) behavior in FeCrVTix (x = 0, 0.1 and 0.3) MEAs irradiated with 1.4 MeV He2+ ions at 298 K, 573 K and 773 K was investigated. The irradiation fluences were 1×10^{16}, 5×10^{16} and 1×10^{17} ions/cm^2, respectively. The results show that there is a monotonic relationship between the He diffusivity and He bubble behavior, i.e., the higher temperature, the higher He diffusivity, the larger He bubble size, and the lower He bubble number density. This is consistent with the kinetic rate equations and classical nucleation theory. Most importantly, we find that the addition of Ti can significantly inhibit the growth of He bubbles in the BCC-matrix phase, which in turn reduces the swelling rate of He bubbles in the FeCrVTix MEAs. Furthermore, it is found that the He bubbles in the dual-phase structure FeCrVTix MEAs are over-pressurized at the irradiation temperature, and the anti-helium ability of the FeCrVTix MEAs is significantly better than that of some single FCC phase HEAs/MEAs (e.g. CoCrFeNi and CoCrMnFeNi). Our findings provide new ideas and methods for the design and development of HEAs/MEAs with excellent resistance to helium ability.

Tentative

P1-157

Suppressing the nucleation and growth of helium bubbles in WTaVCr high-entropy alloys

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High-entropy alloys (HEAs) have been demonstrated to show a particular suppressing effect on helium (He) bubbles, but the mechanism behind it remains unclear. Herein, we systematically investigate the dissolution and nucleation of He in WTaVCr HEAs by first-principles method combined with thermodynamic model. The solution energies of He in WTaVCr are much lower than that in pure W because of the modulation of electron density by different elements in WTaVCr. More importantly, this effect leads to a significant decrease of He-He binding energy in WTaVCr, which is at least 50% lower than that in pure W. This difference increases with increasing cluster size, indicating that the “self-trapping” of He will be significantly weakened in WTaVCr, thereby suppressing He blistering. Besides the self-trapping features, we further explore the growth of He bubbles in WTaVCr using molecular dynamic simulations. It is found that, different from the bubble growth in pure W via loop punching mechanism, the interstitials-type defects are scattered around the He bubbles and no loop formation/punching processes are observed. This interstitial-type cloud will inhibit the segregation of He atoms into bubbles, having a profound effect on the growth of He bubbles in HEAs. Our finding provides new insights into the suppression of He-induced radiation damage in nuclear materials.

Invitation

P1-158

Effect of chemical complexity and presence of vanadium on He dynamics in concentrated refractory alloys

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High-entropy alloys (HEAs) are known for their remarkable resistance to radiation and thermal stability, making them excellent candidates for fusion materials. While many studies have focused on the general response of HEAs to irradiation, the impact of helium formed in nuclear reactions or from fusion plasma on HEAs has received less attention. Using tabGAP combined with the pair potential for He-He, we address the He effects in refractory HEA (WTaVMoNb) as compared to W. We analyze He self-trapping phenomenon in several V-free and V-containing alloys by calculating binding energies of helium-helium and helium-vacancy, as well as the migration energy of helium using the climbing-image nudged elastic band method. In our atomistic simulations, we find an intriguing effect of the chemical environment on He migration and its agglomeration. In pure W and V-free alloys, the migration energy for a single helium atom is much lower than that of a single vacancy, which is, in turn, lower than the binding energy of a helium atom with a vacancy. On the contrary, in V-containing alloys, there is a strong coupling between these energies. This mechanism explains the differences in helium atom diffusion in materials with and without V. Additionally, we examine the motion of a single vacancy bound by one helium at various temperatures analyzing the mean squared displacement. The results show that in W, helium diffusion is hindered by the presence of vacancies even at high temperatures, whereas helium can leave the vacancies in WV for some time at high temperatures.

Tentative

P1-159

Multiscale simulation of hydrogen isotope diffusion in irradiated tungsten

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[1]UKAEA

Recent work has shown that massively overlapping molecular dynamics cascade simulations produce irradiated microstructures directly comparable with experiments performed below the temperature for vacancy migration[1]. For a tungsten divertor monoblock, such a lower-temperature region exists near the copper cooling pipes. Here a large number of irradiation-induced defects will be formed by neutron damage which can trap hydrogen isotopes (HIs), with an estimated 1% atomic fraction hydrogen or more being retained in monovacancies[2].

Here we will consider the question of whether tritium entering the surface of a monoblock at high flux from the plasma will reach this cooler region, and if it does, whether it will be possible to remove it. Recent developments in both empirical[3] and machine-learned potentials for the tungsten-hydrogen system now make it possible to directly compute the diffusion properties of HIs in heavily irradiated microstructures. We will report the effective diffusivity of HIs in local equilibrium with vacancies, dislocation loops and grain boundaries, and show how this can be modelled with simple diffusion equations, elucidating the role of entropy, isotope exchange, and the necessity of considering multi-occupancy traps.

Finally, by solving for the diffusion with a specially-built MOOSE[4] module, we make a prediction for how deuterium and tritium penetrate and accumulate in a divertor monoblock.

[1] Boleininger et al Sci Rep 13: 1684 (2023)
[2] Mason et al Phys Rev Mat 5:095403 (2021)
[3] Mason et al J Phys Cond Matt 35:495901 (2023)
[4] Giudicelli et al SoftwareX 26:101690 (2024)

Tentative

P1-160

Helium behaviors on the microstructure and mechanical properties of the Cansas 3303 silicon carbide composites

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SiC/SiC composites are considered as the prospective candidate material for fusion blanket structure due to their excellent high temperature chemical and mechanical stability, low activation and radiation resistance. This material will suffer from serious performance degradation during long-term service in fusion reactor. Especially, helium ions, which will be generated by nuclear transmutation or deuterium-tritium reaction, can play a significant role for the microstructure evolution of the materials. Therefore, it is important to investigate and unravel the helium behaviors in the SiC/SiC composites. In current work, the helium behaviors on the microstructure and mechanical properties of the Cansas 3303 SiC fiber reinforced SiC/SiC composite are presented.

The Cansas 3303 SiC fiber reinforced two-dimensional SiC/SiC composite was fabricated via chemical vapor infiltration process. Helium ions with the energy of 40 keV were implanted to the fluences of 1E16, 5E16 and 1E17 ions per square centimeter at room temperature by using an ion accelerator. Microstructural feature was characterized by using micro-Raman spectra and transmission electron microscopy. The formation and evolution of helium bubbles was investigated in both the Cansas 3303 fiber and SiC matrix. Helium implantation-induced amorphization accompanied with the decrease of SiC grain size was also observed. Irradiation hardening was evaluated via a nanoindentation hardness test technique. The decrease of nanohardness was detected. By combination of the micro-Raman spectra, nanohardness testing and TEM analyses, the different helium behaviors and the underlying mechanism between the Cansas 3303 fiber and SiC matrix are discussed.

Invitation

P1-161

The Comparison of Hardness between MAX Phase Materials Irradiated with Protons and Neutrons

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MAX phase materials, as candidates for advanced nuclear systems, possess excellent mechanical properties, corrosion resistance, high-temperature stability, and resistance to radiation damage. Extensive ion irradiation studies have been conducted on MAX phase materials. However, quantitatively comparing the irradiation effects of ions and neutrons is still an important scientific subject. Therefore, this study focused on the mechanical parameter of hardness and selected three MAX materials, Ti2AlC, Ti3AlC2, and Ti3SiC2, for proton and neutron irradiation experiments to compare their hardness after the irradiations. The main results obtained were as follows:

(1) Nanoindentation and Vickers hardness tests were conducted using a 50 gf load. The two hardness values were found to be equivalent within error range after applying the conversion method, making quantitatively comparing the hardness of ion-irradiated and neutron-irradiated samples with relatively large damage plateau regions possible.

(2) XRD and TEM results showed that there was almost no change in the crystal structure of samples irradiated with ions and neutrons.

(3)Raman spectroscopy results indicated differences in chemical bond damage between proton-irradiated and neutron-irradiated samples. Even with similar dpa damage levels from protons and neutrons, due to the significant contribution of electron energy loss during proton irradiation and the formation of more defect clusters through cascade collisions during neutron irradiation, the results of chemical bond damage also differed. It is speculated that the interplay between chemical bond damage and the introduction of defects results in the change in hardness of MAX phase materials by the irradiations.

Tentative

P1-162

The Effect of Alloy Composition on Mechanisms of Helium Diffusion in MoNbTaTi High Entropy Alloys

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The evolution of helium (He) induced structural property changes in fusion materials is controlled via the ability of He to migrate or diffuse through the host lattice, the presence of vacancies as trapping sites, and the existence of matrix inhomogeneities capable of promoting heterogeneous bubble nucleation. The excess production of helium in fusion systems necessitates increased radiation tolerance of structural materials regarding He bubble formation, coalescence, and migration. High entropy alloys (HEAs) may possess properties capable of altering the nucleation and growth schemes of He bubbles in metals due to their complex structure and diffusional landscape. Designing complex alloys with a higher resistance to He-induced macroscopic property degradation requires a mechanistic understanding of He diffusion and clustering in complex energy landscapes on an atomistic scale. This work seeks to explore the impact of HEA composition on He diffusion and self-clustering. Large-scale Atomic/Molecular Massively Parallel Simulator (LAMMPS) is used to simulate interactions of He in 17 compositions of a MoNbTaTi HEA. He diffusion coefficients, interstitial He formation energies, and interstitial He atomic volumes were measured for each composition. These values were compared to simulations of He clustering in larger simulation cells randomly pre-seeded with atomic He. The low solubility of He in metals suggests that He may preferentially diffuse to areas with more available free volume. This concept is examined by comparing He diffusion coefficients with the atomic volume occupied by He in various alloy compositions and analyzing local He trapping sites.

Tentative

P1-163

Simulating plasma-surface interactions with deuterium beams experiments: effect of oxygen and carbon impurities on fuel retention and recycling

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Tentative

Harnessing fusion's power is ITER's goal and power exhaust will occur on tungsten tiles. During plasma discharges, tungsten heats inducing an evolution of fuel retention and recycling which can trigger plasma reattachment/detachment [1]. In a recent experimental-modelling study, we demonstrated that tungsten surface oxidation influences fuel retention partitioning between bulk and surface, impacting fuel recycling [2]. Fusion reactors exhibit varying oxygen and carbon contamination of tungsten tiles, from sub-monolayer to micron-thick layers [3]. Thus understanding how oxygen and carbon affect fuel retention and recycling is important. We present beams experiments clarifying these plasma-wall interactions.

Firstly, sub-monolayers of oxygen or carbon are deposited on clean tungsten. Both impurities modify the repartition of deuterium retention (measured by TPD) between the bulk and the surface. Surface retention is reduced by the presence of oxygen or carbon. However, the desorption temperature with respect to clean surface is reduced by 100 K with oxygen but increased by 100 K with carbon. These striking differences should lead to significant effects on fuel recycling.

Secondly, ~100 nm thick tungsten oxide (WO_3) is implanted with a 250 eV/D ion beam. For fluence >1021 D^+/m^2, we measure a ten-fold increase in deuterium retention with respect to native oxide, concomitant with microstructure modifications (FIB-TEM): the surface is oxygen-depleted as expected from sputtering, the sub-surface is amorphized as expected from displacement damage and, surprisingly, an oxygen-depleted layer is found 10 nm below the surface. Rate-equation modelling based on DFT data are presented.

[1] https://doi.org/10.1063/1.5140183 (2020)
[2] https://doi.org/10.1088/1741-4326/ad2a29 (2024)
[3] https://doi.org/10.1088/1402-4896/ac267e (2021)

P1-164

Damage microstructure, hardening and deuterium retention in neutron irradiated tungsten and tungsten-rhenium alloy

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Tungsten is a key candidate armour material for plasma-facing components in ITER and has been planned for use in DEMO. During service, simultaneous high flux thermal loading, plasma-wall interaction and neutron damage accumulation may lead to severe embrittlement and possible loss of structural integrity in the material. A reliable prediction of tungsten performance requires the establishment of a quantitative evolution map of neutron damage microstructure, as well as an accurate correlation model connecting defect statistics and material property degradation. In this work, a summary of recent progress on neutron irradiations in pure tungsten (CVD-W, W-rod) and tungsten-rhenium alloy is presented. The specimens were subjected to neutron irradiation in the Belgian BR2 reactor, across a temperature and fluence range of “1 MeV)”, respectively, to mimic the damage effects of fusion neutrons. By combination of defect microscopy, hardness evaluation, linear plasma exposure and thermal desorption spectroscopy analysis, a damage microstructure map is unveiled, and the irradiation hardening and deuterium retention behaviours are investigated. Notably, a pronounced suppression effect has been confirmed for rhenium in tungsten, upon the evolution of irradiation-induced interstitial and vacancy-type defects.

Tentative

P1-165

The effect of annealing temperatures on plasma induced surface blistering and deuterium retention in tungsten

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Tungsten is a candidate for plasma-facing materials in fusion reactors due to its excellent properties. However, issues like hydrogen isotope retention and surface blistering during the period of service can degrade its performance. Annealing treatment is commonly used to relieve residual stress but can also alter the defect density and influence the blistering and deuterium retention behaviors. To clarify these effects, samples were annealed at 473K, 673K, 873K, 1073K and 1273K, respectively, with unannealed sample and sample recrystallized at 1973K serving as references. Nanoindentation test was conduct, revealing that the hardness of the samples initially increases and then decreases as the annealing temperature rises. The maximum shear stress, calculated based on pop-in events, increases with annealing temperature, indicating a decreasing trend in dislocation density. Then the samples were exposed to deuterium plasma with a flux of 5.2×10^18 D/(m^2・s) and a fluence of 2.0×10^23 D/m^2. Blisters were observed on the surface of samples except the recrystallized one. The average sizes of blisters on each sample were similar, while the number densities of blisters exhibited an abrupt drop when annealed at 1073 K. The deuterium retention of samples was analyzed by thermal desorption spectroscopy (TDS) and it decreases as annealing temperature increases. The desorption curves were fitted by TMAP7 code and the fitted results suggests a decreasing concentration of traps. All the results indicate that annealing reduces defect density, thereby mitigating blistering and decreasing deuterium retention.

Invitation

P1-166

Hydrogen trapping, desorption and clustering in heterophase interfaces of W-ZrC alloy

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Understanding the interaction of hydrogen (H) with defects has been recognized as a central aspect in hydrogen-induced damages in metals. In this work, the role of coherent and semi-coherent interfaces between W and ZrC on H trapping, desorption and clustering in nanostructured W alloys was investigated by density functional theory calculations. It is found that, the trapping ability of pure interface is comparable with the intrinsic defects (grain boundary, dislocations) in bulk W, with segregation energies of H ranging from -1.37 eV to -0.98 eV. While the interfaces with W/C vacancy show stronger trapping capabilities, especially for the incoherent interfaces. This leads to low peak temperatures of 390-567 K for H desorption from the pure interfaces and high peak temperatures of 600-830 K and even 1100 K from the vacancy-containing interfaces. As the number of H atoms increases, they preferentially segregate to the interfacial W vacancy and divacancy, and then stay at the interface and interface-like zones discretely until the adsorption rate reaches 3/8. The strong traps of phase interfaces could effectively absorb H from the edge dislocation cores and bulk vacancies in W, hence preventing H-induced damage. This work advances atomic-level understanding of the role of phase interfaces in H trapping and clustering, not only offering direct explanations for experimental H desorption observations, but also pointing to a feasible route towards interface engineering against H-induced damage in structural metals.

Textura

P1-167

Using Helium Ion Microscopy to understand the mechanisms of helium irradiation blistering in tungsten

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Helium irradiation blistering in tungsten is of intense interest for plasma-facing materials under fusion-relevant conditions. Previous studies have revealed that the surface blistering is closely related to the evolution of subsurface microstructures (e.g., bubbles and cracks), but its dependence on the surface orientation and deformation remains unclear. The present work reports the helium irradiation blistering on the surface planes of poly- and single-crystalline tungsten {100}, {110} and {111}. A systematic study of helium irradiation blistering was conducted at room temperature by utilizing a Helium Ion Microscopy. The cross-sectional microstructural analyses at different ion fluences confirmed that the blistering deformation starts with the nanocrack nucleation, initiated at the peak in the bubble depth distribution via interbubble fracture, and is followed by the gas buildup in a primary cavity. In addition, the comparison of blistering behaviors between 1-μm-diameter and 2-μm-diameter circular areas revealed that only the blisters formed on tungsten {100} can grow to a larger size, whereas those formed on tungsten {110} or {111} tend to become wrinkled or burst when the irradiated area increases. Combined with the micropillar compressions in a Scanning Electron Microscopy, it was also found that the surface deformation is orientation-dependent and associated with the number, symmetry and distribution of effective slip systems in body-centered-cubic transition metals. Our findings provide new evidence and insights into the mechanisms of helium irradiation blistering and surface deformation in tungsten.

Tentative

P1-168

Effect of multiple trapping of hydrogen atoms on formation energy of vacancy-single helium complexes in α-iron

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It has been reported that the mechanical properties of structural materials in fusion reactors deteriorate due to the formation of helium bubbles. Therefore, rate theory calculations have been conducted to predict the growth of helium bubbles during the operation of actual fusion reactors. Since hydrogen atoms are also formed in fusion reactor materials in addition to helium, it is necessary to consider the effect of hydrogen on helium bubble growth. As a first step, we investigate the effect of multiple trapping of hydrogen atoms on the formation energy of a vacancy-single helium complex using density functional theory (DFT) calculations. The DFT calculations were carried out using the Vienna Ab initio Simulation Package code. Core electrons were described by a generalized pseudo-potential of the projector-augmented-wave (PAW) method. The plane-wave energy cutoff was 520eV. 4×4×4 shifted k-point grid of the Monkhorst-Pack scheme was used. The supercell size was 4×4×4 unit cells (128 atoms). The formation energy of vacancy-hydrogen-single helium complexes were obtained. The formation energy of a vacancy-single helium complex was 4.38 eV. When the number of hydrogen atoms trapped at the vacancy-single helium complex increased up to 3, the formation energy decreased to 3.82 eV. In the number of hydrogen atoms of 4-8, it did not change (3.81-3.85 eV). In the number of hydrogen atoms of more than 9, it increased. In the conference, we will present the estimation of the number of hydrogen atoms trapped at the vacancy-hydrogen-single helium complex under fusion reactor operating temperature.

Tentative

P1-169

Disentangling Hydrogen- and Defect-driven Segregation in the Iron-Tungsten-System

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Reduced-activation ferritic-martensitic (RAFM) steels as EUROFER97 are envisaged as structural material and proposed as alternative plasma-facing material in fusion plasma devices [1]. Under hydrogen, i.e., deuterium (D) bombardment at elevated temperatures, they exhibit an unexpected surface morphology formation [2,3] with enrichment of the alloying element tungsten (W).

To disentangle involved mechanisms of the formation of this altered surface layer, a thin (0.4 µm) iron-tungsten (Fe-W) layer with 3at.% W was deposited on Fe by magnetron sputtering. The samples were subjected to various treatments, ranging from heating in vacuum or D atmosphere to D plasma and D ion beam exposures at elevated temperatures (up to 800K). The D energy and flux (applied for 3h) were varied (30 and 200eV; 5e18, 6e19 and 1e20$m^{-2}s^{-1}$). The samples were pre- and post-characterised by Rutherford backscattering spectrometry and scanning electron microscopy assisted by focused-ion-beam cutting.

As the layer modifications are significantly different for annealing in vacuum from those on D-bombarded samples, thermal segregation can be excluded. Furthermore, the thickness of the modified layer (>0.1µm) with W-rich precipitates exceeds the D implantation range, which is only several nanometres for the used energies. The contribution of different W-enrichment mechanisms in relation to radiation-induced effects [4] and the question whether an effect of hydrogen-driven segregation is necessary to explain the W-enrichment will be discussed.

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Invitation

P1-170

Influence of hydrogen on the displacement defect evolution and mechanical properties of tungsten

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Tungsten (W) is considered as the promising candidates for plasma facing materials in future fusion reactors, and thus will be irradiated by neutrons and hydrogen isotopes (HIs). This triggered enormous research effort on the evolution of displacement defects and HIs in W as well as their interactions, especially considering the tritium self-sufficiency. However, most previous studies have focused on the influence of displacement damage on the behaviours of HIs, while the role of HIs in affecting the defect evolution and mechanical properties remains to be elucidated.

Herein, using molecular dynamic simulations, we explicitly demonstrate that the low-energy HIs irradiation can create the displacement defects in W surface directly. This unexpected result can be attributed to the double-hit process and outward collision, and thus provides a reasonable interpretation of defect production at sub-threshold energy. Moreover, the H segregation into vacancy clusters significantly suppresses the vacancy-interstitial recombination, which is triggered by an intriguing inner surface stress release. Accordingly, we develop a predictive model to quantify the H-induced reduction of recombination radius, and our model calculations of the H-defect co-evolution agree excellently with recent experiments. Furthermore, we demonstrate that the H addition substantially amplifies the pinning effect of voids/loops on edge dislocation. This is rationalized by the unusual H overflowing from voids and the strong H-loop attractions.

Our findings showcase the importance of HIs in affecting the response of W-PFMs to irradiation, shedding new lights on understanding the production and recombination of displacement defects as well as the irradiation hardening.

Tentative

P1-171

Changes in the neutron-irradiation damages to W due to the presence of H

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With a shallow migration barrier, self-interstitial atoms generated by neutron irradiation quickly diffuse out of the damage cascade, leaving less mobile vacancies behind. An ideal single crystalline tungsten would self-heal to some extent as the Frenkel pairs will eventually recombine. It is therefore grain boundaries that truly triggers the irradiation damage beyond material's self-healing limit.

Nevertheless, it is still unclear how materials degrade in the presence of grain boundaries, and in particular, in the presence of hydrogen atoms. In this talk, we would like to present our preliminary results from a set of atomistic calculations on how grain boundaries as well as hydrogen atoms would change the irradiation damages in W. Grain boundaries notably absorb hydrogen atoms, which subsequently changes the kinetics of point defects around grain boundaries. We try to quantify this chain reaction with various analysis tools such common-neighbor analysis.

Tentative

P1-172

Helium Segregation and Diffusion Behavior at Grain Boundaries in Tungsten: A Molecular Dynamics Study

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Tungsten and its alloys are considered the most promising plasma-facing materials for future nuclear fusion reactors. However, helium atoms generated by deuterium-tritium fusion reactions and neutron transmutation can significantly degrade the mechanical properties of tungsten. Given that grain boundaries (GBs) act as defect sinks in materials, this study focuses on tungsten and employs molecular dynamics simulations to investigate the segregation behavior of helium atoms near GBs.

In this work, symmetric tilt GB models with rotation axes of and were constructed. Qualitative analysis revealed key factors that influence single helium segregation near GBs. To obtain stable configurations of multiple helium atoms, a random placement method was employed. It was discovered that the binding characteristics of helium clusters (cluster size ranges from 1 to 6) near GBs are closely related to their cluster structures. Notably, under the influence of GBs, helium atoms are driven toward the GB plane and continue to diffuse along the interface. Based on these findings, the diffusion behavior of helium atoms near GBs under different temperatures was systematically analyzed. Within the simulated time scale, as temperature increases, helium atoms exhibit various forms of channel diffusion, accompanied by the generation of multiple transition states. Additionally, due to differences in diffusion activation energies along different directions at varying temperatures, the spatial dimensionality of helium diffusion undergoes shifts. Furthermore, the diffusion behavior of helium clusters near GBs was investigated. The results offer atomic-scale insights into helium diffusion mechanisms in tungsten, providing a theoretical foundation for future research.

Tentative

P1-173

A DFT- and Machine Learning-based Study of Tritium Retention in Titanium Beryllide

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Titanium beryllide is a promising solid neutron multiplier for achieving a sufficient tritium breeding ratio in future fusion reactors. Although it retains less tritium than pure beryllium, the inventory in approximately 450 tonnes of material remains unquantified. Tritium accumulation in neutron multipliers poses safety risks, including accidental release during operation and challenges for waste processing and recycling, and therefore needs to be evaluated.

This study investigates tritium retention in $TiBe_{12}$ based on the assumption that tritium accumulates on the facetted surfaces of helium-filled bubbles, as it was observed in pure beryllium. DFT calculations are used to model the free surfaces of TiBe12 with a tetragonal crystal lattice to identify the most stable and energetically favorable orientations and surface terminations for tritium adsorption. The lowest energy surfaces are {110}, followed by {100}, {101}, {111}, and {001}, with titanium-terminated surfaces being the most stable.

To assess tritium retention, static relaxation was used to identify hydrogen surface adsorption sites, and DFT molecular dynamics simulations, accelerated by on-the-fly machine learning, were used to estimate the maximum tritium surface concentration at different temperatures. The tritium inventory is then assessed using the experimental helium bubble size distributions from HIDOBE-irradiated beryllides and surface occupancies from molecular dynamics.

Future work will include simulation of helium bubble nucleation and growth to refine tritium retention predictions. These findings will contribute to a deeper understanding of tritium behavior in neutron multipliers, and support material design for safer, more efficient fusion reactors.

Template

P1-174

Tritium Material Interactions Research at the Savannah River National Laboratory

Dale Hitchcock

Tritium like all hydrogen isotopes is soluble in all known materials and exhibits deleterious effects on the mechanical properties of most materials. Concerns related to tritium compatibility of materials are closely correlated to both concentration and total inventory of tritium in the system. At relatively low concentrations soft materials (i.e. polymers) may be considered for tritium wetted applications. In areas where high concentrations or inventories are present, especially at elevated temperatures, even the austenitic stainless steels preferred for tritium service experience significant tritium embrittlement.

The Tritium Technology Division at the Savannah River National Laboratory has been researching large scale tritium processing and tritium effects on materials for decades. From polymers to structural materials SRNL has performed accelerated aging on a wide range of materials of interest for tritium processing. Additionally, characterization of hydrogen isotope permeation and retention in a wide range of materials for multiple applications is ongoing. This work includes the characterization of the hydrogen permeability of low activation alloys for fusion applications after neutron irradiation.

This talk will give an overview of the current tritium materials interactions work currently being performed at the Savannah River National Laboratory.

Tentative

P1-175

Oxidation and environmentally-assisted cracking behaviour of Eurofer-97 under simulated water-cooled breeder blanket conditions

Liberato Volpe[1], Peter Andresen[2], Dora Capone[3], Eric Prestat[1], Joven Lim[1], David Kumar[4], Ronald Clark[4], Robert Burrows[4], Fabio Scenini[5]

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The reduced activation ferritic martensitic (RAF/M) Eurofer-97 steel has been selected as the coolant-facing alloy for the water-cooled breeder blanket (WC-BB) section of the European DEMO fusion reactor. However its oxidation behaviour and its susceptibility to environmentally-assisted cracking (EAC) under WC-BB environmental conditions are not known. Therefore, a water chemistry specification must be identified to mitigate both oxidation and EAC under the expected aqueous cooling environment. In this work, coupons and 1T-C(T) specimens of Eurofer-97 were exposed to high temperature water at 300 °C with ~4 ppm of H_2 with either LiOH or KOH as an alkalising agent, and at pH300°C equal to 7.0, 7.4 and 8.4, with and without the addition of 5 ppb of Zn to evaluate the oxidation and the EAC behaviour.

After exposure, a duplex oxide, composed of an inner $FeCr_2O_4$ polycrystalline layer and external octahedral Fe_3O_4 crystallites, formed on the coupons irrespectively of the alkalising agent and pH value. When injected, Zn was only observed on the outer layer of the outer Fe_3O_4 crystallites, showing limited diffusion into the inner oxide. Crack propagation was identified only under cyclic loading conditions, with a steady decrease and eventual stopping at constant loading condition. Moreover, it was not affected by the type of alkalizing agent or the pH value. This behaviour indicates that Eurofer-97 is susceptible to corrosion fatigue but not to stress corrosion cracking. Eurofer-97 showed enhanced resistance to oxidation and stress corrosion cracking, making it suitable for the proposed WC-BB conditions in the DEMO reactor.

Invited

P2-001

Stability of Lab-Scale 14%Cr Oxide dispersion-strengthened steel (ODS) produced by STARS© route

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Oxide Dispersion Strengthened (ODS) steels can be an important solution for applications requiring high strength, radiation resistance and durability in fusion reactor design. The highly demanding characteristics of fusion, including the need for low activation materials, make the development and study of this kind of materials essential.

The STARS© route replaces traditional manufacturing methods that include mechanical alloying of powder. It consists of gas atomization with controlled oxidation, enabling precise stoichiometric control, reduced contamination, and uniform dispersion of nano-oxides. This represents a significant advancement in terms of reproducibility and cost reduction.

In this work, a 14%Cr ODS steel produced using the STARS© route at a laboratory scale has been manufactured and thoroughly characterized, both mechanically and microstructurally.

Understanding of the microstructure has allowed the identification of hardening mechanisms and their contribution to mechanical strength, as well as the most suitable thermomechanical treatments. The STARS route produces a refined grain structure with optimized precipitation of secondary phases, including Y-Ti oxides and Cr-rich carbides, critical for enhancing radiation resistance and mechanical performance. Heat treatments at 1100 °C demonstrated an ideal balance between grain refinement, dislocation density, and precipitate uniformity, leading to outstanding material stability. The demonstrated performance opens the door to the production of large quantities of material for exploration at a semi-industrial scale.

The results underscore the STARS method's potential to revolutionize the field of fusion material engineering, offering unmatched control over microstructural features and paving the way for its adoption in next-generation fusion technology.

Tentative

P2-002

Performance of high-Cr oxide dispersed strengthened steels under high-dose neutron irradiation

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Assessing the radiation resistance of nanostructured ferritic alloys (NFAs) is crucial for their application in high-temperature fusion reactors. To achieve this essential qualification, three NFA variants (14YWT-SM12a, c, d) were irradiated in the High-Flux Isotope Reactor at ORNL up to doses of 20 dpa at temperatures of 265°C and 590-640°C.

After irradiation, 21 tensile tests were performed at both room and irradiation temperatures. The yield strength of the irradiated samples was 5 to 15% higher after exposure at the 265°C and 2 to 15% lower after elevated-temperature irradiation. A similar trend was observed for the ultimate tensile strength. The change in uniform elongation (UE) due to irradiation did not show a clear trend as both increases and decreases were observed. The irradiated UE values remained within ±2% (absolute) from the unirradiated ones. The total elongation of the irradiated samples was consistently lower (by 1 to 12%) than before irradiation.

Detailed transmission electron microscopy and atom probe tomography studies demonstrated that the low-temperature irradiation hardening might originate from the Fe-Cr matrix decomposition, leading to the formation of alpha-prime precipitates, radiation induced dislocation loops, and nano-cavities. The origin of the softening after the high temperature irradiation is less clear. The size and number density of the yttria-based oxides remained stable. However, the diffusion of tungsten from the solid solution towards grain boundaries, leading to the formation of Laves phase, as well as the reduction in dislocation density due to high-temperature annealing, may contribute to the observed decrease in yield strength.

Tentative

P2-003

Preparation and properties of high-density oxide strengthened ODS steel by liquid metal method

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Oxide dispersion strengthened (ODS) steel exhibits exceptional high-temperature, irradiation and corrosion resistance, as one of the most promising structural materials for fission and fusion reactors. However, large-scale fabrication issues limit its industrial application. The liquid metal (LM) fabrication is a developing method for the large-scale production of ODS steel. Nevertheless, it is difficult to obtain high number density and uniformly distributed nano-oxides in steels by LM method, due to density difference and poor wettability of oxide with molten steel. In this study, two kinds of 9Cr-ODS steels strengthened with high-density oxides (i.e., Ta-rich oxide, Y/Ta-rich oxide) were fabricated using intermediate alloy combined with sub-rapid solidification. The density of the two oxides is higher than or close to that of molten base steel (Fe-9Cr-1.5W). The results demonstrate that this approach successfully introduced submicron and nanoscale Ta-rich and Y/Ta-rich oxides into the two samples. The hardness of the samples increased by 35.8% and 50.4%, respectively, compared with that of the base steel. After hot rolling, the grain size and dispersed phase distribution in the samples were further refined, accompanied by additional improvements in hardness. The microstructure and long-term mechanical properties at high temperature of the two samples are under investigation. This study will provide a candidate strategy for the design and fabrication of ODS steel, offering a potential method to overcome the difficulties in their large-scale production.

Tentative

P2-004

The S/TEM Characterization of MA956 ODS Steel after SHI Irradiation

Xuan MENG[1], Yitao YANG[2], Zeyu HONG[2], Chonghong ZHANG[2]

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Studies on the radiation hardening of bulk materials under extreme conditions have received significant attention in the context of nuclear materials. For the purpose of clarifying the basic mechanism of material failure caused by irradiation damage, the microstructures of materials under extreme radiation conditions were evaluated. Oxide dispersion strengthened (ODS) steel is regarded as a promising candidate material for advanced nuclear systems. Therefore, the material used in this study is a commercial ODS steel (MA956), and the SHI irradiation was performed at HIRFL located in the Institute of Modern Physics, Chinese Academy of Sciences. Multiple Fe ion irradiations with various energies (hundreds of MeV) were employed in order to produce a displacement damage plateau.

The radiation defects of dislocation loops were characterized under kinematic bright-field (KBF) conditions by a conventional TEM operated at 200kV. Both two types of dislocation loops (loops and 1/2 loops) were observed. The size and number density of loops were determined using a pair of g = and g = micrographs taken from an orientation near pole . For comparison, the determination of dislocation loops character and number density were also examined by selective dark-field (DF) imaging at off-axis and on-zone conditions in STEM mode, which is referred as diffraction-selected on-zone STEM (DsoZ-STEM) condition. DsoZ-STEM was conducted by a spherical aberration-corrected TEM, and the annular DF (ADF) images and the bright-field (BF) images were obtained to characterize the dislocation loops.

Tentative

P2-005

Effect of milling atmosphere on microstructure and mechanical properties of ODS-Steel fabricated by spark plasma sintering

Cheng Cheh Dennis Tan[1], Baisong Chen[1], Li Tian Chew[2], Kok Heng Cheong[1], Wei Hock Teh[2], Yakai Zhao[1], Fengxia Wei[2], Jing Jun Lee[2], Na Gong[3], Chun Yong Andrew Ngo[1]

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Oxide dispersion strengthened (ODS) steel is a structural material with excellent mechanical properties and radiation resistance, so it is one of the main candidates for structural materials for nuclear fusion. In this paper, the powder was ball-milled in air and protective atmosphere by mechanical alloying method, and then ODS steel (14CrYWTi) samples were prepared by spark plasma sintering (SPS) technology. Finally, the effect of high temperature annealing process on microstructure and mechanical properties was studied. The results show that under protective atmosphere, $Y_2Ti_2O_7$ (Ti-Y-O) nanoclusters are the main precipitated phase. Under non-protective atmosphere, since nitrogen in air can easily combine with Ti in powder to form TiN phase, the large amount of Ti consumption reduces the amount of $Y_2Ti_2O_7$. High temperature annealing promotes the transformation from residual FCC to BCC ferrite, while maintaining a unique bimodal grain structure (composed of micron-sized grains aggregated into round shapes, surrounded by ring-shaped nano-sized grains). The test results of mechanical properties show that the bimodal grain size distribution helps to improve the strength and toughness of the material at the same time. The difference in mechanical properties between the two ball milling atmospheres is not significant, and the effect on irradiation properties needs to be further explored.

Tentative

P2-006

Evaluation of oxide dispersion strengthened high entropy alloys fabricated by laser additive manufacturing

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he application of the metal additive manufacturing (AM) method for nuclear reactor materials has been actively studied. In this study,oxide dispersion strengthened (ODS) alloys were fabricated by the AM method to enhance the strength and irradiation resistance of CrFeMnNi-based FCC-type high entropy alloys (HEAs), which are potential candidate for fusion reactor that exhibit excellent irradiation resistance at high temperatures and low activation properties. The effect of oxide addition on mechanical properties was investigated by comparing HEA and ODS-HEA.

$Cr_{0.8}FeMn_{1.3}Ni_{1.3}$ alloys, with and without 0.3 and 0.6 mass% Y_2O_3 additions, were fabricated using the selective laser melting (SLM) method. Mechanical characterization and microstructural analysis were conducted using tensile tests, EBSD, TEM, and STEM-EDS.

Tensile test results showed that the $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$ base material exhibited higher yield strength than the Y_2O_3-added material. Microstructural observation revealed elongated columnar crystal grains along a build direction, characteristics of AM processing. Grain size measurements indicated that the tensile direction grain size of the Y_2O_3-added material was larger than that of the undoped material, suggesting that the reduction in strength was due to grain coarsening. TEM observations found that the areal density of oxide particles increased with higher Y_2O_3 content, implying that precipitation strengthening contributed to the materials properties. Further characterization of oxide particle morphology including size distribution and chemical composition analysis was conducted and discussed as a function of processing parameters in the specimen fabrication.

Tentative

P2-007

Development of Machine Learning Interatomic Potentials for HfNbTaTiZr Alloys

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HfNbTaTiZr alloys are one of the most studied refractory high-entropy alloys (RHEAs) due to their mechanical properties, including high strength and ductility at elevated temperatures. However, fundamental understanding of key atomic-scale mechanisms remains incomplete. One of the limitations is the lack of accurate interatomic potentials that can effectively bridge the gap between density functional theory (DFT) accuracy and large-scale atomistic simulations (MD). This work presents the development of efficient and accurate machine learning interatomic potentials utilizing tabulated GAP potentials and neuroevolution potentials (NEP). These models are trained on DFT databases in order to capture many-body interactions with quantum accuracy using both equiatomic and non-equiatomic compositions of HfNbTaTiZr. The developed potentials make large-scale MD simulations possible in the study of properties such as phase stability, segregation, and radiation damage. We find that the equiatomic alloy stabilises as BCC or HCP depending on temperature, in line with experimental observations. Compositions with enhanced mechanical and thermal properties can thus be studied further on.

Tentative

P2-008

Disorder-order phase transformation in ion-irradiated Co-free high entropy alloy

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Over the last decade, different high entropy alloys (HEAs) have been investigated for their application in advanced nuclear reactors because of their excellent high-temperature properties and irradiation tolerance. HEAs have shown promising resistance to irradiation-induced defects which is attributed to a strong lattice distortion, decreased electrical and thermal conductivity, and sluggish diffusion of point defects (vacancies/interstitials), all of which influence the defect evolution. Here, we have investigated the irradiation tolerance of a novel Co-free high entropy alloy that is irradiated using 5 MeV Fe ions at 300 °C to 3 dpa (at depth of 500 nm). The microstructural investigation of pristine HEA sample, performed under transmission electron microscope (TEM), revealed prominent planar dislocation slip, suggesting the presence of short-range order and low stacking fault energy. The existence of sharp superlattice reflections in the selected area diffraction from the irradiated region corresponds to the disorder-order phase transformation (i.e., FCC to $L1_2$). Both 1/2<110> type perfect as well as 1/3<111> type faulted loops were observed at the analysed depth. However, no void formation was observed. This can be attributed to lower irradiation temperatures (~0.2 Tm) and sluggish diffusional pathway of vacancies/interstitials, which enhances the vacancy-interstitial recombination. Additionally, the disorder-order phase transformation also assists in enhanced irradiation resistance.

Tentative

P2-009

Effect of Re, Ta Additions on the Mechanical Properties and Microstructure of Spark Plasma Sintered ODS Tungsten Alloys

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Tungsten (W) is a promising material for high-temperature structural applications and plasma-facing components in nuclear fusion reactors due to its high melting point, excellent wear resistance, and superior mechanical strength. However, pure tungsten exhibits a high ductile-to-brittle transition temperature (DBTT) and limited toughness, which restricts its practical applications. To address these challenges, this study investigates the fabrication and mechanical property evaluation of oxide dispersion strengthened (ODS) tungsten alloys. ODS tungsten alloys with compositions of W-3Re, W-3Ta, and W-3Ta-1Re were synthesized, incorporating 0.2 wt.% Y_2O_3 to enhance dispersion strengthening. The alloy powders were processed via ball milling and spark plasma sintering (SPS). Ball milling was performed at 300 rpm for 10 hours, while SPS was conducted at 1800°C and 30 MPa for 10 minutes with a heating rate of 50°C/min. The mechanical properties of the fabricated alloys were assessed through tensile strength, hardness, and density measurements. Microstructural characterization, including grain morphology and phase distribution analysis, was conducted using electron backscatter diffraction (EBSD). The results indicate that the addition of Re and Ta significantly enhances the strength and toughness of tungsten alloys, while Y_2O_3 dispersion improves microstructural stability. These findings provide valuable insights into the development of advanced tungsten-based materials with superior mechanical properties for high-temperature applications. The results highlight the potential of these alloys as key materials for next-generation fusion reactors and other extreme-environment applications.

Tentative

P2-010

Influence of shielding materials on SDDR of the DEMO tokamak for ex-vessel areas at initial operation

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Tentative

The DEMO facility, as a prototype fusion reactor, represents a milestone toward commercial fusion energy. One of the key challenges in its design and operation is managing the Shutdown Dose Rate (SDDR) to ensure safe maintenance and decommissioning activities. Accurate estimation of the SDDR is important for assessing the conditions for personnel access at fusion facilities. This is especially important when examining the SDDR for components inside and outside the vessel at the start of operations to determine when the dose rate exceeds safe levels for human access.

The paper extends the study by testing additional irradiation scenarios, including deuterium-deuterium/tritium plasmas, and varying the shielding materials, at different locations inside and outside the vacuum chamber. The study will assess the influence of promising shielding materials, such as tungsten, SS304B4(B7), various mixtures of B_4C, and metal hydrates $Zr(Ti)H_2$ on radiation attenuation and SDDR reduction. Special attention will be given to the effectiveness of advanced shielding configurations in minimizing activation and ensuring safer operational conditions. The results will contribute to optimizing radiation safety protocols and maintenance strategies for future fusion reactors.

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 - EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

P2-011

Microstructural change in oxide dispersion strengthened medium entropy alloys under electron irradiation

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High/medium entropy alloys (H/MEAs) are gaining attention as potential materials for nuclear reactors due to their unique material properties and the possibility of showing superior irradiation resistance compared to conventional materials. In addition, the dispersion of oxides within the matrix is expected to enhance irradiation resistance by providing nanosized particle interfaces that serve as effective trap sites for point defects. In this study, oxide dispersion strengthened (ODS)-MEA, a novel material, was subjected to electron irradiation experiments to investigate the influence of interfaces on the formation of irradiation defects.

Alloy powders of FCC-type MEA $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$ and 316L with Y_2O_3 were mechanically alloyed (MA) respectively using a planetary ball mill. The MA powders were then sintered at 1000℃ using spark plasma sintering (SPS). MEA and 316L without Y_2O_3 were also prepared using a ball mill and SPS as references. Electron beam irradiation was carried out at 500°C up to 9 dpa, and microstructural changes were monitored through in-situ observations.

ODS-MEA exhibited greater swelling than MEA, suggesting nanosized particle interfaces in ODS-MEA suppressed the recombination of point defects. On the other hand, ODS-316L showed less swelling compared to 316L. Ranking the samples in descending order of swelling: ODS-MEA, MEA, 316L, and ODS-316L. These results indicate that the interfaces in ODS-316L acted as more effective trap sites than those in ODS-MEA. Additionally, a detailed analysis of the oxide particles and their interfaces in ODS-MEA and ODS-316L was conducted to further investigate the relationship between the interfacial structure and defect formation behavior.

Tentative

P2-013

Laser Powder Bed Fusion Printing of Tungsten on G91 Steel for Fusion Reactor Applications

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The development of robust plasma-facing components is essential for the viability of future fusion reactors, including ITER and DEMO. This study explores Laser Powder Bed Fusion (LPBF) printing of tungsten on G91 ferritic/martensitic steel to create a strong, thermally stable interface suitable for first-wall applications. By systematically varying processing parameters, we assess their influence on interfacial bonding, microstructural evolution, and mechanical performance. To comprehensively evaluate the tungsten-G91 interface, a combination of nanoindentation, thermal cycling, and advanced microstructural characterization via SEM, EBSD, and TEM is conducted. Additionally, innovative interface strength measurement techniques along with high-resolution digital image correlation (DIC) are employed to measure interface strength and assess local strain distribution during loading. These techniques provide localized mechanical insights critical for assessing defect formation and failure mechanisms under extreme fusion-relevant conditions. These results pave the way for additive manufacturing of plasma-facing components in fusion environments, offering new pathways for material design and qualification in advanced fusion reactor applications.

Tentative

P2-014

SINGLE ION REIRRADIATION OF DUAL ION IRRADIATED TEMPERED MARTENSITIC ALLOYS TO MEASURE POST INCUBATION VOID SWELLING RATES

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Void swelling (fv), which represents a significant challenge to the dimensional lifetime of ≈ 9Cr RAFM steel fusion structures, involves two key stages [1,2]: a) an incubation stage of stable growth of helium bubbles; and, b) beyond a critical bubble size, defining an incubation dose (dpai), due to growing voids, at swelling rates fv' up to ≈ 0.2 %/dpa [2]. The dpai depends on the He/dpa ratio (appm He/dpa). RAFM dpai are very large in low He/dpa (10 nm) connects to void facets. Notably, carbides present in the undamaged material are largely absent after irradiation, suggesting susceptibility to irradiation-induced shrinkage at peak damage.

References

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Tentative

P2-015

Effects of different microstructures of WCrY SMART material on the oxidation resistance and mechanical properties

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Self-passivating Metal Alloys with Reduced Thermo-oxidation (SMART) with a composition of W-11.4%-0.6%Y (in wt.%) is a plasma-facing candidate material for fusion power plants. In the case of an accident of loss-of-coolant and air ingress into the vacuum vessel, this material can prevent severe oxidation of tungsten and suppress volatilization of radioactive volatile tungsten oxides into the environment. Synthesized by mechanical alloying (MA) and field-assisted sintering technology (FAST), SMART material has shown to be the homogeneous, submicron-grained alloy with yttrium oxides distributed at grain boundaries - a standard microstructure studied in the previous years. In this work, the microstructural factors of SMART material, as well as their effects on oxidation resistance and mechanical properties were studied. Grain refinement, phase decomposition and oxide dispersion were implemented during the synthesis to study the effect of grain size, chemical homogeneity and oxide particles, respectively. Grain refinement was achieved by lowering sintering temperature to 1300 °C during FAST; phase decomposition was done by annealing the as-sintered homogeneous samples at 1000 °C; and oxide dispersion was implemented by replacing yttrium powders with yttria powders during MA. Compared to the standard microstructure, those approaches have demonstrated the altered oxidation behaviour in humid air at 1000 °C, enhanced flexural strength to varying degrees but no improved fracture toughness at elevated temperatures. The overview of the present studies will be given in the contribution.

Tentative

P2-016

Design of novel nanostructured tungsten and ferritic superalloys with high sink strength for fusion

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The 'BCC-superalloy' microstructure-template of a disordered matrix reinforced by ordered-intermetallic precipitates offers a potent design strategy for high temperature strength. Such microstructures are central to the success of fcc Ni-superalloys and are equally applicable to bcc-based systems, which offer advantages of increased melting point, lower cost, and/or lower density, but requirement careful alloy design to generate the target micro/nanostructures. Further, for nuclear applications, the generation of bulk nanostructures offers the design of high sink strength to mitigate irradiation damage.

This talk will present work from a EUROfusion Researcher Grant and UKRI Future Leaders Fellowship toward nanostructured tungsten and ferritic superalloys. This includes alloy design within W-Ti-Fe [1], W-Cr [2] and Fe-Al-V [3] systems, production, characterisation (SEM, TEM, S-XRD, APT) and mechanical properties. Further, early stage irradiation trials using ion irradiation at 300°C 1dpa to explore the irradiation damage tolerance, have indicated nil-apparent irradiation nanohardening (vs a EUROfer control showing substantial hardening) the mechanisms of which are explored using TEM, including loop formation and intermetallic disordering. These new alloys and microstructures open up new design approaches for high temperature capable irradiation resistant alloys that warrant further study to demonstrate the required balance of properties.

References:

1 - W-Ti-Fe https://doi.org/10.1016/j.apmt.2021.101014

2 - W-Cr https://doi.org/10.1016/j.apmt.2024.102430

3 - Fe-Al-V https://doi.org/10.1016/j.msea.2022.144031

Tentative

P2-017

Understanding refractory metals and alloys with machine-learned interatomic potentials

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Refractory metals and alloys in various forms, from pure tungsten to conventional and high-entropy alloys, are candidates for fusion reactor materials and need to be thoroughly understood and characterised both experimentally and computationally. In simulations, machine-learned interatomic potentials have now become valuable tools for accurate atomistic modelling. In recent years, we have focused on developing simple, fast, and accurate machine-learned interatomic potentials (tabGAP [1]) for metals and alloys of increasing numbers of elements, including high-entropy alloys. These potentials are now helping us understand fundamental properties of refractory alloys. Here I will present recent modelling results on various group 4-6 metals and tungsten-based alloys, including short-range ordering and segregation in alloys [2] and high-energy and high-dose radiation damage.

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Tentative

P2-018

Study on Microstructures and Helium Ion Irradiation Effects in WTaV Multi-Principal Element Alloys

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Tungsten and its alloys are regarded as highly promising plasma-facing materials for nuclear fusion reactors. However, the performance of tungsten-based materials can be significantly degraded under the extreme conditions of fusion reactors. In recent years, high-entropy alloys, also known as multi-component alloys, have garnered considerable attention for their unique properties and potential applications in extreme environments. This study investigates the microstructures and helium ion irradiation damage in WTaV refractory multi-component alloys. The effects of homogenization temperature on grain morphology and elemental distribution were systematically examined. The cast WTaV alloy exhibited a dendritic microstructure. After homogenization at 1400°C for 5 hours, the alloy displayed a transition from dendritic to equiaxed grain structures, accompanied by elemental segregation with V enriched in the inter-dendritic regions, while W and Ta enriched in the dendritic regions. At an irradiation temperature of 600°C, helium bubbles with sizes of 1-2 nm were observed in both pure W and homogenized WTaV alloys. As the temperature increased to 800°C, the size of helium bubbles grew, and their number density decreased, which is attributed to the migration and coalescence of bubbles. Notably, the homogenized WTaV alloys demonstrated superior resistance to helium-induced swelling compared to pure W. This enhanced performance is attributed to the formation of smaller helium bubbles at 600°C and the tendency of bubbles to decompose into smaller clusters at 800°C, highlighting the potential of WTaV alloys as advanced PFMs for fusion applications.

Tentative

P2-019

Advancements in pressure codes & standards for fusion power plants

Thomas P Davis[1], Emily R Lewis[2], Andrew Sowder[3]

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The design and construction of any fusion reactor should make use of appropriate codes and standards to provide quality assurance and control for the structural integrity and safety of these plants, such as pressure vessels which cover vacuum vessels, breeder blankets, and high-pressure cooling components. The first edition of American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (BPV) Section III Division 4 “Fusion Energy Devices” construction code and standard was released in June 2023. In 2024, Division 4 released its 5-year strategy to develop a construction code and standard for pressure vessels for fusion power plants. This marks the beginning of the activities, and one task is to write the materials qualification requirements to bring in “fusion-grade” structural materials within the Division for use in the future. This presentation has three objectives: 1) provide update on the development of the codes and standards for fusion power plants 2) outline a proposal for the materials qualification route for Division 4 which outlines the testing requirements, standards, assessment methodologies, environmental effects such as corrosion and irradiation, and 2) receive feedback from the fusion community on the approach. It is important that the code reflects the best practice and community needs/requests, and this talk is the first of many outreach activities Division 4 will be conducting over the years to ensure the construction code is fit for purpose for if/when fusion power plants become a reality.

Tentative

P2-020

Microstructure changing and irradiation hardening behavior of high-purity vanadium alloys

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Low-activation vanadium alloy is regarded as an attractive candidate structural material for the liquid lithium blanket systems in fusion reactors. The composition of V-4Cr-4Ti was selected as a reference according to the previously comprehensive research. In the V-Cr-Ti systems, alloying element titanium (Ti) not only scavenges the interstitial impurities (e.g., carbon, nitrogen, oxygen) by forming Ti(CON) precipitates to keep the impurity low level in the matrix and thus improve the ductility, but also can be dissolved into the matrix to provide solid-solution strengthening. On the other hand, it would produce long-term radioactive isotopes under fusion neutron irradiation environments, which is disadvantageous for the materials recycling and waste management. Therefore, it enables to further improve the low-activation characteristics of V-4Cr-4Ti alloy and reduce the recycling periods via refining the harmful high-activation elements and decreasing the Ti concentration. High-purity vanadium alloy containing much lower high-activation elements and interstitial impurities was developed. Even though precipitates can be formed with a lower Ti concentration, reducing Ti results into the strength degradation. To enhance the strength, increasing chromium (Cr) concentration is expected in V-Cr-Ti system. In the previous study, effects of Ti and Cr concentrations on microstructure and mechanical properties were systematically investigated. Comprising to the reference V-4Cr-4Ti alloy, alternative candidates with low-Ti and high-Cr levels were selected in the viewpoint of tensile strength. To further optimize the Ti and Cr concentrations in high-purity vanadium alloys, heavy ion irradiation at 300 - 500°C were employed, microstructural evolution and irradiation hardening behavior will be presented.

Tentative

P2-021

Optimizing Laves Phase in Refractory HEAs for Fusion Applications

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Refractory high entropy alloys (RHEAs) are promising materials for nuclear fusion, due to their combination of high-temperature strength, good irradiation resistance, and high melting point. High-density of finely-dispersed precipitates is useful in two ways: (1) impede dislocation motions to provide effective precipitation strengthening, and (2) creates abundant interfaces as strong sinks to protect from irradiation damage. The main goal of this work is to optimize the precipitation kinetics of the TiVNbCrTa RHEA so that nano-sized, thermally-stable precipitates can form within the alloy matrix. Novel small-scale mechanical tests as well as high-resolution microscopy will be performed on the precipitate-containing matrix to elucidate the influences of precipitation density, size, and morphology on mechanical behavior, under extreme nuclear fusion environment.

Tentative

P2-022

Effect of V/Ti Ratio, Reinforcement Phases/Oxide Particles on the Microstructure and Mechanical Properties of WMoVTi Alloys Synthesized by Mechanical Alloying

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This study investigates the combined effects of V and Ti content, reinforcement phase addition, and processing parameters on the microstructure and mechanical properties of WMoVTi alloys synthesized via mechanical alloying (MA). Yttria (Y_2O_3) was introduced as a dispersoid to enhance grain refinement and improve high-temperature mechanical performance. The influence of mechanical alloying on particle size reduction, homogenization, and phase evolution was systematically analyzed. Furthermore, the impact of sintering temperature on densification, grain growth, and microstructural stability was evaluated. Results revealed that increasing V/Ti ratio improved alloy ductility and strength due to solid-solution strengthening and phase transformation. The Y_2O_3 dispersoid significantly improved grain boundary stability and reduced grain coarsening during sintering. Optimized sintering conditions facilitated enhanced densification and mechanical properties, making these alloys promising for high-temperature structural applications in nuclear reactors.

Tentative

P2-023

A preliminary study on low-activation refractory high-entropy alloys with improved mechanical properties

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Low-activation refractory high-entropy alloys (LARHEAs) as structural materials have a great demand in nuclear reactors. However, currently developed LARHEAs generally have poor room temperature ductility, which greatly limiting their wide applications. In this study, two kinds of LARHEAs with chemical compositions of TiZrTaCr and TiZrTaV in equal atomic percentage were designed based on low neutron absorption cross section, low neutron activation, and mixing enthalpy parameters. The LARHEAs samples were prepared by high vacuum arc-melting techniques, and their phase structures and mechanical properties were studied. It was found that the TiZrTaCr LARHEA exhibited two BCC and one Cr_2Zr intermetallic compound phases, and it had a yield strength of ~ 1307 MPa, but showed very brittle; Meanwhile, the TiZrTaV LARHEA formed three body-center-cubic (BCC) solid solution phases, and it performed a high yield strength of ~ 1401 MPa with a room temperature compression ductility of ~ 13%. The Cr_2Zr intermetallic compound formation in the solid solution matrix could lead to the brittleness of the LARHEAs, which was mainly due to the large negative mixing enthalpy of the Cr element with the other elements in this alloy system. This preliminary study could propose a basic method for LARHEAs design to optimize the microstructures and improve the mechanical properties as structural materials.

Tentative

P2-024

Enhancing Passive Transport Performance in Microchannel-based Capillary Porous Systems for Liquid Metal Divertors: A Comprehensive Analysis

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Liquid metal divertor (LMD) is being developed as a robust alternative to conventional tungsten divertor. The capillary porous system (CPS), a microfluidic system featuring a fully wetted porous structure as the plasma-facing surface (PFS), has been receiving increasing attention in the field of divertors. In these systems, the porous structure provides capillary forces to drive the passive filling of the liquid metal to protect the solid substrate material and remove heat flux. In previous studies, numerous CPSs with complex structures have been proposed to enhance capillary-driven flow. However, the complex structures fail to strike an optimal balance between the dynamic competition of resistance and capillary pressure. Moreover, capillary flow mode in CPS is rarely reported. In this work, we investigated the capillary flow behavior of liquid metal and the heat transfer performance within the microchannel-based CPS, and delved deeply into the influencing factors and optimization directions of capillary flow in CPS. In the first part of the study, the flow-field distribution of liquid metal in the microchannels was examined. Subsequently, theoretical and numerical simulation analyses were carried out on the impacts of geometric structures and types of liquid metal on capillary flow, with a focus on the enhancement of capillary performance by corner flow. Finally, the capillary performance of the optimized microchannels was compared with that of traditional and non-traditional CPSs to demonstrate its superior passive transport performance. These results provide crucial guidance for the rational design of CPS with enhanced transport performance for applications in liquid metal divertors.

Tentative

P2-025

Parametrical study of the thermal response of liquid metal-W capillary porous systems exposed to high thermal loads

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The behaviour of liquid metals in capillary porous systems (CPS) will be critical for various engineering applications, including heat exhaust of plasma-facing components, avoiding the problems of degradation and permanent damage of the solid first wall and divertor components in future fusion reactors.

This study presents a parametrical investigation on the design and performance of liquid metal-tungsten CPS exposed to high thermal loads. Through transient thermal numerical simulations performed in ANSYS Parametrical Design Language (APDL), we explore the effects of geometry, porosity, pore size and saturation, using mixed thermal properties and random pore models. Our results show that these parameters significantly influence the surface temperature rise and temperature penetration range for both models. The numerical results have been compared with experimental measurements carried out in the OLMAT (Optimization of Liquid Metal Advanced Targets) High Heat Flux facility to validate these models and evaluate their accuracy and limitations. It is found that these models are in good agreement with experimental data during the thermal load exposure, but need to be improved to simulate radiative cooling phases. Additionally, SEM experiments have been carried out to characterize the CPS samples to develop a more-realistic geometrical model in CAD.

This work offers a validated framework for future modelling studies and provides new insights into optimizing porous materials for efficient heat transfer in liquid metal-based systems. These findings contribute to the design of enhanced performance solutions for plasma-facing components.

Tentative

P2-026

Deuterium permeation behavior through liquid LiPb supported by V-based alloy for hydrogen isotope extraction

Yuya Ando[1], Yuto Murata[1], Teppei Otsuka[1], Takuya Nagasaka[2]

[1]Kindai University, [2]National Institute for Fusion Science

Liquid LiPb (LiPb) has been proposed as one of the candidates for plasma-facing walls of liquid metals as a tritium breeding and a coolant material. It is necessary to recover breeding tritium from LiPb utilized hydrogen isotopes permeation phenomenon through a high hydrogen permeable metal. In this study, the hydrogen isotope permeation behavior through LiPb supported by a circular membrane of pure V and V-4Cr-4Ti alloy (NIFS-Heat2, NH2) was examined.

Deuterium permeation experiments through a LiPb/V system and a LiPb/NH2 system were examined repeatedly at temperatures ranging from 623 K to 873 K and pressure ranging from 0.67 kPa to 13 kPa.

The deuterium permeation flux through LiPb/V was high as same as that through only pure V membrane at 673 K. As the temperature increased sequentially from 723 K up to 873 K, the permeation flux did not change so much and showed less temperature dependence. Revisited measurement of the permeation flux at 673 K indicated 1 order magnitude smaller value than before. The cross-sectional observation with SEM-EDS revealed that an intermediate alloy layer with the thickness of 0.01 mm was formed near the interface between LiPb and V. The results suggested that deuterium permeation through the whole LiPb/V system was much influenced by that through the growing intermediate alloy layer.

In the presentation, deuterium permeation behavior through the LiPb/V and the LiPb/NH2 systems will be discussed and modeled with respect to a growth process of the intermediate alloy layer, and hydrogen diffusion and solution in the intermediate layer.

Invitation

P2-027

Characterization of alloy layers formed by reaction between a liquid LiPb and supporting metal substrates

Yuto Murata[1], Yuya Ando[1], Teppei Otsuka[1]

[1]KINDAI UNIVERSITY

Liquid lithium-lead (LiPb) is one of promising candidates for liquid blanket materials due to its tritium breeding capability and efficient cooling properties in advanced fusion reactors. However, its corrosive interaction with structural supporting metals, i.e., alloy formation at the interface, raises concerns regarding the thinning of these materials. Additionally, such alloy layers could impede hydrogen isotope permeation, thereby affecting tritium recovery from LiPb [1]. This study aims to elucidate a growth mechanism of alloy layers between LiPb and supporting metals through microstructural analyses and depth profiling of elements.

Liquid LiPb was supported by metal substrates (M = Ni, Fe, V) of 0.5 mm thickness (hereinafter called LiPb/M) at a constant temperature ranging from 673 K to 873 K in a deuterium gas atmosphere for varying durations. The microstructure and composition of the alloy layer around the interface were analyzed using SEM-EDS, XRD, and GD-OES.

For LiPb/V, GD-OES depth profiling of Li, Pb, and V revealed two distinct alloy layers: one formed by diffusional penetration of Li and Pb into V, and the other by the dissolution of V into LiPb. Some of these findings were supported by SEM-EDS line profiles near the interface.

In the presentation, the formation and growth mechanisms of the alloy layers between LiPb and various metal substrates will be discussed with respect to growth rates of the layers at different temperatures.

[1] Y. Ando, et al., SOFT-33, Marseille, France (2024)

Tentative

P2-028

The design of 3D-printing solid tungsten-liquid lithium combined divertor target plate and its performance investigation

[1]Zongbiao Ye, [1]Fujun Gou

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3D-printing tungsten-based porous matrix coupled with liquid lithium was designed and its intrinsic wicking capacity as well as its interaction with high-density plasma investigated through simulation and experiment. On account of present available fabrication technology, porous channels, inclination angle and arrangement were designed and optimized as finger-type unit based array with critical size of 150 μm. The simulative results showed that the porous matrix temperature, relevant thermal stress and wicking rate of lithium would be consistently within the safety limit of evaporation exhaustion and structural collapse under 10 MW/m^2 heat load without the consideration of vapor shielding effect and redepositing effect. The experimental results verified its excellent wicking ability, temperature locking function and redeposition behavior. The liquid lithium wetting stability and composite structure reliability were confirmed through real-time surface monitoring during plasma irradiation and post-mortem appearance examination. Furthermore, ≥10 individual target parameters were optimized and relevant performance with plasma was investigated. It reveals an obvious enhancement in permeability and capillary performance factor, and significant suppression in matrix temperature and thermal stress. These results reveal the application potential and feasibility of 3D-printing porous capillary structure in plasma-facing components and provide a reference for further liquid-solid combined target design.

Tentative

P2-030

Plasma-Material Interaction Science enabled by the new capabilities of MPEX

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The Material Plasma Exposure eXperiment (MPEX) [1] is a new linear plasma device being built at ORNL to address Plasma-Material Interactions (PMI) under fusion prototypic conditions. These include the ability to recreate the plasma conditions expected in the divertor of a toroidal magnetic confinement device. MPEX also will be able to expose targets in the near-surface magnetic field geometry prototypic for those devices. MPEX is unique in its ability to independently control electron and ion temperatures, as well as plasma density. This control allows distinguishing the physics processes related to electro-static (sheath) and magneto-static (Lorentz force) driven redeposition of erosion products. MPEX will be a superconducting magnetic linear device, allowing for very long pulse exposures of plasma facing components. This will allow, among many other aspects, the determination of hydrogen retention at high fluxes and high temperatures as well as the characterization of evolving surfaces due to redeposition and sub-surface damage. A novel surface analysis station will allow for in-depth characterization of the evolving surfaces in these long pulses by a set of state-of-the-art microscopy techniques. In addition, MPEX will allow the plasma exposure of hazardous (alkaline metal) and neutron irradiated materials.

References:

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Acknowledgments:

This work is supported by the DOE Office of Science, Office of Fusion Energy Science, under contract number DE-AC-5-00OR22725.

Tentative

P2-031

The effect of W precursors on in-situ induced Y2O3/W dispersive particles synergistic strengthened copper alloys

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In this study, Y_2O_3/W synergistic dispersion strengthened copper alloys (Cu-Y_2O_3/W) were fabricated by mechanical alloying (MA) and spark plasma sintering (SPS). Especially, Y_2O_3 and W nano-particles were introduced through in-situ reaction with three kinds of W precursors, including ammonium metatungstate (AMT), H_2WO_4, and WO_3. The effect of W precursors on mechanical and thermal properties was conducted. It was shown that W precursors act as a process control agent, inhibiting cold welding and refining Cu powders during ball milling. During the ball milling and sintering processes, W precursors reacted differently with Y. In addition to the in-situ generated Y_2O_3 and W, the WO_3 samples also produced Y-W-O compounds. These compounds absorbed oxygen from the copper matrix, thereby purifying the copper matrix. As a result, the copper matrix composites exhibited favorable mechanical properties, with a tensile strength of 330 MPa and a fracture elongation of 17.6%. Furthermore, they maintained a high thermal conductivity of 372 and 305 W/(m・K) at both room temperature and elevated temperatures, respectively. These findings substantiate the feasibility of incorporating W precursors, offering significant insight into the preparation of high-strength, high-conductivity copper-based alloys.

Tentative

P2-032

Plasma-Focused Ion Beam Analysis of WEST Monoblocks

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Four monoblocks from the CEA WEST tokamak in Cadarache, France, were sent to the United States for analysis. The monoblocks were examined via scanning electron microscopy (SEM)-energy dispersive X-ray spectroscopy (EDS) and X-ray photoelectron spectroscopy (XPS), which were reported previously (https://doi.org/10.1016/j.jnucmat.2024.155525). These results, particularly XPS, found regions of mixed B-W-O-C-N deposited onto the monoblocks, especially in the regions of thick deposition on Monoblock #3 and #9 (MB3 and MB9). XPS is excellent for elemental quantification and chemical bonding measurements, but is not suitable for measuring thickness, morphology, small-scale inhomogeneities, and arrangement of such deposits. Therefore, this work used plasma-focused ion beam (PFIB) combined with SEM-EDS to cross-section areas strategically selected based upon XPS to measure the thickness and morphology of the deposits. For instance, MB9 from the “thick deposits” region was analyzed by milling large (≈50×50 µm) flat faces with the Xe-ion beam in the PFIB and then performing X-ray spectrum imaging on the polished face via SEM-EDS. XPS maximum sputtering depths were estimated to be <200 nm, and found deposits of predominantly B+W, and B+C (consistent with tungsten borides and carbides from XPS), with some O+C+N. PFIB+SEM-EDS found regions of 10 µm or more thickness, consisting of B-rich W-O-C-N deposits that took complex folded morphologies, and sometimes intermixed with what appeared to be metallic tungsten. This talk will review the PFIB results on the four monoblocks and compare to XPS.

Tentative

P2-033

Investigation of the Thermal Stability of Potassium-doped Tungsten fabricated by the Spark Plasma Sintering method

Hyoung Chan KIM[1], Eunnam Bang[1], Sungjin Kwon[1], Jeongsek Kim[2], Heung Nam Han[2]

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Tungsten (W) is the primary plasma facing material for divertor and the first wall of ITER. However, for the divertor of next-generation Fusion reactors, tungsten-based materials with enhanced performance as armor materials are required. Among the key material properties of tungsten, stability against high heat flux and resistance to recrystallization at high temperatures are critical for the stable operation of divertor under higher fusion power conditions. One limitation of the currently available industrial W plates produced through severe deformation process is the pronounced degradation of their mechanical properties upon recrystallization at high temperatures (> 1250 °C).

In this study, we investigated the thermal stability of potassium (K)-doped tungsten fabricated using the spark plasma sintering (SPS) method and compared it with the behavior of commercial W plate produced by hot rolling. While high level of dislocation density was not generated in the SPS process, fine-grained (~3.2 micro m) microstructure and high strength (Vickers hardness ~460) were achieved along with the improved ductile-to-brittle transition behavior by small amount of K content (0.0033 wt%). In order to manufacture mockups for high heat flux testing, scaled-up K-doped W samples (45 mm in diameter) were successfully fabricated while maintaining homogeneous material properties. Analysis of the heat-treated specimens (1200 °C, 1450 °C) using electron backscatter diffraction (EBSD) and hardness measurements showed almost no change in K-doped W, whereas significant hardness reduction accompanied by grain growth was observed in the commercial W plate. The results of high heat flux testing for the fabricated mockups are presented and compared.

Tentative

P2-034

Deuterium permeation and retention behaviors in dispersion-strengthened W

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Tungsten (W) is the primary candidate for plasma-facing materials (PFMs) in fusion devices. However, the brittleness, poor machinability, and low strength at high temperatures limit its performance. To overcome these limitations, dispersion-strengthened W are W-based materials strengthened with small amounts of dispersed second-phase particles. Dispersion-strengthened W exhibits high strength/ductility, low ductile-to-brittle transition temperature, and excellent thermal shock resistance, making it a possible candidate PFM. However, the effects of dispersed particle addition on deuterium retention/permeation behaviors have not yet been thoroughly investigated.

In this study, deuterium permeability and diffusivity in various dispersion-strengthened W (strengthened with ZrC, La_2O_3, Y_2O_3, ZrO_2) were systematically evaluated using gas-driven permeation (GDP) experiments. Thermal desorption spectroscopy (TDS) was employed to quantify deuterium retention and trapping characteristics. The effects of new defects introduced by particle addition on deuterium transport and trapping were discussed in combination with microscopic characterization. In addition, combined GDP and TDS methods with plasma irradiation experiments, the effects of He-induced fuzz on deuterium permeation and retention were studied. This study provides the first experimental evidence of fuzz-induced deuterium permeation enhancement. Detailed results will be presented at this conference.

Tentative

P2-035

$(CrVW)_{1-x}$ $(TiTa)_x$ as thermal barrier for nuclear fusion

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High-entropy alloys have been investigated for nuclear fusion applications, particularly for the divertor monoblock. Due to the extreme operating temperatures inside fusion reactors, the materials used in the monoblocks must withstand harsh conditions. Tungsten was chosen as the plasma-facing material, while a CuCrZr alloy will be used in the cooling pipes. However, the gradient between the working temperatures of these materials requires the use of a thermal barrier interlayer between them to guarantee a continuous operation. The high entropy alloys has demonstrated exceptional combination of thermal, mechanical and irradiation resistance properties which are essential for thermal barrier interlayer in these types of applications. In this work, $(CrVW)_{(1-x)}(TaTi)_x$ alloy, where (x=10 at.% and 20 at.%), were produced by mechanical alloying, followed by spark plasma sintering. Heat treatment and differential thermal analysis were performed in the samples consolidated by spark plasma sintering, accessing their thermal stability up to 1250 °C. All samples were characterized by scanning electron microscopy and by X-ray diffraction. The $(CrVW)_{90}(TaTi)_{10}$ sample consolidated by spark plasma sintering revealed a bcc type structure in agreement with the thermodynamic calculations, and with a peak intensity inversion after annealing treatment. The $(CrVW)_{80}(TaTi)_{20}$ sample consolidated by spark plasma sintering show a dominant bcc type structure, also in line with the thermodynamic calculations but with a minor $CTi_{0,55}W_{0,45}$ phase that corresponds to a fcc type structure, which can indicate an enhancement in mechanical and tribological properties. After annealing, this sample demonstrated a dominant C14 Laves phase, which is known to improve the alloy's resistivity.

Tentative

P2-036

Brittle Fracture Assessment for Tungsten

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Tungsten materials are promising candidates for armour applications in future nuclear fusion reactors due to their good thermo-physical properties. A particular challenge in the design of tungsten components is the inherent brittle behaviour in the temperature range below the Ductile to Brittle Transition Temperature (DBTT) of tungsten. In addition to common deterministic design rules, suitable probabilistic design rules are required for specific failure modes, like brittle fracture, to include details about failure probability. Aiming these rules, it is decisive to get database for fracture toughness as a function of temperature and probability. To build this database, results from fracture mechanics and four-point bending tests are required. Four-point bending tests support the determination of the Weibull parameters needed for the prediction of failure probability, while fracture mechanics tests determine the fracture toughness and its scatter. Together with dedicated Finite Element (FE) Analyses, the probabilistic aspect can be considered for each temperature of interest. In these FE-analyses, a probabilistic cohesive zone model (CZM) is utilized to simulate the fracture process observed in fracture mechanical tests including the Weibull parameters. Thereby the focus is on the development, introduction and usage of the probabilistic CZM within FE models.

Results from (modified) Weibull analysis between room temperature and 400°C and fracture toughness test results for several orientations as well as an approach for the link to the CZM will be presented and discussed. In addition, an overall scheme for Brittle Fracture Assessment (BFA) of tungsten and tungsten alloy components will be introduced.

Tentative

P2-037

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

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Tungsten is the primary candidate materials for plasma-facing components (PFCs), but pure tungsten has drawbacks such as room-temperature brittleness, recrystallization brittleness and irradiation-induced embrittlement. To address these challenges, we investigated nano-carbide dispersion and grain boundary/phase interface control methods, and developed high performance nanostructured W-ZrC alloy plates optimized for industrial production. The W-0.5ZrC alloy exhibits high strength, low DBTT (~100 °C), high recrystallization temperature (~1600 °C) and enhanced neutron irradiation resistance. After 1 dpa/600 °C neutron irradiation, the W-0.5ZrC alloy exhibits a DBTT of 300-500°C, much lower than that of ITER-grade W (>600°C). These favorable properties make W-ZrC alloy a promising PFM. W-ZrC/Cu mock-ups were also prepared and the behaviors under high heat flux (10-20 MW/m^2 for 1000 cycles) were studied.

Noting that tungsten alloys generally suffer from severe neutron irradiation embrittlement at low temperatures. The upper limit of ITER-grade CuCrZr alloy (~500°C) is much lower than the lower operating temperature limit of W (~800°C). This critical mismatch underscores the need to develop advanced heat sink materials with high thermal stability. Recently, nanostructured (NS) Cu-W alloys reinforced by nanoscale W particles (~7.6nm) with high strength (~700MPa), high thermal conductivity and excellent high-temperature stability (stable up to 800 °C) were prepared by sol-gel method and spark plasma sintering. Besides, numerous Cu/W interfaces can act as sinks for irradiation-induced defects, thereby enhancing irradiation resistance. The high thermal stability of NS Cu-W is conducive to expanding the operating temperature range and reducing the operating temperature gap with W-based PFMs.

Invitation

P2-038

Simulation of net erosion effects on tungsten nano-tendril bundles growth under He/Ne plasma irradiation

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Island-like nano-tendril bundles (NTBs) composed of fuzzy nanofibers can form on a tungsten (W) surface under simultaneous sputtering by helium (He) as well as seeded impurity like neon (Ne) and argon (Ar). In future fusion devices like ITER, a high-energy ion flux, which consists of He and seeded impurity Ne/Ar may facilitate the growth of NTBs through the sputtering and deposition of W. Although the exact growth mechanism of NTBs remains unclear, the formation conditions has already been identified, which requires a net sputtering yield (Ynet) of~10^{-2}-10^{-3}, a W surface temperature range of 900-1600 K, and the presence of a pre-existing fuzzy structure on the W surface.

A series of experiments have observed that changes in Ynet can lead to significant structural variations in NTBs, which indicated that the sputtering-deposition process plays an important role in the NTBs growth. Dedicated modeling has been conducted to explore the impacts of Ynet on NTB growth under He/Ne plasma irradiation using the three-dimensional kinetic Monte Carlo code SURO-FUZZ. The surface morphology of NTBs with different Ynet has been simulated by SURO-FUZZ. It is found that cone-like NTBs become more agglomerated as the Ynet increases. Additionally, the changes in NTB morphology under different incident angles have been investigated in detail.

Tentative

P2-039

The retention and release of hydrogen isotopes from boron coated tungsten

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Boronisation is a vital task for improving tungsten as a first wall material. Boron layers shield tungsten from irradiation damage and prevent the tungsten dust from mixing with the plasma. However, the retention and release behavior of hydrogen isotopes in the boron layer is not fully understood. In this work, the hydrogen isotopes retention properties for boron coated tungsten samples provided by CFS (Commonwealth Fusion Systems) have been investigated using DELPHI (Device for Exposure to Low-energy Plasma of Hydrogen Isotopes) and TDS (Thermal Desorption Spectrometry) located at UKAEA.

The TDS analyses showed that the boron coatings contained contaminants from the deposition process and from being produced ex-situ. The boron-coated samples were heat and argon plasma treated to remove moisture, hydrogen and oxide layers. Pre-treated samples were then exposed to deuterium plasma while varying the exposure time, temperature, and ion energy. TDS was used to obtain the retained deuterium content in the boron coated tungsten samples post plasma exposure. The results of deuterium retention and release from the boron coated tungsten with varying pre-treatments and plasma loading conditions will be presented and discussed.

Tentative

P2-040

Geometric properties relationships for electron beam melted tungsten alloys

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Tungsten and tungsten alloys are the current frontrunner as plasma facing materials in fusion reactors. Some of tungsten alloy's major advantageous properties include a high melting temperature, high thermal conductivity, low vapor pressure, low tritium retention, and a high sputtering resistance. However, being a brittle material at room temperature with a high ductile brittle transition temperature, tungsten alloy manufacturability remains sub-par. However, to understand the processing method impact the microstructure and related mechanical properties, W3Re walls with 3, 6, and 10mm wall thicknesses were manufactured using the same processing parameters. A systematic study linking the microstructure, and mechanical properties was conducted to inform future AM of tungsten alloys. The objective is to be able to tailor a microstructure regarding the mechanical needs of the components, by adjusting printing parameters. The study provides a direct impact from the geometry of printed components from samples extracted from different locations and wall thicknesses. A clear distinction of mechanical properties between the top and bottom part of the wall can be seen with repeatable test results. The ultimate tensile strength at the outside is on average 30 MPa higher for samples from the top half in comparison with samples from the bottom. Samples extracted closer to the center of the walls exhibited significantly higher ductility with strain of up to 50 %, while the UTS and YS were slightly lower. The number of high and low angle grain boundaries at the outside combined with a more isotopic texture leads to increased strength.

Tentative

P2-041

Effects of helium irradiation damage on deuterium plasma-driven permeation through tungsten coated RAFM steel

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In future fusion reactors, the first wall will be irradiated with the hydrogen fuel (deuterium (D) and tritium (T)) and helium (He) with low energies. In this work, the effects of He plasma irradiation on D permeation through tungsten (W) coated reduced activation ferritic/martensitic (RAFM) steel were systematically investigated by performing plasma-driven permeation (PDP) measurements in the temperature range of 523-833 K. The W coatings were prepared by magnetron sputtering. He plasma irradiation with different fluences were carried out in a linear plasma device. The results showed that as He irradiation fluence increased, the surface damage of W coatings intensified, resulted in a high steady-state D-PDP flux and effective diffusion coefficient. The effect of incident ion energy on D-PDP was also influenced by He pre-irradiation. After He irradiation with a fluence of 6.0×10^{24} He m^{-2}, the D-PDP flux was found to be decreased with increasing incident ion energy despite the fact that the sample temperature was gradually increasing. The He pre-damage affected the overall D transport by re-balancing the bulk diffusion and surface recombination processes, therefore determining the PDP flux. D-PDP through W + RAFM under simultaneous D + He mixture plasma exposure was studied as well. The results indicated that the presence of He reduced the ionization rate of D plasma and decreased D-PDP flux. D-PDP regime under the mixture plasma exposure was verified by calculating the transport parameter. It was found that D-PDP regime was not changed by He introduction, which remained in the RD regime.

Tentative

P2-042

Surface damage and deuterium retention in tungsten under high-flux detached recombining linear plasmas

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In the future deuterium-tritium fusion phase, it is essential to achieve partial detachment of divertor plasmas in order to maintain an acceptable level of particle and heat loads. There should be a clear, evidence-based prediction of D/T retention within plasma-facing materials (PFMs) throughout their operational lifespan, utilizing existing experimental or numerical methodologies. A compact linear plasma device has been developed based on cascaded arc which can provide stable detached deuterium plasmas with flux up to 10^24 $m^{-2}s^{-1}$, heat load up to 15 MWm^{-2}. The spectroscopic measurements shown high-n transitions from electron-impact recombination process and flux decrease with increasing impurity injection. In partially detached plasmas, the ions are not solely D+ or D_2^+, but instead a substantial amount of D/D_3^+ at elevated pressures. When impurities like nitrogen are introduced, compound ions like ND_4^+/N_2D^+ can form. The surface damage is inhomogeneous in space. The blister preferred to exist in medium region and also the sputtering and nano-blistering morphology. They are both grain orientation dependent. The nucleation and growth of μm-blisters are deeper and saturated due to micro-cracking of blister dome. The deuterium retention and blister distribution varies with bias voltage, showing an inflection point at -40V. The TDS revealed that the retention was dominated by blister-induced defects. It changed from trapping mode to diffusion mode at elevated temperature. The divertor detachment (0V bias) is beneficial to managing tritium inventory, ensuring that deuterium does not diffuse too deeply into the material and accumulate in deeper traps.

Invitation

P2-043

Low-ion energy sputtering yields of high-Z refractory coatings for helicon plasma source window material

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Radio-frequency (RF) sheath-induced sputtering of the helicon source window has led to impurity generation in Proto-MPEX1, a linear plasma device (LPD) which was a prototype for developing the physics basis for MPEX. MPEX will address the plasma-material interactions relevant to the divertor of future fusion reactors by providing the desired plasma conditions2. To reduce the impurity generation in MPEX operations, we have proposed using a Faraday screen to lower the rectified sheath voltage drop below 300 V and to minimize erosion by applying high-Z refractory coatings (such as Ta_2O_5 and HfO_2) on the Si3N4 window inner surface3. The proposed coatings have higher sputtering thresholds than the window substrate and are hence hypothesized to reduce the window erosion. In this work, sputtering yields of these coatings were measured in PISCES-A LPD4 (deuterium ion fluence ~ 10^{26} m^{-2}) under different RF bias conditions ($\leq$ 200 VRF(peak)). The effective ion energy under the given RF bias conditions was equivalent to the zero-to-peak RF voltage4. Post-surface composition analysis shows no surface enrichment of the high-Z components, with re-deposition of impurities originating from the sample holder. The higher sputtering yields are indicative of sputtering caused by the spread in the ion energies and the impurities in the plasma.

Tentative

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P2-044

Industrial Scale Up for Tungsten Fiber-Reinforced Tungsten Composites for Plasma-Facing Applications

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Tungsten fiber-reinforced tungsten composites (Wf/W) present a promising solution to mitigate the intrinsic brittleness of tungsten (W), a leading candidate for plasma-facing components (PFMs) in fusion reactors. These composites leverage extrinsic toughening mechanisms to enhance fracture toughness and pseudo-ductility, crucial for withstanding extreme thermal and mechanical stresses in the divertor region.

This work highlights recent advancements in Wf/W manufacturing, focusing on scalable techniques for component development. Using powder metallurgy (PM) and Chemical Vapor Deposition/Infiltration (CVD/CVI), These composites exhibit significant improvements in fracture toughness via fiber bridging, crack deflection, and fiber plastic deformation.

Notably, CVI with a three-dimensional temperature gradient has enabled efficient processing of textile-based fiber architectures, overcoming limitations of Spark Plasma Sintering (SPS) and conventional CVD.

A recent breakthrough in long fiber Wf/W (L-Wf/W) production demonstrates enhanced fracture toughness while retaining application-relevant properties. Industrialization efforts focus on optimizing tungsten powder selection and yttria interface thickness to balance densification and fiber integrity. Optimized samples achieve densities of 93% and exhibit superior pseudo-ductility. High-temperature tensile tests (up to 1000°C) and in-situ crack propagation studies confirm improved crack resistance. Moreover, Wf/W composites maintain comparable sputtering resistance and deuterium retention to baseline tungsten under simulated fusion conditions.

These advancements mark significant progress in eveloping robust, scalable, and damage-tolerant PFMs for future fusion reactors with the perspective for industrial scale production.

Invitation

P2-045

Fabrication and Characterization of Potassium-Doped Bulk Tungsten with Superior High-Temperature Microstructural Stability and Low-Temperature Toughness for Future Fusion Devices

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In future fusion devices, pure tungsten as plasma-facing material will degrade under high-flux steady-state thermal loads, necessitating dispersion strengthening via second-phase incorporation. Potassium-doped tungsten (AKS-W), with low impurity concentrations (50-100 ppmw K) and superior high-temperature strength above 1700°C, outperforms rare-earth oxide or carbide-doped alternatives. Its strengthening mechanism relies on the size and distribution of potassium bubbles, optimized in incandescent filaments through near-100% plastic deformation. However, fabricating bulk AKS-W components (thickness ≥12 mm) with comparable microstructure and properties for ITER-like divertor applications presents significant challenges, including the production of homogeneous fine-grained as-sintered ingots, high-strain plastic deformation without introducing defects, and the development of complementary thermo-mechanical treatments.

Enhanced sintering kinetics and efficient impurity removal were achieved by strategically combining AKS-W powders with varied K contents and particle size distributions, successfully producing rod-shaped (diameter ≥70 mm) and plate-shaped (thickness ≥43 mm) as-sintered ingots with 80?100 ppmw K and homogeneous fine-grained microstructure. Through systematic optimization of forging parameters, deformation ratios, and post-processing heat treatments, the developed AKS-W plates (thickness ≥12 mm) exhibit exceptional properties: a low ductile-to-brittle transition temperature (100-200°C), outstanding high-temperature microstructural stability with a recrystallization temperature exceeding 1700°C, and high thermal conductivity ≥165 $W \cdot m^{-1} \cdot K^{-1}$ at room temperature.

Furthermore, ITER-like divertor mono-blocks fabricated from developed bulk AKS-W in this study showed no macro cracks or significant plastic deformation after 1000 thermal fatigue cycles under specific conditions of 20 MW/m^2 (15 s on/15 s off), demonstrating its potential for fusion applications.

Tentative

P2-047

The influence of Ne admixture on erosion and fuzz formation of W exposed to D+He+Ne plasma in a PSI-2 linear plasma device

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Tungsten (W) is currently a main candidate as a divertor plasma facing material in fusion devices. When exposed to a plasma containing He ions, W can exhibit fuzz formation. Additionally, in the future fusion devices W will be exposed to excessive heat fluxes during off normal events such as ELM's, leading to significant erosion. One of the options to mitigate excessive power loads is by divertor impurity plasma seeding, using Ne puffing, to dissipate the energy.

Polished W samples were exposed to simultaneous D + 5%He + Ne plasma and laser loading used to mimic ELM-like loads. Influence of different Ne content (0, 1 and 5%) on erosion and fuzz formation was investigated. During each experiment two samples were exposed to mixed plasma, one of which also to laser loading. Tests were performed with an absorbed power density of 100 MW/m2, repetition frequency of 200 Hz and pulse duration of 0.5 ms. Incident ion energy was in the range of ~ 170 eV and the total ion fluence was 1 × 1025 m2.

Before and after the PSI-2 exposure samples were characterized by means of electron microscopy techniques. Sample exposed only to plasma containing 1% Ne exhibited early stage of fuzz formation (with the thickness of ~ 50 nm) as well as erosion of about 160 nm. For the sample exposed to simultaneous plasma and laser loading, outside the laser spot, no fuzz was observed, and the erosion was above 350 nm. In the laser spot extensive surface roughening was observed

Invitation

P2-048

A Study on Ductile-to-Brittle Transition Temperature of Pure and Potassium-Doped Tungsten

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Tungsten has been considered as the most promising candidate for plasma-facing materials. Pure tungsten, however, exhibits a high ductile-to-brittle transition temperature (DBTT) and serious brittleness after recrystallization, which limits its applications as a structure material. We investigated the DBTT of polycrystalline pure tungsten with various microstructures prepared by cold-rolling and post-annealing, and potassium-doped tungsten. The practical fracture analysis diagram was suggested to predict the DBTT with non-destructive approaches. Temperature dependences of the local effective yield and fracture stresses, which were essential prerequisites to construct the diagram, were assessed by systematic techniques combined with nanoindentation, dislocation dynamics (DD) and molecular dynamics (MD) simulations: (i) temperature dependence of local effective yield stress is obtained by thermally activated pop-in phenomena during nanoindentation, which represents the resistance to the onset of micro yielding occurring in small volume. (ii) MD simulations provide the temperature dependence of brittle fracture stress, where the atomic bonds break due to intergranular or intragranular fracture. The diagrams were completed through a statistical process of the data and were practically used to determine the DBTT of each specimen. The DBTT of each specimen was then compared with the direct observation of failure responses in mechanical tests. We confirmed that K-doped tungsten has a lower DBTT than pure tungsten and elucidated its mechanism using the DD simulations. Furthermore, we confirmed the good agreement between the DBTT from the diagram and the direct experimental measurements. This indicates that the method proposed here has the potential for examining the DBTT of brittle metallic materials.

Tentative

P2-049

Microscopic and macroscopic investigation of He bubbles formation on tungsten via in situ spectroscopic ellipsometry

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In tokamaks, the tungsten (W) divertor and first wall endure intense heat and ion fluxes, primarily from hydrogen isotopes and helium (He) ions. The interaction of charged particles with W can cause material modifications, from microscopic lattice defects to mesoscopic damage such as near-surface (<50 nm) bubbles. These structures, observed on W exposed to low-energy (<100 eV) He ions at high temperatures (<1273 K), can alter the material's optical properties by increasing surface roughness.

This work investigates the surface morphology and optical properties of W exposed to He plasma through in situ spectroscopic ellipsometry, which measures changes in polarized light (400-1000 nm) upon reflection. The setup features a radio-frequency plasma chamber where real-time ellipsometry data are gathered during He plasma exposure. He ion flux and fluence are $2\times10^{19} m^{-2}s^{-1}$ and 5×10^{23} m^{-2}, respectively, with ion energies around 80 eV.

We conduct plasma exposures by varying parameters such as sample temperature, ion flux, and initial surface state. Ex situ techniques (SEM, AFM) are used to characterize He bubble size and density based on temperature and grain orientation. These results are combined with in situ data to model temporal changes in porosity and surface roughness using an effective-medium layer that accounts for He bubble volume fractions.

Finally, the findings are validated through the FEniCS model describing He bubble growth and transport, offering insight into surface evolution during He plasma exposure.

Tentative

P2-050

A Study of Ultra-strong Tungsten-based Multiphase High Entropy Alloys

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As a method for strengthening tungsten (W), the new material design of high entropy alloys (HEAs) has attracted attention. Inspired by the previous design (W-Ta system) of low activation-W system-high entropy alloys (L-W-HEAs) and the research on various alloy design for high strength HEAs, we have carried out research and development of further strengthened L-W-HEAs. Our previous study revealed that iron (Fe) strongly enhances the hardness of W-HEAs. According to this result and the consideration of important factors in HEA design, such as mixing entropy, difference in electronegativity, difference in atomic radius, and valence electron concentration, we have studied a new ultra-strong L-W-HEAs (W-Fe system).

A new process combining multi-stage casting and hot isostatic pressing is adopted to manufacture the new alloys; W-HEA has more than four phases, including two BCC crystal structures, and some nanocrystals smaller than 1 μm. Hardness was mostly in the range of about 1500-2000 HV, but there were also regions exceeding 2500 HV, with a maximum of 3600 HV. This hardness was comparable to the hardest HEA material to date (nitrided HEA), and the maximum hardness exceeded the maximum values of HEA materials and SiC. Recently, the interface hardening theory (Raabe theory), in which the formation of duplex phase leads higher hardness, was reported in the field of Fe-Mn system HEAs. The present results show that this theory is applicable to W-based alloys, and that the multi-phase crystal structure leads in further super-hardening of the present alloys. Further investigation, including HIP, will be conducted.

Tentative

P2-051

Deuterium retention and thermal stability on boron surfaces

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Wall conditioning techniques using low-Z materials are used in fusion devices to trap impurities and reduce fuel recycling. Boronization is a common wall conditioning technique that has been studied in many fusion devices. Boron coating of plasma-facing components (PFCs) has been achieved by glow discharge [1] or powder injection [2]. Fuel retention in boronized PFCs must be studied for tritium inventory monitoring. We have evaluated the time-dependent retention of deuterium (D), as a proxy for tritium, on boron surfaces. In these experiments, the surfaces of solid boron powders and films at room temperature (297 K) were exposed to D2 gas or irradiated by 170 eV D^+ ions using a tectra ECR Plasma Source mounted on an ultra-high vacuum (UHV) analysis chamber. The effect of the boron substrate temperature during irradiation and during waiting intervals post-irradiation was also evaluated, ranging from 297 K to 600 K. Following waiting intervals after D exposure, temperature programmed desorption (TPD) mass spectrometry - also known as thermal desorption spectroscopy (TDS) - was used to evaluate the amount and thermal stability of the deuterium retained in the boron samples. TPD measurements were carried out at different time intervals after the deuterium exposure, ranging between 0 hours and 3 days, in order to determine the time-dependent retention of deuterium in the boron substrate in UHV.

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Invitation

P2-052

Deuterium Erosion and Retention Behavior of Ultra-high Temperature Ceramic TiB_2

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Ultra-high temperature ceramics (UHTCs) have been identified by the U.S. fusion community as promising candidates for plasma-facing component applications. UHTCs' appeal for fusion applications stems from their refractory properties, high temperature strength and thermal conductivities, as well as reduced radiative plasma loss from sputtered mid- and low-Z atoms compared to that of high-Z metals. Despite promising fundamental material properties, UHTCs are relatively untested under fusion-relevant plasma loading conditions. This work aims to address uncertainties in UHTC plasma-material interaction behavior by characterizing deuterium-plasma induced erosion and retention in UHTC titanium diboride (TiB_2). TiB_2 samples were exposed to low-incident ion, Ei, energy deuterium plasma (Ei ~20 eV, below the physical sputtering thresholds for boron and titanium) in the PISCES-RF linear plasma device. Sample temperatures spanned 70-825°C to quantify and assess temperature-dependent erosion processes in the materials. Microbalance mass loss measurements were performed on the 25.4 mm diameter samples. The 70°C plasma exposure induced 0.11±0.05 mg mass loss, the 180-610°C plasma exposures induced 0.91-1.00±0.05 mg mass losses, and the 720-825°C plasma exposures induced 1.70-1.73±0.05 mg mass losses under identical plasma fluences, indicating that erosion processes are temperature dependent. Mass loss measurements correlate to sputtering yields of 0.7-11.8E-4 atoms/ion assuming stoichiometric, TiB2, sputtering and 1.5-25.4E-4 atoms/ion assuming only boron sputtering. We will explore possible explanations for temperature dependent erosion behavior (e.g., ad-atom, oxidation, and/or deuterium-chemical erosion) and thermal desorption spectroscopy will be performed to measure deuterium retention and desorption temperatures for TiB2.

Invitation

P2-054

Thermal stability of tungsten fiber-reinforced tungsten composites fabricated by powder metallurgy

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Tungsten-based materials are considered as armor of plasma-facing components for future fusion reactors. To overcome the intrinsic brittleness of tungsten, tungsten fiber-reinforced tungsten composites (Wf/W) have been developed and fabricated by powder metallurgy. Two types of Wf/W composites, with either continuous aligned potassium-doped tungsten wires in a dense matrix or randomly oriented short fibers in a porous matrix produced by field assisted sintering (FAST) are investigated. Specimens are annealed at 1450 °C for different amounts of time up to two weeks to assess their thermal stability. Scanning electron microscopy and electron backscatter diffraction reveal major changes in the microstructure. After 4 hours of annealing, in both composites, initiation of recrystallization in the fibers and simultaneous grain growth in the matrix is observed. Recrystallization commences at the outskirts of the fibers causing formation of a rim of small, recrystallized grains. Longer annealing times increase the rim of recrystallized grains inwards into the fiber. After 3 days of annealing, all fibers are completely recrystallized, and the matrix is coarsened significantly by grain growth. While the short fibers can still be identified in the porous matrix after one week of annealing, matrix and continuous fibers cannot be distinguished any longer in the dense matrix. Short fibers with large, recrystallized grains can still be distinguished after 2 weeks of annealing, while the porous matrix disintegrates by particle coarsening.

Tentative

P2-055

Development of high-entropy-type REBCO superconductor with multiple sites substitution (Y, Gd, Dy, Yb)$_{0.25}$Ba$_{2-x}$Sr$_x$Cu$_3$O$_7$-δ

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We developed high-entropy-type (HE-Type) REBa$_{2-x}$Sr$_x$Cu$_3$O$_7$-δ (RE: rare-earth, REBCO) superconductors by incorporating the high-entropy alloy (HEA) concept into specific crystal sites [1-5]. Inspired by the irradiation resistance of HEA [6], we investigated irradiation resistance of superconducting properties for HE-type REBCO by He ion irradiation and found that they kept a high superconducting transition temperature [5]. This suggests the enhancement of irradiation resistance in HE-type REBCO, possibly due to the increased migration barriers from atomic distortion and disorder. Our findings indicate that HE-type REBCO is a promising candidate for superconducting magnets in next-generation nuclear fusion reactors. These materials could improve magnet performance and longevity in extreme environments, advancing fusion reactor technology.In this study, we introduced simultaneous substitution of Sr for Ba site of HE-type REBCO. HE-type REBa$_{2-x}$Sr$_x$Cu$_3$O$_7$-δ thin films were prepared using a pulsed laser deposition (PLD) method, and the superconducting properties were evaluated before and after Ar ion irradiation. As a result, the HE-type samples retained their superconducting state even after high irradiation damage where the superconductivity disappeared in YBCO. We will discuss the multiple-site substitution effect on the superconducting properties and the ion irradiation.

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Tentative

P2-056

Evaluation of radiation resistance by in-situ observation of electron irradiation of $REBa_2Cu_3O_{7-\delta}$ with multiple element substitution

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We are focusing on copper oxide superconductors for superconducting magnets in fusion reactors. The environment inside a fusion reactor is harsh for materials, with 14 MeV neutrons flying around. The superconducting magnets are also damaged by neutrons which penetrate through the blanket and degrade their superconducting properties. We are trying to incorporate the concept of “high-entropy alloys” into superconductors to give them radiation resistance. In this study, we have made high-entropy copper oxide superconductors ($REBa_2Cu_3O_{7-\delta}$) by substituting multiple rare-earth (RE) elements for the RE sites. The samples are single-crystal whiskers, and are named with the number of substituted elements for RE (RE1, RE2, RE3, RE5-1, and RE5-2). Each sample was thinned to about 100 nm using a focused ion beam (Ga+). The thin film samples were irradiated with an electron beam using an high voltage electron microscope (1000 keV, liquid nitrogen cooling holder) and in-situ observation was performed. From the observation of the lattice image and the electron diffraction pattern before and after irradiation, it was found that all the samples were amorphized by electron irradiation, but the irradiation behavior up to amorphization differed for each sample. The high-entropy samples have irradiation resistance compared to the conventional REBCO (RE1).

Tentative

P2-057

The Influence of Particle Irradiation on the Electrical Properties of REBCO High-temperature Superconducting Strips for Compact Fusion Reactors

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In compact fusion devices, REBCO tapes will encounter a harsh irradiation environment. Investigating the Jc (critical current density) variation of the strips with different particle implantations and delving into the microscopic response mechanism are significant to the operation of superconducting magnets. The superconducting layer with doped second-phase particles possesses its irradiation resistance by introducing phase interfaces and local strains under low-dose He irradiation. In addition, the changes in superconducting characteristic structure, especially point defects evolution with different doses of Au ion irradiation, greatly dominate the electrical properties of REBCO tapes. Here, through spectroscopic tests and aberration-corrected TEM techniques, the lattice parameters, oxygen vacancies and electronic structures of superconducting microstructures are analyzed to deeply explore the mechanism of variation of electrical properties. To simulate the winding and electromagnetic force acting on the tapes under fusion operating conditions, we studied the impact of strain coupling with γ irradiation on the superconductivity of the REBCO strips. These studies offer a reference for evaluating the performance of REBCO strips in the real fusion irradiation environment and deeply understanding the damage mechanism of the strips' superconductivity with the assistance of advanced characterization methods.

Tentative

P2-058

Fabrication of high-purity Zr_3AlC_2 MAX phase by spark plasma

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MAX phase materials are considered to have broad application prospects in the nuclear field due to their radiation resistance, high temperature resistance and corrosion resistance.Zr has a smaller thermal neutron absorption cross-section, making Zr-based MAX phases highly promising for nuclear applications. However, previous studies have reported challenges in synthesizing high-purity Zr_3AlC_2, with the highest purity achieved being only 59wt%. In this study, a high-purity Zr_3AlC_2 (>90wt%) was successfully synthesized using a stacked powder sintering method.

A two-layer powder stacking method was employed to prepare the sample, with the lower layer consisting of a mixture of ZrH_2, ZrC, and Al powders in a molar ratio of 2:3:2 while the upper layer comprised pure ZrC. Sintering was conducted in a vacuum environment using the SPS process at 1470 ℃, under an applied pressure of 35 MPa, with a holding time of 4 hours. After sintering, the phase identification and weight-fraction estimation were conducted using XRD, and its microstructural morphology was examined by SEM. He ion irradiation was used to study the defect evolution of the material under irradiation.

High-purity Zr_3AlC_2 was successfully synthesized through a dynamic sintering process using stacked powder sintering. XRD analysis confirmed that the final product contained more than 90 wt% Zr_3AlC_2, with only trace amounts of ZrC and Zr_2Al_3 as impurities. SEM observations revealed that the initially Al-free ZrC layer reacted with the Zr-Al liquid phase from the lower layer, leading to the formation of high-purity Zr_3AlC_2.

Tentative

P2-059

Evaluation of Helium Irradiation Damage to Dielectric Mirrors in Laser Fusion Reactors

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In laser fusion systems, dielectric multilayer mirrors are being considered for focusing laser beams onto fuel targets. These mirrors, which consist of alternating layers of high and low refractive index thin films deposited on a substrate, achieve extremely high reflectivity. This laser focusing system, known as the final optics, must withstand both the fluence of intense incident laser beams and radiation from the target. High-energy neutrons and helium particles generated by the fusion reaction become the main emission products, and there are concerns that these will damage the final optical system and degrade the mirror's focusing performance. To understand mirror damage caused by irradiation, multiple studies have conducted neutron irradiation experiments on dielectric mirrors and evaluated their neutron resistance. However, no research has been conducted on helium at the energy levels expected in fusion reactors. In this study, we conducted experiments irradiating 1.5 MeV helium ion beams onto HfO_2/SiO_2 dielectric mirrors, which are commonly used as mirrors for high-power lasers, to evaluate the effects of helium irradiation on dielectric mirrors. Changes in the reflectivity of the dielectric mirrors and structural changes in their surface and cross-section were assessed before and after irradiation. The results revealed that helium irradiation at a fluence equivalent to one day of commercial laser fusion reactor operation caused a significant decrease in mirror reflectivity. It was determined that this occurred because helium bubbles formed on the dielectric mirror surface lifted the multilayer film, creating stress, and ultimately caused delamination of the mirror.

Invited

P2-060

Effects of neutron irradiation on LaB6-emitters used in neutral gas pressure gauges

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Crystal emitter neutral gas pressure gauges to characterize neutral gas pressure as well as particle exhaust have been regularly used in different fusion experiments and will be used in future fusion devices such as ITER and beyond. Reoccurring deactivation of the commonly used LaB_6-crystals has been observed in LHD, especially during deuterium operation [1], leading to failure of the neutral gas pressure gauge and suspension of the neutral gas pressure measurements. Even though the neutron fluxes obtained in LHD during deuterium operation were comparably low, the natural boron in the LaB_6-emitters consists of 20% ^{10}B, which has a large neutron cross section and is transmuted into ^{7}Li and ^{4}He when interacting with neutrons. To study the effect of neutrons on the LaB_6-emitters at different fluxes, new LaB_6-crystals were irradiated with neutrons in the TRIGA MARK-II-reactor Vienna and subsequently underwent surface analysis to characterize surface structure and composition. The effects on the performance of the LaB_6-crystals as thermionic emitters in neutral gas pressure gauges will be studied in the future.

[1] Haak et al. 2024. Performance of LaB6-Emitters for Neutral Gas Pressure Gauges during Plasma Operation in the Large Helical Device. IEEE Transactions on Plasma Science, 2024, 52(9), pp. 3511?3515

ACKNOWLEDGMENT

This work was supported by the International Research Exchange Support Program of the National Institutes of Natural Sciences.

Tentative

P2-061

Generative AI-expert system for the characterization of optical materials in nuclear fusion engineering

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Machine learning-powered materials databases have complemented traditional handbooks dedicated to optical materials in nuclear fusion engineering. Cutting-edge artificial intelligence (AI) solutions will be another giant step in removing barriers to help on our ultimate path for designing and construction of commercial fusion devices. The main objective of this work is discovering high-level insights using innovative generative AI research to extract hidden knowledge in huge data sets obtained from the characterization of optical materials. It will be useful in the design and construction of ITER and DEMO diagnostics. Specifically, research activities will focus on the creation of a generative AI platform to improve, study and predict the behaviour of optical materials through the implementation of AI services that can operate on foundational models hosted in hybrid clouds. Furthermore, the available experimental information generated from previous projects in ITER and DEMO and their scientific analysis (based on metadata and complementary information introduced by expert scientific supervisors) will be used to evaluate the possibility of replacing, or at least complementing, complex analysis in classical materials databases with predictions processed by more powerful conversational search engines. The use of new AI paradigms in the cloud will add value and increase scientific productivity from the data generated in material testing and fusion experiments, for instance, material testing that is being carried out to validate the design of the diagnostics belonging to the ITER tokamak, as it is the wide-angle viewing system (WAVS) diagnostic of ITER

Invited

P2-062

Neutron Irradiation of DEMO Functional Materials: Bridging Current Research and Future Requirements for Diagnostic, Control and Heating Systems

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Invitation

Insulators and optical materials, such as optical fibres, microwave windows, magnetic sensors and mirrors, are critical components for all demonstration nuclear fusion devices and future fusion reactors.

These materials are essential for diagnostic, safety, and heating systems, which demand highly specialized functional materials capable of maintaining their properties under the extreme operational conditions of fusion reactors. Understanding the mechanisms of radiation-induced damage, particularly from high-energy neutrons as well as from gamma irradiation, is therefore essential to ensure their performance and longevity.

As part of the EU Materials Work Package, a series of fission-neutron irradiation tests have been conducted to assess the impact of radiation at doses relevant to the DEMO reactor. Thereby data on a variety of materials was generated that is required to enhance the materials technology readiness level and acquire valuable insights in material degradation mechanisms, as well as information required in the ongoing selection process. This work will present the actual status of this materials down-selection, including discussion of the individual selection criteria, specially related to their performance under neutron irradiation.

However, despite significant progress for individual materials, a global and more strategic approach is needed to address the complexities of the various systems.

Therefore, recent design interface meetings with specialists for Diagnostics, Control, Heating, and Divertor have facilitated the development of a comprehensive testing framework and robust test matrix for generating operationally relevant material property handbook data. The present status of this collaborative effort will be presented

P2-063

Developing Metallic Glasses for Nuclear Fusion applications

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Metallic Glasses (MGs) exhibit an impressive set of properties offering high potential to overcome several materials challenges in Nuclear Fusion (NF). They range from high corrosion resistance over high mechanical strength to high radiation tolerance including possible self-healing of irradiation-induced changes at the atomic scale. However, dedicated design of MGs for NF is practically non-existent. Seemingly contradicting constraints arise when attempting to reconcile high glass-forming ability (GFA) on the one hand with low-activation behaviour on the other. However, the high compositional flexibility of MGs allows them to be tailored for optimal use in various areas of NF or may enable several related applications in the first place. After a short introduction to MGs in general, we present preliminary results of our research on MGs for NF. Outstandingly high hardness for metallic materials has been found in combination with high glass-transition and crystallisation temperatures. We also propose the development of a dedicated roadmap to expedite the future efforts for design and application of MGs for a wide variety of applications in the NF sector.

Tentative

P2-064

In-vessel radiofrequency power transmission circuits development for first mirror plasma cleaning in 55.C2.ETS and 55.E6.VSRS ITER optical diagnostic systems

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The final qualification of radiofrequency (RF) First Mirror plasma cleaning systems necessitates experimental validation using representative in-vessel circuit components. These include sections of mineral-insulated (MI) cables, vacuum connectors and interfaces, and low-loss power transmission lines on the ex-vessel side. Development of these RF transmission components requires accurate assessment of power budgets and lifetime plasma cleaning performance. At present, in the prototyping experiments, First Mirror plasma cleaning systems employ standard RF components that are incompatible with the operational and environmental constraints of the ITER tokamak. The development of First Mirror plasma cleaning has demonstrated that effective mirror cleaning requires RF powers up to 2.5 kW for the Edge Thomson Scattering (ETS) and 0.3 kW for the Visible Spectroscopy Reference System (VSRS) ITER diagnostic systems. This work presents experimental studies conducted on the ETS First Mirror plasma cleaning prototype, incorporating realistic circuit components such as 6-mm mineral-insulated Okazaki cable sections and custom-designed Staubli vacuum coaxial interfaces exploring multiple impedance matching scenarios. For the VSRS system, a methodology is proposed for qualifying power delivery to the load. This approach utilizes voltage standing wave ratio (VSWR) and frequency tuning measurements, supported by both DC-bias and RF voltage monitoring. The voltage monitors capture signals at the vacuum pre-matcher located adjacent to the First Mirror. The results of these investigations provide a framework for defining qualification procedures for power transmission circuits used in RF plasma cleaning within the ITER environment.

Tentative

P2-065

Optimization of Sintering Conditions for Tungsten Boride Cermets for Application as Radiation Shielding Materials

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Tungsten boride (WB) exhibits suparior neutron and gamma-ray shelding performance compared to conventional radiation shielding materials such as water and lead. Therefore, it However, low sinterability, brittleness, and poor thermal conductivity of WB pose challenges for structural application in ST. To improve the thermal and mechanical properties of WB, the authors fabricated WB cermets with the addition of Cu (WB-Cu) and evaluated their material properties. Simulation results using PHITS suggested that Cu addition does not significantly affect the shielding performance of WB. Additionally, the cermet formation with Cu improved sinterability, fracture toughness, and thermal diffusivity. Since wettability between WB and Cu is a critical factor in cermet formation, we aim to further improve the properties of WB-Cu cermets by enhancing wettability and optimizing sintering conditions. The measurement results of the material properties will be presented at the conference.

Tentative

P2-066

Volumetric Swelling in Helium Ion-Irradiated WB2-Based Neutron Shielding

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The compact size of spherical tokamaks necessitates advanced neutron shielding to minimise irradiation damage and reduce heat deposition in the superconducting magnets at the central column. Tungsten diboride (WB?) is an excellent candidate for this application due to tungsten's neutron reflection capability and boron's high neutron absorption cross-section via $^{10}B(n,\alpha)^{7}Li$ reaction. To understand the impact of the 1.77 MeV alpha particle generated in the $^{10}B(n,\alpha)^{7}Li$ reaction, hot-pressed samples of WB_2 were irradiated at the Dalton Cumbrian Facility with 2.1 MeV helium ions across a temperature range of 300-700°C, achieving fluence of 8 x 10^17 ions/cm^2 (peak damage of 10 dpa). Swelling due to irradiation induced damage as a function of temperature was examined by characterising the samples with transmission electron microscopy (TEM), scanning electron microscopy (SEM), and glancing incidence X-ray diffraction (GIXRD). At ~300°C, WB_2 undergoes amorphisation caused by high electronic stopping power of helium ion. Above 400°C, WB_2 exhibited no signs of amorphization, and the lattice strain was anisotropic, with greater expansion along the c-axis than the a-axis. In the shallow region (0 to 3 μm, ~1 dpa) of the samples, swelling decreased with increasing irradiation temperature, likely due to thermal recovery of point defects. In contrast, in the Bragg peak region (3 to 3.5 μm, ~10 dpa), swelling increased with temperature as He nanobubbles coarsened, which led to enhanced cracking. The study discusses the potential net effect of lattice swelling and helium bubble swelling for reactor relevant conditions.

Invitation

P2-067

Establishing and testing of alternative low activation tungsten carbide shielding materials for fusion energy systems

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Tungsten carbide (WC) is under consideration for future fusion power plants, specifically the central column shield of spherical tokamaks. This is because it is an effective shielding material against 14 MeV neutrons due to the combination of favourable neutron attenuation properties and high density, The inclusion of Co improves the mechanical properties of WC components, allowing for the fabrication of various geometries. But this results in the formation of deleterious 60Co and significant decay heat after neutron exposure. To mitigate this issue, samples of binder-less WC have been assessed in terms of shielding performance, waste production and decay heat using the Monte Carlo N-Particle code (MCNP) and activation code FISPACT-II codes respectively. The results with research grade WC compositions with FeCr binders, and low-concentration Co binders. In addition to the neutronic characteristics, thermal loads from the fusion plasma and volumetric nuclear heating from neutron interactions substantiates the need to understand the thermal properties of candidate shielding materials, while the size/density of the shielding component necessitates the need to explore thermo-mechanical properties. Here, we will present a comprehensive comparison of the specific heat capacity, thermal conductivity, thermal diffusivity, thermal expansion, hardness, flexural strength, and fracture toughness for WC with 2.6 wt.% Fe-Cr, 3.3 wt.% Co, and binder-less WC samples. The development challenges for establishing WC variants to achieve adequate technical readiness for fusion pilot plants will be discussed.

Tentative

P2-068

Preparation of tungsten-based shielding material with large size

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As fusion devices evolve towards more compact designs, the available space for shielding becomes increasingly constrained, necessitating the urgent development of high-performance shielding materials to fulfill shielding requirements. This work focuses on advanced tungsten-based shielding materials, optimizing their sintering processes and achieving the preparation of large sized blocks through hot pressing sintering (HPS) and hot isostatic pressing sintering (HIPS). By refining the HPS parameters, the porosity of the tungsten-based shielding materials was minimized, resolving issues of delamination and cracking arising from the removal of pressing aids. Consequently, favorable physical and mechanical properties, including elastic modulus, bending strength, and fracture toughness, were successfully attained. The study revealed a granular structure and phase aggregation distribution within the materials sintered by HPS. Residual internal stress at the phase interfaces on the granular microstructure edges was found to facilitate crack propagation, thereby reducing fracture toughness. However, this residual stress also imparted improvements in hardness and bending strength. Furthermore, the density and uniformity of the materials were enhanced through the use of atomized powder and the HIPS process. HIPS enabled the production of dense at a relatively low temperature of 1250°C. Nevertheless, the proportion of brittle phases increased after HIPS, leading to a higher number of crack sources during bending tests and consequently lower bending strength compared to HPS blocks. Despite this, mechanisms such as microcrack toughening, crack deflection, and bridging significantly enhanced the fracture toughness of HIPS blocks compared to those of HPS blocks.

Template

P2-069

A new concept of radiation shielding material - W-B-C system for improved mechanical properties.

Seiji Nakabayashi[1], Naoya Matsuta[1], Kei Miura[1], Shuhei Nogami[1], Shigekazu Yamazaki[1]

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A shielding material protecting a superconducting coil of fusion reactors from radiation is installed in a limited space. It is required to develop a material excellent in radiation shielding efficiency. Tungsten boride compounds is excellent in the shielding efficiency, but poor in fracture toughness, etc. It is necessary to improve the mechanical properties. The concept of on-going development is to improve the properties of tungsten boride compounds with the addition of carbon. To evaluate the shielding efficiency, Monte Carlo particle transport simulation code (PHITS) was applied. The following properties were measured for the fabricated materials: density, Vickers hardness, fracture toughness, and thermal diffusivity. Cross-sectional observation was also conducted. The density of developed material has achieved almost up to the true density. This presentation will show details of the shielding efficiency, the properties, and cross-sectional observation.

Tentative

P2-070

Advanced microstructural characterisation of tungsten borides as high-performance shields for fusion

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Tungsten borides are highly effective shielding materials against the 14 MeV neutrons due to the combination of favourable neutron attenuation properties and high densities. However, significant uncertainties exist in the stoichiometry and crystallography of some of the compounds identified in the tungsten - boron (W - B) system, partly because of the challenges associated with producing a single-phase sample that is large enough to use with conventional methods of analysis. We conducted interdiffusion studies on W - B diffusion couples to understand the growth mechanism and microstructural evolution of the phases formed in this system. Here we focus on clarifying the compositional and crystallographic aspects of the ε phase W_2B_5 - the preferred phase for shielding the high temperature superconducting (HTS) magnets in the centre column of Spherical Tokamak designs that Tokamak Energy is championing. We present the data, focusing on the ε phase formed in the interdiffusion zones of W - B diffusion couples using advanced microstructural analyses, including Electron Probe Micro Analysis (EPMA) and Transmission Electron Microscopy (TEM), and compare the results with the existing W - B phase diagram. Single crystals of the ε phase produced using the aluminium flux method were used as internal standard for EPMA analysis to determine its composition to high accuracy. Coupled with nano-hardness measurements, we show that the diffusion couple technique is likely an efficient method to understand composition - microstructure - property relationships in the W - B system.

Invitation

P2-071

Preparation and Properties of Tungsten Monoboride Shielding Material

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Tungsten Boride, Tungsten Powder, and Boron Powder were used as raw materials, and the hot-pressing method was used to produce dense tungsten monoboride (WB) ceramic material. The density and open porosity of the WB material were tested using the drainage method, and the microstructure of the material was analyzed using scanning electron microscopy (SEM). The thermal expansion coefficient and thermal conductivity of the WB ceramic were evaluated using a thermal expansion analyzer and a laser thermal conductivity meter, respectively. Furthermore, the shielding performance of the WB ceramic against fast neutrons and gamma rays was studied through theoretical and experimental analyses. The results show that after hot-pressing sintering at 2000°C and 30 MPa for 1 hour, the density of the WB ceramic reached 15.1 g/cm^3. The material exhibited excellent thermal stability and demonstrated effective shielding performance against fusion-generated fast neutrons and secondary gamma rays. These properties make it a highly promising candidate for superconducting magnet shielding applications in fusion reactors.

Tentative

P2-072

Ion Irradiations of tungsten carbide and borides for fusion neutron shielding

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Tentative

To address the unique challenge of neutron shielding in a highly limited space within compact spherical tokamaks such as STEP, high performance shielding materials must be utilised. Suitable materials must moderate and capture neutrons across a broad energy spectrum throughout operation; possess a high packing density and thermal conductivity; maintain structural integrity under irradiation; limit the retention of tritium; minimise transmutation to high activity radioisotopes over maintenance timescales; and minimise radioactive waste inventory.

Refractory ceramics such as tungsten carbides and borides are leading candidates for such applications due to their high cross-sections for neutron moderation and capture across a broad energy spectrum. However, little is currently understood about their response to irradiation at fusion relevant temperatures and doses. Of particular concern for ceramic shielding materials are irradiation induced swelling and crack formation, which may significantly degrade shielding performance.

In this work, high energy He, Li, Re, W, and Ar ions have been used to irradiate cemented W-B and WC. The irradiation doses, implanted ion concentrations and temperatures are in-line with expected operational conditions in the inboard shield of a compact spherical tokamak. As such, we have attempted to simulate both fusion neutron displacement damage up to 50 dpa and the behaviour of expected transmutation products up to 15,000 appm. Irradiations have been performed up to 600 °C in-situ in a transmission electron microscope and up to 400 °C in bulk samples, with extensive post-irradiation examination performed to characterise radiation damage defects and their effect on material properties.

P2-073

Study of WC based cemented carbides as shielding candidates for compact spherical tokomaks

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Compact spherical tokomaks offer multiple benefits compared to conventional tokomak reactors. These include higher magnetic field strengths meaning superior reactor power densities, shorter build times and lower costs. However, spherical tokomaks pose a unique challenge owing to their design: limited space for the central column shield. The shield is vital as it protects the superconducting magnets housed within the central column from neutron radiation induced damage and high temperatures. Hence, materials with high shielding efficiency are required.

Tungsten carbide (WC) based cemented carbides are one of the preferred shielding candidates due to several desirable properties: high neuron attenuation across a wide energy spectrum, high melting points (~ 2700 °C) and high thermal conductivities (70-180 $Wm^{-1}K^{-1}$). Even though cemented carbides have been studied extensively due to their commercial use in the cutting and drilling tools industry, there is a lack of knowledge in contemporary literature regarding their mechanical properties with respect to irradiation damage and operational temperatures in the context of a fusion reactor. This work aims to explore this facet.

We performed W self-ion irradiations on WC-FeCr samples at a variety of doses (0.13, 1.3 and 13 dpa) at both room and elevated temperatures (100, 250 and 400 °C). The mechanical properties of the samples (i.e. modulus, hardness and toughness) were then measured via nanoindentation and micro-cantilever bending tests at irradiated temperatures and compared to unirradiated sample properties to understand the effect of irradiation damage.

Tentative

P2-074

Development of additive manufacturing method for high-purity SiC and its material properties

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Silicon carbide (SiC) is a promising candidate for fusion reactor blankets due to its high strength, excellent radiation resistance, and thermal stability. However, its low fracture toughness makes conventional machining difficult, limiting the complexity of structures that can be achieved. Additive manufacturing (AM) offers a solution by enabling the fabrication of complex SiC structures. Current AM-SiC technology primarily focuses on achieving high mechanical strength and density, but residual impurities, particularly molten silicon, pose significant concern for nuclear applications.

In this study, we aimed to develop a high-purity SiC fabrication process by optimizing the debinding conditions. While complete binder removal is essential for purity, excessive debinding before sintering can cause structural degradation. To address this challenge, we optimized debinding parameters to establish conditions that effectively eliminate binder while preserving the shape integrity of the debinded SiC powder structure.

We found that performing a chemical debinding before the thermal debinding is effective in achieving complete debinding while maintaining the shape of the AM-SiC structure. Based on the weight change analysis during the chemical debinding process, the reaction rate of chemical debinder was not dependent on the sample size but rather only on temperature below 10 mm thick. This means that the rate of chemical debinding can be controlled solely by temperature, which is critically important for preventing interlayer delamination and deformation. This makes it particularly useful for debinding thick structures that are prone to such defects.

We will also present the details of strength tests and microstructure analysis of the sintered AM-SIC.

Tentative

P2-075

Effect of Zr/Hf addition on ODS steel

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Oxide-dispersion-strengthened (ODS) steel is regarded as one of the promising candidate materials for the blanket components in DEMO reactors. However, the underlying mechanism of irradiation hardening at T<0.3Tm, bubbles behavior under irradiation needs detailed investigation. Energetic heavy-ions, indentation tests, and TEM were used to study the change in microstructures and hardness, as well as the bridge between them.

Zr/Hf-added FeCrAl ODS alloys were investigated (denoted as P1 and P2). The Hf addition will further refine oxides. The oxides in P1 is $Y_4Zr_3O_{12}$ and YAM while in P2, ther are $Y_2Hf_2O_7$. The $Y_4Zr_3O_{12}$ and $Y_2Hf_2O_7$ oxides are coherent with the matrix whereas some YAM oxides exhibited semi-coherent relationship. Hf addition will improve the coherency of oxides. By the way, the strengthening mechanism was dicussed vis dispersion barrier hardening (DBH) model. The barrier strength factor, α of coherent oxides being 0.20, lower than that of semi-coherent counterparts, 0.36. Meanwhile, lower irradiation hardening was observed in Hf-added specimens which benefits from the enhanced sink strength that are mainly calculated based on dispersed oxides. Effect of sink strength on hardening was investigated by comparing with other ODS steels irradiated at -50 °C. The temperature effect that higher irradiation temperature will reduce irradiation hardening and increase the ability of sinks to affect irradiation hardening was revealed.

Lastly, the Ar bubbles attached to oxides in MA956 was investigated. The attached Ar bubbles were apparently dissolved in the swift Xe ions deposition region. High energy PKAs and the multi-cascades can lead to efffective dissolution of Ar-bubble dissolution.

Tentative

P2-076

Mechanisms of hydrogen and oxygen adsorption on vanadium super-permeable membrane surfaces

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Understanding the fundamental physics governing hydrogen uptake and release from surfaces is essential for predicting tritium permeation through membranes used in direct recycling of fusion exhaust. Key processes such as chemisorption, dissociation/recombination, and surface-to-bulk transport are sensitive to surface conditions, including impurities. Traditional surface analysis techniques often struggle to detect hydrogen or provide comprehensive insights into its behavior.

To address these limitations, we have used low energy ion scattering (LEIS) and direct recoil spectroscopy (DRS) to investigate hydrogen interactions with oxygen and carbon impurities on vanadium (V) surfaces. These techniques are complemented by high-resolution thermal desorption spectroscopy which is used to measure surface coverage of H and O, as well as XPS for surface chemistry analysis. By characterizing hydrogen chemisorption on various V oxide phases, we aim to assess the transport of hydrogen through oxide layers. Ongoing experimental and modeling work has focused on chemisorption on V(111) and V(100) surfaces. To decipher the mechanisms that drive hydrogen permeation into the vanadium membranes, we have performed DFT simulations of hydrogen adsorption onto these materials to quantify the effect that chemisorbed impurities, namely oxygen, have on the ability of hydrogen to diffuse through the membrane surface by preventing recombination. Our research will also include work at the IGNIS-2 facility at Penn State University for in-situ characterization of hydrogen-oxygen interactions on V surfaces, allowing for controlled oxide layer deposition and subsequent analysis.

Invitation

P2-077

New manufacturing strategy of SiC fiber required for fusion reactor application

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Breeder blankets in fusion reactor are exposed to high temperature and significant neutron doses with high energy. Thus, the materials applied to blankets must possess heat tolerance, and neutron radiation resistance. A promising candidate that meets these requirements is silicon carbide (SiC) fiber reinforced SiC ceramic matrix composite (SiC/SiC CMC).

In particular, SiC fiber for blankets is required to have high strength, crystallinity, and purity. Conventional methods for producing high-quality SiC fiber with these properties involve high-cost processes, including electron beam irradiation based polycarbosilane (PCS) curing process, which significantly increases the production cost of SiC fiber. In addition, the precursor of SiC fiber in those methods contains oxygen atoms at the molecular ends, which are incorporated into pyrolyzed SiC fiber and degrade the fiber's quality.

In this study, we present a novel manufacturing method for high-quality SiC fiber aimed at breeder blanket applications for fusion reactor. To avoid the electron beam irradiation, we adopted the dry spinning process that doesn't need curing steps. Moreover, to reduce the produced SiC fiber's oxygen content, we chose the starting material not containing oxygen atoms in the molecular structure.

Further investigation revealed that the PCS structure significantly affects the strength of the produced SiC fiber. Based on the relationships between properties of PCS and SiC fiber, we also discuss suitable process conditions for PCS synthesis and subsequent pyrolysis to obtain high-quality SiC fiber in our method.

Tentative

P2-078

High entropy oxides with tunable thermal-mechanical properties for fusion applications

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High-entropy ceramics (HECs) are inorganic compounds with one or more Wyckoff sites shared by equal or near-equal atomic ratios of multi-principal elements. Attributed to the rich crystal and electronic structures, it provides huge space for properties tuning with enhanced thermal and mechanical characteristics. In this report, we present the design and synthesis of a series of new high entropy fluorite oxides. XRD and EDS mapping confirm a single cubic fluorite phase with a uniform elemental distribution. In-situ high-temperature XRD shows that the phase is stable with no phase separation or secondary phases. These fluorite oxides have tunable coefficients of thermal expansion and thermal conductivities. Our results add to the compositional space of high-entropy ceramics for a wide range of applications as thermal and electrical insulators for fusion devices.

Tentative

P2-079

Evaluation of SiC and SiC-based ceramics for use in advanced vacuum-vessel designs

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Tentative

The Liquid Sandwich Vacuum Vessel (LSVV) is a novel disruption proof vacuum vessel design which features a non-conductive structural material shell enclosing a conductive liquid. The leading candidate material for the shell is Nano Infiltration Transient Eutectic Phase Silicon Carbide Matrix reinforced with Silicon Carbide fibers (NITE-SiC/SiC) This assembly is contained within the liquid immersion blanket of the ARC Tokamak.

Experimental work has been done to develop Transient Grating Spectroscopy (TGS) as a tool for measuring changes in thermal diffusivity in 3C-SiC during ion irradiation. Results will be presented that highlight the change in thermal diffusivity of 3C-SiC due to ion irradiation, examining the effects of irradiating species, dose rates, and temperature. The TGS setup at MIT provides in-situ measuring capability with a fluence resolution of ~0.001 dpa and allows material properties to be probed within the “uniform” damage layer. This opens the door to studying the effects of transmutants which will be the next phase of the study.

Results will also be presented to show the response of NITE-SiC/SiC to fluoride molten salts, specifically FLiNaK, LiF/NaF and FLiBe. The microstructure of the sample cross section is characterized using SEM/EDS. Immersion in FLiNaK at 700 C for 24 hours results in significant intergranular corrosion, likely aided by the presence of alumina in the NITE-SiC/SiC. Large cracks were seen in the microstructure which might be explained by stress corrosion cracking, resulting from the misfit stress between the fibers and matrix. The misfit stress was estimated using Raman Spectroscopy.

P2-080

Measurement of low concentration deuterium in lithium using gas chromatography method with water dissolution - isotope exchange reaction

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Evaluation of neutron-induced material damage is essential for the realization of a fusion reactor. In Japan, the construction of A-FNS is planned as a materials irradiation facility. The high intensity neutron beam is generated by the reaction of flowing Li and deuterium D beams. Hydrogen isotopes are major impurities and are removed by hot trapping with yttrium. Since the design value of permissible concentrations of hydrogen isotopes is around 10 wppm, an effective method to measure a low concentration hydrogen isotope is required. In this work, a gas chromatography (GC) method with water dissolution - isotope exchange reaction is proposed and its performance for D measurement in Li was evaluated.

Li sample was heated in a Mo crucible to 400 °C with D_2/Ar gas flowing for D absorption. After the D_2 exposure, distilled water was slowly dripped to the Li. HD and HDO generated by the dissolution reaction of Li with water were transported by the Ar flow to a Pt catalyst. Pure H_2 gas was introduced to the catalyst bed to convert HDO to HD. HD concentration in the outlet gas was measured by the GC operated with a career gas of H_2.

In cases of 19780 and 984 ppm D_2/Ar absorption, D concentration in Li was determined to be 451 and 26.8 wppm, respectively. Considering the detection limit of GC, it can be said that it is possible to sufficiently measure D concentrations up to about 10 wppm.

Invitation

P2-081

Mechanistic Evaluation of Deuterium Trapping in Strengthening Microstructure Features of Ferritic-Martensitic Steels

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The development of fusion structural materials requires a delicate balance between mechanical strength and tritium retention. The microstructural features introduced for strengthening, such as precipitates, dislocations, and boundaries, often act as hydrogen trapping sites, potentially compromising fuel efficiency and posing radiological safety risks in fusion reactors. This study fundamentally investigates the role of individual microstructural features in deuterium trapping and their correlation with mechanical strengthening in reduced-activation ferritic-martensitic (RAFM) steels. A series of model alloys and engineering steels were fabricated and subjected to controlled heat treatments to tailor the density and distribution of deuterium trapping sites, including grain boundaries, dislocations, $M_{23}C_6$ precipitates, and TiC precipitates. Deuterium retention behavior was characterized through D_2 gas charging and thermal desorption spectroscopy, complemented by multiscale microstructural characterization (electron backscatter diffraction and scanning/transmission electron microscopy) and synchrotron X-ray diffraction. The results reveal that TiC precipitates exhibit the highest deuterium trapping capacity, followed by $M_{23}C_6$ precipitates, with dislocations and grain boundaries showing the lowest and comparable trapping efficiencies. Furthermore, the relationship between deuterium trapping capacity and mechanical strengthening was quantified, demonstrating that TiC precipitates provide the highest deuterium trapping efficiency per unit of mechanical strengthening. These findings provide key insights into optimizing microstructural design for future RAFM steels, balancing mechanical performance with hydrogen isotope management to ensure fuel efficiency and radiological safety in fusion reactors.

Tentative

P2-082

Purge gas effect on tritium release behavior from neutron-irradiated FLiNaBe mixed with Ti powder

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FLiNaBe is a candidate as a liquid blanket material for fusion reactors. Since FLiNaBe has low hydrogen solubility, leakage of bred tritium is concerned. To improve the effective tritium solubility, the addition of Ti powder to FLiNaBe was proposed. However, tritium release behavior from Ti mixed FLiNaBe is not well understood. In this study, solid state FLiNaBe mixed with Ti powder was irradiated by neutrons at the Kyoto University Research Reactor, and tritium release behavior from the molten FLiNaBe was investigated at Kyushu University.

Neutron-irradiated sample was heated in a Mo crucible to 600 °C with flowing Ar gas or 1% H_2/Ar. Tritiated water vapor (T_2O and HTO) was collected in a cold trap at about -60°C, and TF was collected in the first water bubbler, and gaseous tritium (HT and T_2) was collected in the second water bubbler after oxidation in a CuO bed. The tritiated water vapor accumulated in the cold trap was collected in a water bubbler at room temperature. Tritium concentration in each water bubbler was measured by a liquid scintillation counter.

Most of the tritium was released as gaseous tritium not TF for both of Ar and 1% H_2/Ar purge. Enhancement of release of gaseous tritium by H_2 in purge gas was not observed. The total amount of tritium released by 1% H_2/Ar purge was lower than that by Ar purge. This is presumed to be because H2 in the purge gas dissolves into the molten salt and improves the tritium absorption performance of Ti.

Tentative

P2-083

Comparison of deuterium desorption from tungsten fuzz between multiple plasma devices

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Tentative

Tungsten (W) nanostructure, so-called fuzz, can form on a divertor plate exposed to a plasma containing helium (He). Our previous research revealed that a W fuzz layer exposed to deuterium (D) plasma starts to release D from a low temperature of ~350 K (K. Saito et al., NME 2025). However, there are only a few cases, showing such low-temperature desorption of D. To explore whether the low temperature desorption of D from W fuzz is common, we have compared D desorption behavior from W fuzz samples prepared in two different plasma devices.

A sample (a) was exposed to a He plasma in the compact rf plasma device APSEDAS. The incident He ion fluence, sample temperature, and incident ion energy were 5.0×10^25 He/m^2, ~1100 K and ~80 eV, respectively. Another sample (b) was exposed to a He plasma at 1.0×10^26 He/m^2, ~1123 K and ~80 eV in the PISCES-A linear plasma device with a dc arc plasma source. These samples were then exposed to a D plasma (1.0×10^25 D/m^2, ~400 K, ~40 eV) in PISCES-A.

A thermal desorption spectrum of the sample (a) exhibited a pronounced D desorption peak of ~5.3×10^17 D/m^2s at ~385 K, while the sample (b) had a much lower peak of ~2.7×10^16 D/m^2s at ~415 K. The difference might be caused by different fuzz structures grown in the different plasma devices. The fuzz structures will be analyzed with transmission electron microscopy and energy-dispersive X-ray spectroscopy in detail.

P2-084

Deuterium retention properties of He-W co-deposited layers formed at different sample temperature and flux ratio of W to He

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In the divertor region of fusion reactors, part of sputtered tungsten (W) particles redeposits with helium (He), forming He-W co-deposited layers on the divertor plates. The tungsten-deposited region of the divertor is expected to have a wide distribution of both surface temperatures and flux ratios of W to He (Γ_W/Γ_He). In this study, He-W co-deposition is explored in the linear plasma device APSEDAS with a particular focus on how altered deuterium (D) retention properties depend on these parameters. Several He-W co-deposited layers were formed on pristine W surfaces exposed to He-Argon (Ar) plasma at sample temperatures of ~600 K and 860 K, and Γ_W/Γ_ion values of 10^(-4)-10^(-3). The gas pressure ratio of Ar to He was kept 0.022-0.023. The resultant ion fluences were kept constant at 2.2×10^25 m^(-2). These layers were then irradiated with D plasma under identical conditions at 2.5×10^25 m^(-2). After plasma exposure, porosity was calculated from the thickness of the layer measured with scanning electron microscope and the estimated thickness from the electronic balance. The D retention properties were investigated through thermal desorption spectrometry measurements. It was observed that He-W co-deposits formed at higher temperatures, 860 K, showed lower D retention compared to pristine W, and the co-deposits at lower temperatures. Also, the D retention increased with increasing Γ_W/Γ_ion while their porosity was decreasing. Notably, the ratio of D to W increased as the ratio of He to W in the co-deposited layer increased.

Tentative

P2-086

Tritium Production and Confinement in High-Temperature Gas-Cooled Reactors for Fusion Fuel Supply

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Introduction

D-T fusion reactors are designed to be self-sufficient in fuel, but the initial tritium procurement remains uncertain, posing a critical challenge for Japan's fusion development. We propose tritium production using high-temperature gas-cooled reactors (HTGRs), utilizing $LiAlO_2$ encapsulated in Al_2O_3 containers. A key issue is tritium confinement. Our previous study showed that hydrogenic tritium gas rapidly permeates Al_2O_3 tubes. This study examines tritium permeation under water vapor tritium conditions and evaluates the reduction effect of Zr spheres. Furthermore, we discuss technical challenges for tritium production in HTGRs.

Experimental Method

We used Li_2TiO_3, which releases tritium as water vapor tritium (HTO, T_2O), similar to $LiAlO_2$. A neutron-irradiated Li_2TiO_3 pebble was enclosed in an Al_2O_3 tube, sealed at one end, and heated to 700°C inside a quartz tube. Argon gas purged the exterior, and permeated tritium was captured in water bubblers. Bubblers placed before and after a copper oxide column distinguished HTO from HT (T_2).

Results and Discussion

A few Bq of hydrogenic tritium was detected in each experiment. However, a separate tritium release experiment with Li_2TiO_3 measured ~70 kBq, suggesting that tritium was confined for ~80 hours. These results highlight Al_2O_3's potential for tritium containment and emphasize the need for further studies on Zr's effect in reducing permeation. This study contributes to evaluating the feasibility of tritium production in HTGRs and addressing key technical challenges in fuel supply for fusion reactors.

Tentative

P2-087

Deuterium enrichment by AEM water electrolysis with varying water utilization

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The hydrogen isotopes deuterium (D) and tritium (T) are essential materials for fusion reactions. Due to the similar chemical properties of these isotopes, it is difficult to separate individual isotopes. To overcome this problem, we have studied D separation using solid polymer water electrolysis. Water electrolysis uses an anion exchange membrane (AEM), so that a non-precious metal catalyst can be used. The water content in the membrane has a significant impact on the mass transportation of D. In our study, we investigated D separation depending on the ratio of water consumption and supply. IrO_2 was used as anode and Pt/C as cathode catalysts. The initial D concentration of the water was 10 at.%. The hydrogen gas produced by electrolysis was measured using a quadrupole mass spectrometer to determine the ratio of H_2, HD, and D_2. The Fourier transform infrared spectrometer (FTIR) analyzed the water before and after electrolysis and determined D concentration. The separation factor (αl) of water drained from the anode and cathode of the cell was calculated. The value of αl is an index to evaluate the D separation performance. High factor value means that the D is concentrated in the electrolyte. When the ratio of water supply to consumption became nearly equal, the separation performance improved remarkably. This is discussed by D mass balance in AEM electrolysis.

Tentative

P2-088

Tritium Separation using Water Electrolysis and Fuel Cell with Polymer Electrolyte Membrane

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The separation and purification technology of hydrogen isotopes are important for nuclear fusion reactor. Since the isotopes have similar chemical and physical properties, the separation processes have low performance. It is well known as that tritium(T) and deuterium(D) evolve less than protium(H) during water electrolysis. However, it consumes a large amount of electricity. To overcome this issue, we have investigated the novel combined system of water electrolysis/fuel cell with polymer electrolyte membrane. Fuel cell can generate electrical energy by feeding hydrogen gases from electrolysis, which recovers a portion of consumed electricity. In the present study, we used tritiated water and evaluated the separation performance. JARI standard cell was employed and operate under constant current mode. The T concentration in the liquid was measured by scintillation counter, and one in the gas was calculated from mass balance. T was enriched in the electrolyte in electrolysis, while it was concentrated in the product water in fuel cell. This was well agreed with the D separation. The performance was higher than that for D. We changed the gas consumption rate in the fuel cell. Increasing the rate improved the apparent separation. The introduction of fuel cell contributed to high separation and low energy consumption, compared with the electrolysis alone.

Tentative

P2-089

Retention and Surface Morphology Evaluation of Fine-grain Dispersion-Strengthened Tungsten for Plasma-Facing Component Applications

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Novel fine-grain dispersion-strengthened tungsten materials developed by Energy Driven Technologies were exposed to low energy, high fluence plasmas to evaluate retention behavior, surface morphology, and microstructure resilience compared to industry standard tungsten (W). The tested materials include fine-grain dispersion-strengthened tungsten (DSW) with 3 wt% zirconium carbide (ZrC) dispersoids, fine-grain dense W without any dispersoids, and ITER-specification tungsten supplied by A.L.M.T. Corp. Samples were exposed at Sandia National Laboratories to high flux deuterium (D) and helium (He) plasmas at fusion-relevant plasma fluences (~10^25 atoms/m^2) and ion energies (75 eV) over a wide temperature range (200-450 C for D, 850 C for He). Helium ion microscopy revealed that for the D plasma-exposed samples, the ZrC dispersoids showed near-surface degradation at temperatures above 300 C, while no detrimental W surface morphology was observed. For the He plasma-exposed samples, intentional nano-structured fuzz formation was observed in the tungsten matrix. The DSW materials notably maintained their dispersoids despite the surrounding fuzz growth, with clear delineation between regions. Thermal desorption spectroscopy revealed that the DSW retained much more deuterium than the ITER W, up to 250% more at 200 C and 460% at 300 C. The fine-grain W without dispersoids was much more comparable, ranging from 27-150% higher retention than ITER W. Conversely, the fine-grain tungsten exhibited the highest He retention at 850 C, about 26% more than ITER W and the DSW. This study was conducted through the DOE-funded INFUSE initiative.

Sandia is managed and operated by NTESS under DOE NNSA contract DE-NA0003525.

Tentative

P2-090

Using TOMAS to study ITER and W7-X wall conditioning schemes

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TOMAS (TOroidally MAgnetized System) located in Forschungszentrum Juelich is used to study wall conditioning (ICWC,ECWC and GD) effects on fusion reactor sample materials by exposing them to the produced toroidal plasma. The device has various diagnostics tools to monitor the plasma prior and during sample exposure, the most important of which are a ToF-NPA (Time of Flight Neutral Particle Analysis) to measure the neutrals' spectrum (meaning flux in function of energy) , a RFEA (Retarding Field Energy Analyzer) to measure the ions' spectrum and a horizontal and vertical QLP (Quadrupole Langmuir Probe) to measure the electron density and temperature. TOMAS is very versatile as the plasma species (H,He,D or Argon), pressures, wall conditioning power and exposure time may all be varied. Using the aforementioned tools a plasma can be tuned to be relevant for other devices. The experiments planned for 2025 consist of ITER-relevant wall conditioning tests whereby boronized tungsten, doped with deuterium (to mimic tritium), will be subjected to ICWC and GD under ITER-relevant conditions in TOMAS to make de-trapping estimates for ITER, this will be done under ITER contract ADCYGE. Experiments relevant to the W7-X device will also be carried out if time permits, namely W7-X-relevant wall conditioning schemes with boronized graphite samples, whereby the boron is hydrogen doped. Days on the device may be booked by externals as will be presented.

Tentative

P2-091

Hydrogen transfer from pressurized water to gas phase through Inconel

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The high-temperature and high-pressure water is expected to be used as a coolant in JA DEMO. In the tritium breeding blanket, it is inevitable that a certain amount of tritium permeates into the cooling water. Therefore, it is necessary to deepen the understanding of the tritium transfer from the primary cooling water to the secondary cooling water in a heat exchanger. In this study, a double-tube permeation apparatus consisting of Inconel 600 tube and SS316 tube was assembled and hydrogen permeation behavior from water to gas phase was investigated.

A one-side sealed Inconel 600 tube was filled with purified water, and pressurized with helium gas, and sealed. Ar gas was introduced between the Inconel tube and the SUS316 tube and sealed it. The double tube part was heated to 300 ℃, and the pressure was increased to 17 MPa. The argon gas was released periodically, and the hydrogen concentration was measured using gas chromatography. Also the water vapor concentration in the argon gas was monitored using a dew point meter.

Hydrogen was obviously transferred from the pressurized water to the Ar gas through the Inconel. Also, water vapor concentration increased during the confinement of the Ar gas. It was found that hydrogen permeated from the pressurized water was mainly released to the Ar gas as water vapor. The hydrogen permeation rate decreased with heating time. This is considered to be because the oxidation reaction on the Inconel surface progressed, and the hydrogen production rate decreased.

Tentative

P2-093

Tritium permeation behavior from pressurized tritiated water through Inconel

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From a safety standpoint for fusion reactors, it is important to understand tritium permeation from the coolant through structural materials. In JA DEMO, the high-temperature and high-pressure water is expected to be used as a coolant. At high temperatures, hydrogen isotope gas can easily permeate through metal walls. However, the permeation behavior of hydrogen from water passing through metal walls is not completely understood. In this study, the tritium permeation behavior from pressurized tritiated water through Inconel 600 was investigated.

An experimental device with a double-tube structure was used. The inner tube made of Inconel 600 was filled with 33 kBq/cc of tritiated water, and pressurized with helium gas, and sealed. The outer tube made of SUS316 was filled with purified water, and pressurized with helium gas, and sealed. The double tube was heated to 300 °C, and then the pressure was increased to 15-17 MPa. The water in the outer tube was periodically sampled and tritium concentration was measured using a liquid scintillation counter.

At the first measurement of 5 days after the start of heating, tritium permeation to the water was observed. Then, tritium permeation rate gradually decreased and at around 40 days after, the permeation was not observed. It is considered that part of the tritium gas generated by the reduction reaction of tritiated water dissolved into the Inconel and permeated toward the water but the growth of oxide layer formed simultaneously with the reduction reaction decreased the dissolution rate with heating time.

Tentative

P2-094

Thermal desorption of tritium from stainless steel as the first wall used in the LHD deuterium plasma experiment

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From the viewpoint of fuel balance and tritium safety, tritium behavior in a fusion system must be clarified. To investigate the behavior and inventory of tritium in the vacuum vessel, tritium produced due to the D-D fusion reaction in the Large Helical Device (LHD) was used as a tracer. A part of the produced tritium is implanted into stainless steel, which is the first wall of LHD. To analyze tritium in the first wall, an imaging plate method and a thermal desorption method using an induction heating technique were applied. The tritium analysis was conducted on three first wall samples (clad plate of SUS316 and oxygen-free copper, thickness: ~1 cm, width: 13~22 cm, height: 20~23 cm) that were used in six LHD deuterium experimental campaigns. In the thermal desorption analysis, the heating temperature was set to 873 K or 973 K under the condition of nitrogen atmosphere at atmospheric pressure, and the sample was heated for 6 hours in an operation. As the preliminary results, the analysis of the three samples confirmed that the tritium inventory varied depending on the location in the vacuum vessel, that the sample with the carbon redeposition layer had a higher tritium inventory, and that the chemical form of the released tritium was predominantly in the molecular form. The presentation will also include the results of tritium surface distribution measured by imaging plates before and after the tritium thermal desorption operation.

Invitation

P2-095

Study on depth profiles of tritium in the LHD plasma-facing wall using a sputtering treatment

Suguru Masuzaki[1], Mingzhong Zhao[2], Miyuki Yajima[1], Masayuki Tokitani[1], Gen Motojima[1]

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In deuterium plasma experiments conducted in the Large Helical Device (LHD), the fast tritons produced by DD reactions collided with the plasma-facing wall. Here, the results of tritium (T) depth profiles measured in material samples placed on the plasma-facing wall after the experiments using sputtering (SP) processing with a glow discharge optical emission spectrometer (GD-OES) and the tritium imaging method (TIPT).

Deuterium plasma experiments were conducted at LHD from 2017 to 2022. In deuterium plasma, T was produced by a DD fusion reaction, and the total amount of T produced was about 26 GBq.From the exhaust gas analysis of the vacuum pump, it was estimated that about 50 % of the produced T remained in the LHD vacuum vessel. To determine where and how the residual T remains in the LHD vacuum vessel, it is helpful to know the depth profiles of T in the divertor plates used in the experiments and in the material samples installed on various positions on the first wall. A combination of TIPT and SP processing using GD-OES was conducted to obtain T depth profiles, and results will be presented.

Tentative

P2-096

Evaluating Electrode Material Properties in an Electrolysis-Based Tritium Extraction Process

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Kyoto Fusioneering, through an INFUSE project with Savannah River National Laboratory (SRNL), is evaluating cubic garnet-type $Li_7La_3Zr_2O_{12}$ (LLZO) solid electrolytes for use in a proposed tritium extraction method via direct electrolysis of LiT in molten Li and PbLi. Results from the project will be used to inform potential inclusion of the technique in the UNique Integrated Testing FacilitY flow loop (UNITY-1). The LLZO electrolytes provide stability in molten Li-bearing mixtures and eliminate the need for halide salts as an extraction solvent. Thermochemical stability and mechanical properties of the electrolytes were evaluated at temperatures relevant to the extraction process. Both Ta- and Ga-doped LLZO were found to maintain their high Li-ion conductivity cubic garnet structure upon immersion in molten Li and PbLi mixtures for up to 100 h at 350 °C. Thermal shock testing of the solid electrolytes was carried out; controlled heating and cooling of the electrolytes up to 500 °C had no adverse effects on the load-bearing capabilities during three-point bending experiments. The utilization of direct LiT electrolysis for the removal of impurities in molten Li was also explored. Electrode and process properties as well as impurity removal rates were better defined in this work. Key results to date will be discussed.

Tentative

P2-097

Statistical Shannon Entropy of Potential Energy Computations in Molecular Dynamics of Retention of Tritium in Graphene Material

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The main goal of building a magnetic fusion device is to recreate the interactions that happen at the molecular level between the plasma and the surfaces of the materials that are used to build it. In Tokamak reactors, it is a well-known problem to keep the wall structurally sound due to the retention of the effects of Tritium ions on the surface structures of graphene materials. The Shannon entropy of retention materials can provide insight into the behavior of tritium in graphene, which is a possible material for the Tokamak wall structure. In molecular dynamics, it is possible to calculate the Shannon entropy of potential energy levels and the quantity of tritium that is contained in the graphene structure. This will establish a framework for assessing the material reliability of graphene in future investigations by using the Shannon entropy level.This will help us address the retention problems in Tokamak reactors for future designs.

Tentative

P2-098

The spontaneous ignition of liquid lithium pools in controlled gaseous atmospheres

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[1]CIEMAT

The exploitation of liquid lithium in fusion-related facilities, including the IFMIF-DONES accelerator neutron source, entails inherent risks to both human health and the environment associated with potential lithium leaks and fires. To minimize the risk of lithium fires in these facilities to the lowest reasonably achievable level, it is imperative to develop a more comprehensive understanding of the mechanisms governing lithium ignition and combustion. The present work aims to provide a concise overview of the principal outcomes from past experimental activities conducted in different facilities investigating the spontaneous ignition of liquid lithium pools under controlled gaseous atmospheres. Furthermore, the most recent data collated during the first experimental campaign of the LiFIRE facility, recently commissioned at CIEMAT, have been outlined and evaluated in the context of the existing literature. This comparative analysis confirms that lithium ignition is a complex and hardly predictable phenomenon, which may be influenced by a multitude of factors, predominantly related to the characteristics of the metal specimen and to the experimental conditions employed.

Tentative

P2-099

Optimization and Thermal Performance Evaluation of Chamfering Structures for the ITER-like W/Cu monoblocks Divertor in EAST

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Since 2021, EAST has operated with an ITER-like tungsten (W) divertor, featuring W/Cu monoblock targets. Severe melting phenomena at the leading edges between neighboring cassette modules (inter-CMs) were observed on both the upper and lower divertor targets, despite the use of dual chamfer shaping structures. Although the misalignment between adjacent monoblocks in the lower divertor is maintained within 1 mm-significantly better than in the upper divertor-optimization and evaluation of chamfering structures for the lower W divertor are still necessary. To achieve this, thermal performance and chamfer optimization were investigated through finite element simulations of the W/Cu monoblocks. The results suggest that chamfering structures at inter-CMs can be designed with dimensions of 2 mm × 23 mm. For the leading edges of intra-CMs, the 1 mm × 1 mm dual chamfer structure, previously employed without damage, is also viable. This configuration would enable EAST to operate with a parallel heat flux of up to 120 MW/m^2 during typical configurations. With radial misalignment controlled within 0.6 mm, the divertor targets can withstand at least 190 MW/m^2. As the melting threshold increases with reduced incidence angle and radial misalignment, further enhancements to heat flux resistance through improved misalignment control and magnetic field configuration are also discussed. For standard operation scenarios, a configuration with the outer strike point on the horizontal target during high auxiliary heating discharges is found to be optimal. A comparison between ITER and EAST chamfer performance under varying heat flux conditions is also presented.

Tentative

P2-100

Evolution of microstructure and changes in hydrogen embrittlement resistance of CLAM steel after long-term aging

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The China Low Activation Martensitic (CLAM) steel has been proposed as a candidate structural material for nuclear fusion reactors. It is essential to investigate the changes in hydrogen embrittlement (HE) resistance of CLAM steel during thermal aging considering its service environment. In this study, samples of CLAM steel were subjected to thermal exposure in air at 550 °C and 600 °C for 1000 h, 3000 h, and 5000 h, respectively. The evolution of microstructure, including the evolution of subgrains, the growth of $M_{23}C_6$ carbides and MX carbonitrides, as well as the formation of Laves phase, were carefully characterized. These alterations in microstructure have a significant impact on the mechanical properties of the CLAM steel. The observed trend revealed an initial increase followed by a subsequent decrease in HE resistance upon prolonged aging durations. The mechanisms for changes in HE resistance were analyzed based on the orientation relationship between precipitates and matrix.

Tentative

P2-101

Structure and Composition Effects on Advanced Multicomponent-W for First Wall Applications

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Technically pure tungsten (W) is the presently chosen plasma facing materials (PFMs), however, this choice presents its own challenges when faced with harsh environments within a fusion reactors divertor region. Multicomponent tungsten (W) materials have been studied for their improved radiation tolerance and thermal properties for fusion reactor first wall environments where PFMs are subject to extremely high temperatures and particle fluxes [1]. For such applications, new material compositions are under development and need verification of performance in emulated fusion reactor environments. In this work, multicomponent-W of W-5%NbTa, W-5%NbTi, and W-10%Ti have been irradiated under various fusion relevant radiation conditions. Helium (He) and heavy ion irradiations have been conducted at a temperature of 800°C with an energy of 1 MeV at fluences of $1x10^{14}$ cm^{-2} and $1x10^{16}$ cm^{-2}. Multicomponent W thin films of the same compositions have also been investigated using in-situ TEM helium implantation and annealing at temperatures up to 1000°C in order to further investigate the nanoscale kinetics and effects of alloying elements on helium bubble formation and migration. Post irradiation, the accumulation of sub-surface He bubbles were characterized using scanning transmission electron microscopy (STEM) techniques. Additionally, the resulting change in thermal conductivity due to irradiation will be investigated using time-domain thermoreflectance (TDTR). The present results have been conducted with support from molecular dynamics (MD) simulations that offer a more in-depth analysis of the mechanisms behind He bubble growth and formation. References [1] O. El-Atwani et al., Sci. Adv. 5, eaav2002 (2019). DOI: 10.1126/sciadv.aav2002.

Tentative

P2-102

An Comprehensive Evaluation of Displacement Damage and Radioactivity of High Entropy Alloys Under Typical Fission and Fusion Neutron Spectrum

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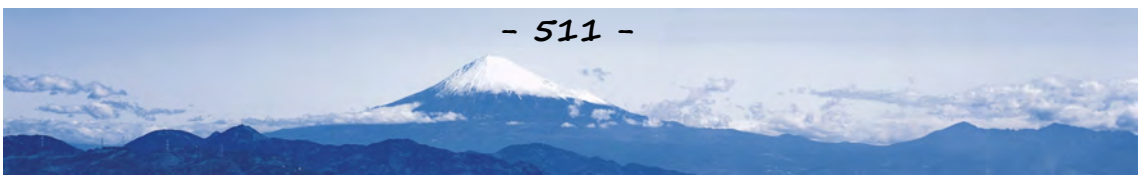
The structural materials served in nuclear reactors would undergo neutron irradiation, which can lead to the lattice displacement damage and environmental radioactivity. The two effects in general depend on the chemical composition of materials, neutron fluence, and neutron energy spectrum. In this study, several popular high-entropy alloys (HEAs) which had been demonstrated with promising radiation tolerance, were evaluated in perspective of displacement damage, helium production rate and radioactivity through an open-source Monte Carlo simulation code of FLUKA. The typical fission and fusion neutron spectrum were considered in this evaluation, and the JEFF-3.3 nuclear data library was adopted. The results informed that compared to reduced activation steel of Eurofer97, fcc-FeCoNiCr and bcc-NbZrTiV HEAs both exhibited a higher displacement damage rate but less in helium production under fusion neutron spectrum, while the neutron-induced radioactivity is also considerable. The above issues would be more sensitive in the fission situations. It means that the future compositions of HEAs should be modulated in the orientation of neutron application.

Tentative



P2-103

Data-Driven Approach for Evaluating Small Punch Test Results of Irradiated Materials Using Finite Element and Gaussian Process Regression Surrogate Modelling

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Accurate determination of material properties is crucial for fusion reactor component design and assessment. Conventional methods require extensive testing and strict safety procedures, making them impractical for irradiated materials. The Small Punch Test (SPT) is a promising experimental procedure, providing insights into tensile properties, fracture toughness, and irradiation-induced hardening. However, interpreting SPT results is not straightforward and requires extensive simulations for each test. Current approaches, such as those in ASTM E3205 and EN 10371, correlate specific SPT load-displacement curve points to isolated tensile properties, failing to comprehensively capture material hardening behaviour, especially in radiation environments. Moreover, these correlations are criticised for being material-dependent. Other methods, such as inverse analysis, require significant computational resources for each test.

This study proposes a novel, material-independent approach by developing a surrogate model using Gaussian Process Regression (GPR) and Finite Element (FE) simulations. This framework enables the reconstruction of full stress-strain curves from SPT data, providing a more detailed understanding of irradiation and temperature effects on material hardening. FE simulations were conducted using Johnson-Cook and Voce-Law hardening models, representing a wide range of materials, to train the GPR surrogate model. The surrogate model is capable of capturing the complex, nonlinear relationships between SPT results and material properties. The surrogate model predictions were validated against experimental SPT data from irradiated steels.

This advancement enhances SPT result interpretation, reducing reliance on extensive material testing and inverse analysis. It offers a computationally efficient, accurate, and easy-to-apply framework, essential for high-temperature, high-fluence radiation environments in fusion reactor materials research.

Template

P2-104

Simulation and Analysis of High-Efficiency Operating Modes for Helium-Hydrogen Separation Bed

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In Tritium Extraction System (TES), it is crucial that helium-hydrogen separation efficiencies of over 95 percent at low hydrogen isotope pressures. We've developed a low equilibrium pressure Zr0.8Ti0.2CO hydrogen storage alloy, then successfully loaded it onto a radial helium-hydrogen separation bed. In consideration of the operational parameters, a two-bed tandem three-bed alternating adsorption mode that is deemed to be suitable for radial flow helium-hydrogen separation beds has been analysed. In this mode, the outlet concentration of the precursor bed is used as the inlet concentration of the secondary bed. The outlet concentration of the secondary bed is set to meet the target adsorption efficiency. Due to the low amount of hydrogen adsorbed by the secondary bed, it does not need to be heated and desorbed after one cycle of operation. The secondary bed can be used as the precursor bed for the next cycle of operation. A penetration mathematical model was developed by coupling the kinetic equations for hydrogen absorption in materials, the momentum equations for fluid flow, and the mass transfer equations through the use of COMSOL software. The model analysis demonstrates that, under the assumption of 99% adsorption efficiency, this adsorption mode can reduce the amount of filling powder by approximately 45% in comparison with the conventional double-bed alternating adsorption mode. This mode ensures efficiency while minimising the quantity of alloy powder, thereby effectively diminishing the retention of hydrogen isotopes and facilitating the rapid heating desorption of hydrogen isotope gases.

Tentative

P2-106

Fracture behavior of laser weld joints with reduced activation ferritic/martensitic steel F82H

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Fiber laser welding is one of candidates of joining method for fusion reactor components with reduced activation ferritic/martensitic steel F82H. This method can suppress welds (FZ: Fusion Zone, HAZ: Heat Affected Zone), and is expected to ensure the radiation resistance. However, their mechanical properties in irradiation remain unknown because of their small geometry. Meanwhile, in terms of quality assurance, it is necessary to deeply understand dependency of the joint strength to the weld geometry, particularly after irradiation. Therefore, as a first step, this study aims to elucidate UTS (Ultimate Tensile Strength) and YS (Yield Strength) of FZ and HAZ in unirradiated environment, and to clarify the possibility of fracture in FZ or HAZ. A new approach for the evaluation on the deformation of laser weld joint is proposed utilizing SS-J3 specimen, which has been conventionally used for post-irradiation testing, and finite element analysis (FEA). Some SS-J3 specimens are modified and notched in FZ or HAZ so that measure engineering stress-strain relation (ESSR) until UTS for each area by in-situ observation with digital microscope and digital image correlation. Hardness and tensile tests are also conducted. UTS of FZ was greater than that of base material. The hardness of FZ and HAZ were also higher. Consequently, it is anticipated that base material begins to formerly fracture. In conference, UTS and YS of HAZ will be reported in addition to the fracture observation. Besides, fracture possibility of FZ and HAZ will be discussed considering fracture behavior predicted by FEA reflecting measured ESSR.

Tentative

P2-107

Transitioning from Tungsten to low-activation Steels using Laser-Directed Energy Deposition for Fusion Applications

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Joining plasma-facing components made of tungsten (W) with low-activation steels presents a significant challenge in the development of advanced divertor components for fusion reactors. This challenge arises due to the inherent metallurgical incompatibilities between tungsten and steel, including intermetallic phases, differences in thermal expansion coefficient and mechanical properties. These incompatibilities limit the applicability of conventional joining techniques such as welding or brazing, which often result in the formation of brittle intermetallic phases, cracks, or residual stresses that compromise the structural integrity and performance of the joint. To overcome these limitations, this study introduces a novel approach utilizing directed energy deposition, an additive manufacturing technique, to fabricate functionally graded materials (FGMs) that enable a seamless transition between tungsten and reduced-activation ferritic/martensitic steels. By gradually varying the composition across the transition region, FGMs mitigate abrupt changes in mechanical and thermal properties, thereby reducing the risk of failure due to stress concentration or phase incompatibilities. The transition pathway, consisting of vanadium- and iron-based alloys, is determined using computationally aided material selection, incorporating thermodynamics and diffusion kinetics simulations. This ensures a single-phase body-centered cubic transition, avoiding the formation of deleterious phases. Additionally, a comprehensive optimization framework involving fine-tuning of key processing parameters such as laser power, scan speed, hatch spacing, and layer thickness is developed to ensure the successful fabrication of defect-free, high-density FGMs. This work represents a significant advancement in the development of durable divertor structures for future fusion energy systems, while laying the groundwork for broader applications of FGMs in extreme environments.

Tentative

P2-108

Effect of Vacuum Environment on the Electron Beam Welding of Vanadium Alloys

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A major challenge with fusion welding of vanadium alloys, like V-4Cr-4Ti, for large structures such as the breeder blanket, is that these materials are prone to the pick-up of oxygen, hydrogen, nitrogen, and carbon at elevated temperatures. Electron beam welding (EBW) may be advantageous in limiting interstitial impurity pick-up compared to other welding methods because of its rapid cooling rates and narrow heat affected zones. In this study, bead-on-plate welding trials were conducted to identify the optimal parameters (accelerating voltage, beam current, focus, and welding speed) for standard electron beam welding and reduced pressure EBW. 44 different weld conditions were evaluated by non-destructive testing. For certain conditions, keyhole collapse, partial penetration, or sagging of the welds was observed. Charpy impact testing of the best effort welds from the standard and reduced pressure EBW conditions were conducted to compare the effect of vacuum environment on embrittlement. In addition, microstructure characterization by transmission electron microscopy is presented to elucidate the role of the Ti(CON) precipitates on weld embrittlement. Based on these results, the feasibility of reduced pressure EBW for the fabrication of vanadium alloy structures will be discussed.

Tentative

P2-109

Development of W/stainless steel first-wall component with ODS-Cu intermediate layer for high-temperature wall in QUEST

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Small-scale sample of new type of first-wall component of tungsten (W)/stainless steel (SUS) with an oxide dispersion strengthened copper (ODS-Cu) intermediate layer was successfully developed via Advanced Multi-Step Brazing (AMSB) in our previous work. The principle of AMSB is a repetitive application of the advanced brazing technique (ABT). The initial purpose of the ABT was to braze W to ODS-Cu (GlidCopR) with the Ni-11%P filler material. Later, we confirmed that the ABT is able to produce a very tough GlidCopR and SUS (GlidCopR/SUS) joint. The 0.254 mm thickness, e.g., W-sheet, was used in the previously developed small-scale sample, because W-sheet would have some advantages such as low hydrogen retention properties compared with vacuum plasma spray W (VPS-W) and certain levels of ductility.

In this study, middle-scale sample of W-sheet/SUS first-wall component with a GlidCopR intermediate layer; (W/GlidCopR/SUS) was developed by ABT for the high-temperature first-wall component of a spherical tokamak (QUEST) in Kyushu University. The first-wall armour material of QUEST consists of APS-W and aims to achieve steady-state discharge while maintaining the temperature of the first wall at ~500°C. In this study, heating tests of the developed middle-size sample of W/GlidCopR/SUS up to ~400°C and subsequent irradiation tests on the QUEST first-wall position around ~350°C during a plasma operation phase were conducted. The temperature of the entire W surface was uniformly distributed during a heating test. There was no specific damage on W/GlidCopR/SUS component such as delamination or cracks were not observed after the irradiation test in QUEST.

Tentative

P2-110

Development of Joining Technology and Analysis of Sustainability of FeCrAl Alloy as Inner Lining of Liquid Metal Blanket Piping

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FeCrAl alloys form protective oxide film through high-temperature oxidization, and are useful as structure materials for liquid metal blankets which are exposed to flowing liquid metal at temperatures above 773 K. However, from the perspective of radioactive waste, Al is an element that should be avoided in the reactor core as much as possible. We are investigating the application of FeCrAl alloys as clad materials to minimize their amount. In the present paper, our recent results of explosive welding of FeCrAl or FeCrAl-ODS alloy on a SUS316L steel are reported. We evaluated interdiffusion between FeCrAl(-ODS) and SUS316L by applying heat treatment to their joined materials.

Cladding materials were APMT (Fe-21Cr-5Al-3Mo, wt.%) and FeCrAl-ODS (Fe-15Cr-7Al-0.5Ti-0.4Zr-0.5Y2O3), while the base material is SUS316L. Clad materials were produced by explosive welding using several different charge ratios. Bonding interface microstructure and mechanical property were evaluated by SEM and 4-point bending test at RT and 773 K. Interdiffusion heat treatment was also performed on theclad material from 973 K to 1423 K, and their interface was investigated using EPMA and EBSD.

Clad material was successfully obtained by explosive welding. However, after heat treatment, interdiffusion zone of Al and Ni was observed on the joint interface. EBSD results show that the γ→α phase transformation occurred on the SUS316L side. Microstructural observation revealed the decrease in Al concentration and the formation of Fe-Al compounds on the FeCrAl alloy side. The mechanical properties of the material before and after annealing will be presented in the conference.

Invitation

P2-111

Mechanical characterization of copper-stainless steels joints in IFMIF-DONES oriented SSTT specimens

Nerea Garcia-Rodriguez[1], Marcelo Roldan[1], Daniel Rojas-Martinez[2], Francisco Canillas[1], Fernando Arranz[1], Fernando Sanchez[1], Edgar Leon-Gutierrez[1]

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Copper/stainless steel 304 joints are crucial elements in the components of fusion nuclear power plants, particularly for cooling terminals. The bond between these materials is essential for the construction and safety of fusion reactors. Thus, welding joining technologies of dissimilar materials must prioritize low temperatures techniques to minimize deformation and maintain a small heat-affected zone (HAZ).

Controlling these parameters enhances the homogeneity of the base metal interface. This study evaluates joints produced by electroforming (ELF) and electron beam welding (EBW). Small specimen testing technology (SSTT) is used to assess mechanical properties, including elastic modulus, yield strength, ultimate tensile strength, and elongation. SSTT involves testing small samples to determine their suitability for fusion reactors. In this work, tensile tests have been conducted on samples with different thicknesses to analyze its influence with the mechanical properties. Moreover, the tensile tests were performed at room temperature and at 200°C to evaluate temperature effects on the joint area.

Additionally, Vickers hardness tests were conducted to compare hardness differences between base metals and the HAZ, assessing the transition between these zones. The microstructure of the joint was examined using scanning electron microscopy (SEM) to determine if modifications occurred during the joining process.

This methodology aligns with the future applications of the IFMIF-DONES irradiation source, which will require specialized test specimens to ensure material performance under fusion reactor conditions. By employing these techniques, the study aims to advance the reliability and safety of components in fusion energy systems.

Tentative

P2-112

Qualification tests for tube-to-tube welding of CuCrZr to Inconel 625

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In the monoblock divertor target, the heat sink tube of each plasma facing unit (PFU) requires extension by a stainless steel tube. This enables connection to the manifolds through TIG welding. The stainless steel tube is connected to the CuCrZr heat sink tube via an Inconel 625 using electron beam welding. To guarantee the high thermal and mechanical properties of the welds, non-destructive testing, tensile testing, and fatigue testing should be carried out on the welded tubes for qualification purposes. According to ITER's technical specifications, the weld must pass visual inspection, radiographic examination, metallographic examination, bend testing, tensile testing, and helium leak testing after the rotary bending test. The following findings were obtained: 1) The quality of the tubes serves as the foundation for the weld; 2) Electron beam welding (EBW) needs to be optimized to ensure satisfactory performance during the bending test; 3) The coaxiality of the two tubes after welding is a crucial factor affecting the result of the rotary bending fatigue test, as vibrations may occur; 4) Post-weld heat treatment is also a key factor. Qualification tests are essential for preventing leakage in the divertor during operation. Nevertheless, it is equally necessary to define the test items and acceptance criteria based on the potential loading conditions of the divertor PFU.

Tentative

P2-113

Conceptual design of welding between vacuum vessel sectors for a torus fusion device

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[1]National Institute for Fusion Science

On a design of a torus fusion device such as tokamak and helical systems, a vacuum vessel must be equipped to keep plasma space and to install blanket blocks and diverters. To increase its neutron shielding ability, the VV will have a double-wall structure and the space between walls will be filled with water. On the inside surface of the VV, the blanket blocks will be fixed with housing and the diverter will be installed at the bottom the VV structure. Both of the blanket and diverter (in-vessel components) will be replaced after a certain D-T operation period, for the materials of these components do not endure the hard neutron irradiation. To replace the in-vessel components, high installation precision is required. The most concern is the welding residual deformation on the welds of sector-sector joints. The material of the VV will be 316 stainless steel and the thickness of around 60 mm will be required, when the ITER size device is assumed. The welding procedure is considered to be TIG welding, for all welding position at the field is necessary. Since the residual deformation occurs due to the temperature difference between the splice plates and the VV walls, the temperature rise of the splice plates must be kept as low as possible. In addition, the splice plates must have high stiffness to carry the compressive residual stress. Based on the fundamental discussions, an image of the weld joint shape is proposed along with the new blanket design concept.

Tentative

P2-114

Effect of cold rolling and annealing temperature on microstructure of diffusion bonded $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$ / F82H joint

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Nowadays, the demand for fusion reactor materials has reached unprecedented levels. High-entropy alloy (HEA), Co-free $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$, offers good irradiation resistance. However, the application of HEAs in the nuclear industry remains limited due to incomplete research on the welding. Cold rolling (CR) is a process that refines the grain size and introduces dislocations. This study focuses on investigating the role of CR in diffusion behavior in $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$/F82H joints.

$Cr_{0.8}FeMn_{1.3}Ni_{1.3}$ were arc-melted, homogenized at 1160 °C for 24 h, then cold-rolled to 20% and 30%, followed by stress-relief annealing at 1000 °C for either 2 or 4 h. Diffusion bonding with F82H steel were successfully welded using the Spark Plasma Sintering (SPS) technique at 800 °C. Analysis of microstructures (EPMA, TEM, SEM, EBSD) were conducted for joints. Mechanical properties were evaluated by tensile testing with miniature specimens (SS-J1).

CR largely refined the grain size of $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$. CR 20% followed by 2 h annealing of HEA ingots resluted in refinement of grain size (151 μm) and increase of dislocation density (8.2×10^{11} m^{-2}) compared to as-homogenized specimen. Increasing the CR degree and annealing time to 30% and 4 h resulted in the finest grain size (107 μm) and the largest diffusion distance (26 μm). This process also yielded the highest UTS (460 MPa) and ductility (~14%) due to optimized recrystallization, refined grain size. Furthermore, the Cr-rich BCC precipitate was observed near the interface, the precipitates show a tendency to precipitate on dislocations, which indicates that dislocation provides pipe diffusion channel for Cr.

Invitation

P2-115

Corrosion of austenitic steels (316L, 316Ti, 310S and 317) and their welding in molten tin at 900°C

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Liquid metals are being investigated as plasma-facing materials in fusion devices due to their potential as breeding blankets, plasma-facing walls with benign plasma-interaction properties, neutron shields, and working fluids. Renaissance Fusion aims to experimentally demonstrate lithium-based liquid walls at temperatures of up to 900o in simplified stellarator geometry. In preparation for that, it is currently developing a magnetohydrodynamic experiment with a free-surface, hot liquid tin (Sn) flow in cylindrical geometry. Liquid metals such as Li and Sn are highly corrosive at high temperatures. Thus, the selection of high temperature corrosion resistant materials and suitable joining techniques are crucial. This study examines the corrosion behaviours of various grades of austenitic stainless steels and their welding, submerged in a liquid Sn bath at 900°C for 24 hours under a pure argon gas atmosphere. The findings indicate that the dissolution of the specimens and the penetration of tin into the test specimens varied significantly among the different austenitic steel grades. The segregation of molybdenum-rich areas was detected in both the base metal and weld zones in all the steel specimens. These insights are critical for understanding the corrosion mechanisms of steels, thus guiding the material selection and their joining techniques for use as liquid metal facing components.

Tentative

P2-116

Dislocation - mediated rapid atomic diffusion in diffusion bonded reduced-activation $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$ high entropy alloy/316 stainless steel joints

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Materials in fusion reactors suffer from irradiation-induced swelling and mechanical degradation, necessitating the development of advanced materials. High entropy alloys (HEAs) offer superior properties, with the Co-free $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$ HEA demonstrating good irradiation resistance. However, its joining characteristics remain underexplored. Diffusion bonding with 316SS is a promising method for nuclear applications. Dislocations, acting as "channels" for pipe diffusion, could enhance bonding efficiency. This study investigates the role of dislocations in diffusion behavior in $Cr_{0.8}FeMn_{1.3}Ni_{1.3}$ HEA/316SS joints.

HEA ingots were arc-melted, homogenized, cold-rolled (50%), and annealed at 1000 °C for 4 hours. Secondary cold rolling (0-20%) was applied to vary dislocation density. Diffusion bonding was performed at 900 °C using spark plasma sintering. Microstructural analysis (EPMA, TEM, SEM, EBSD) were conducted on the joints. Mechanical properties were evaluated via nanoindentation and SS-J1 tensile testing.

At a 20% deformation rate of the HEA base metal, the dislocation density increased to 7.8×10^{13} m^{-2}, resulting in an Fe diffusion distance of 8.3 μm in joint, while the diffusion coefficient rose to 2.15×10^{-14} m^2s^{-1}. Meanwhile, an increase in deformation rate led to a notable 41% reduction in the activation energy of the diffusion layer in HEA side, indicating that higher dislocation density effectively lowers the energy barrier for atomic migration. Furthermore, Cr and Mn oxides were observed at the interface, with Cr_2O_3 formation becoming more prominent at higher deformation rates. Mechanical testing revealed a 9% increase in the tensile strength of the joint, attributed to improved atomic intermixing and reduced local stress concentrations.

Tentative

P2-117

The novel methods for achieving high-strength W/steel bonding via regulating interface diffusion behavior or induced interface alloying

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Heterogeneous W-steel joints will produce brittle intermetallic compounds and significant residual stress, which limits the manufacture of breeder blankets. Adding a Cu interlayer is the mainstream solution to release the thermal stress and inhibit brittle phase formation. However, as a binary immiscible metal system, realizing the alloyed Cu/W bonding is a new challenge. We propose a composite joining technique of casting and brazing to realize the Cu/W microalloying interface by overcoming the positive generative energy and opening the interatomic diffusion channels. The superior tensile strength (264 MPa) was achieved, and the joint survived a 1000-high heat flux of 1 MW/m2 without "hot spots" and cracking. We Successfully manufactured the U-shaped first wall (FW) mock-up with W armor by brazing. In addition, we propose a method to form a Cu-based and Fe-based composite transition layer in situ to achieve W/steel directly bonding via introducing Cu-Ge filler by brazing. Through regulating the interfacial diffusion behavior, the Fe and Cr from the steel were across the Cu-based fillers and deposited onto the W side, forming a Cu-based layer and a Fe-rich transition layer to reduce the thermal expansion mismatch, mitigating the thermal stresses and achieve the W/Fe alloyed interface. And through multi-component solid solution into the Fe2W brittle phases to promote lattice distortion and form stacking fault to achieve the in-situ modification of Fe2W. This research achieved an unprecedented high shear strength of 380 MPa. Our work provides new strategies for W/steel or W/Cu interlayer/steel high-strength bonding and achieves excellent mechanical properties.

Tentative

P2-118

Evaluation of grain boundary strength of tungsten materials using micro-cantilever bending test

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Tungsten is recognized as a key material for plasma-facing components in fusion reactors. To improve low-temperature ductility of tungsten, thermomechanical processing (e.g., rolling) has been applied to form texture with fine grains. However, recrystallization of tungsten used in divertors with high heat flux is considered to cause grain boundary cracking. Moreover, high-energy particle irradiation can lead to further embrittlement. Therefore, evaluating the grain boundary strength of tungsten before and after recrystallization/irradiation is meaningful for predicting and mitigating grain boundary embrittlement. We conducted micro-cantilever bending tests at room temperature on micro-sized specimens containing a single grain boundary which were fabricated by focused ion beam on recrystallized, rolled, and proton-irradiated tungsten materials. Characteristics of grains were identified by electron backscatter diffraction (EBSD) to determine the locations of bending specimens. The displacement-bending stress relationship was successfully measured using nanoindenter. As a result, the recrystallized tungsten specimens fractured at lower stress (~1.1 GPa) compared to the rolled specimens with a long transverse direction (~1.4 GPa [1]) and exhibited grain boundary fracture. The rolled specimens with a longitudinal rolling direction showed the highest strength (~3.5 GPa [1]), resulting in anisotropic fracture behavior. In the proton-irradiated recrystallized specimens, crack propagation was observed not only at a grain boundary but also within a grain. The differences in grain boundary fracture mechanisms in tungsten arising from the degree of recrystallization and the effects of irradiation on the fracture mode change will be discussed at the conference.

[1] X. Wu, Ph.D. Thesis, Tohoku University, Japan, 2024.

Tentative

P2-120

A Nanoindentation Study on Characterization of Irradiation Damage Depth in RAFM steel

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The fusion reactor operates at extremely high temperatures and high-energy neutrons collide with the plasma-facing materials (PFMs) at high velocity. These neutron collisions cause surface damage to the materials, ultimately leading to material degradation and failure. As a result, many researchers are actively developing reduced activation ferritic-martensitic (RAFM) steel capable of withstanding such extreme environments. To ensure the reliability of RAFM steel, it is essential to quantitatively analyze neutron irradiation-induced damage Typically, transmission electron microscopy (TEM) is used to measure the irradiation damage depth. However, TEM analysis has limitations, including complex specimen preparation and lengthy processing time. In this study, we propose an alternative approach using the nanoindentation technique to characterize irradiation damage depth without TEM observation. We continuously analyzed hardness variations from the surface to the interior of He-ion irradiated RAFM steel specimens subjected to different irradiation doses and temperatures. When the irradiation temperature was at room temperature, the maximum irradiation damage depth was estimated at the point where the hardness values became similar to those of the unirradiated RAFM steel. At elevated irradiation temperatures, the maximum irradiation damage depth was determined as the point where the hardness values converged. Using this concept, we successfully measured irradiation damage depth using only nanoindentation, achieving results consistent with previous studies.

This work was supported by National R&D Program through the National Research Foundation of Korea (NRF) by the Korea Government (Ministry of Science and ICT) (No.RS-2022-00156205)

Tentative

P2-121

Temperature-dependent defect evolution of displacement damage under heavy ion irradiation in CVD tungsten.

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This study comprehensively evaluates the irradiation defect evolution of chemical vapor deposited tungsten (CVD-W) with columnar grains (grain size ~40 μm, [001] preferential orientation), with comparative analysis against ITER-like W. W samples were irradiated by 6 MeV Cu^{3+} ions to 0.6 dpa across RT-973 K, with systematic characterization of radiation-induced microstructure and mechanical response. Advanced specimen preparation combining focused ion beam (FIB) with flash polishing achieved artifact-minimized TEM lamellae for accurate defect analysis.

Nanoindentation results integrated with Nix-Gao modeling demonstrate convergence of irradiation-hardened states: CVD-W's initial 24% hardness deficit (4.6 vs. 5.7 GPa) diminishes to <5% difference post-irradiation (6.3 vs. 6.5 GPa at RT), with both materials reaching 7.1, 7.4 GPa at 973 K. The pop-in events were also notably suppressed by irradiation-induced defects.

TEM results show that the lower initial dislocation density in CVD-W leads to the lower hardness. A dense damage layer of approximately 2 μm depth forms after irradiation, indicating that irradiation-induced defects primarily govern the hardness of CVD-W at RT. As the irradiation temperature increases (RT-973 K), the total dislocation density decreases by approximately 31%, while voids were observed in samples irradiated at 573 K. Between 573 and 973 K, void diameter remains stable at approximately 1.1 nm, with void density increasing by about 23%, suggesting that vacancy-type defects are the main contributors to high-temperature irradiation hardening.

The irradiation hardening response and similar defect evolution trends between CVD-W and ITER-like W validate comparable radiation tolerance of CVD-W, providing important insights for ITER material selection.

Tentative

P2-122

Microstrcture and nano-hardness evolution in tungsten under strain field

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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 strain filed is not sufficient. In this study, pure tungsten specimens with two different surface strains were chosen as original specimen to investigate the effect of strain field. Implantation experiments of a H/He mixed beam were conducted. After irradiation, the irradiation induced hardness change and microstructures were measured by nano-indentation and transmission electron microscopy (TEM). In terms of nano-indentation results, the surface strain state affected not only the nano-hardness of the original specimens, but also the hardness increments and evolution induced by irradiation. In terms of TEM results, the surface strain state affected the growth of irradiation induced bubbles and dislocation loops. The experimental results were in accord with the simulation results of cascade collision process. They both indicate the compression strain would restrain the bubbles and dislocation loops, and then the nano-hardness increments.

Template

P2-123

Probing Helium-Induced Defect Evolution in Ferrous Alloys for Fusion Application: Insights from Positron Annihilation Spectroscopy and Multi-Energy Ion Implantation

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To comprehend and predict the early-stage radiation aging and helium (He)-induced defect evolution of structural materials for future fusion facilities, this study presents a systematical positron annihilation lifetime spectroscopy (PALS) characterization of ferrous alloys exposed to harsh radiation conditions. By integrating both high-energy, high-fluence He ion implantation and spallation neutron-irradiation experiments, we provide both qualitative and quantitative insights into He-vacancy clusters and propose a novel semi-empirical model linking positron annihilation parameters (lifetime, trapping rate) to defect characteristics (radius, He/V ratio) and irradiation conditions. The obtained results demonstrate the equilibrium He/V ratios in radiation-induced cavities are constrained to $\leq$ 1 for small clusters (few vacancies) and 0-0.1 for cavities > 1 nm, contradicting theoretical predictions of He/V > 1. The model explains that temperature-driven evolution dominates He-filled cluster behavior, at low temperatures (1 within operational temperature ranges (25 - 300 °C). The semi-empirical framework, validated against TEM and irradiation data, bridges ion-implantation simulations and neutron-irradiation effects, offering a cost-effective tool for predicting material performance in fusion environments. From an engineering perspective, the study offers insights into alloy design by suggesting temperature ranges where maintaining lower helium-to-vacancy ratios may contribute to reduced embrittlement risks.

Tentative

P2-124

Towards reliable evaluation of ion-irradiation induced mechanical property change based on micro/nano-mechanical testing

Ke Jin[1], Jiaohui Yan[1], Shang Chen[1], Lei Li[1], Xun Guo[1], Yitao Yang[2], Bin Long[3], Yankun Dou[3], Xinfu He[3], Wen Yang[3]

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Evaluating irradiation induced mechanical property degradation is crucial for nuclear structural materials. While ion irradiation is widely used for fundamental studies on irradiation effects, mechanical property testing on ion-irradiated materials remains challenging due to their shallow damage layers. Micro/nano-mechanical testing techniques have gained traction in recent years for ion-irradiated materials, but their validity and reliability are still debated, particularly due to issues like size effects. This presentation reports our recent progresses on applying four types of micro/nano-mechanical testing methods-nanoindentation, micro-pillar compression, micro-cantilever bending, and micro-tensile testing-to various ion-irradiated alloys. We focus on addressing the critical factors influencing the assessment of irradiation-induced hardening and embrittlement. For cross-sectional nanoindentation, we emphasize the importance of considering the lateral span of deformation zones to unify the hardness-dose correlation obtained from the samples irradiated to different fluences. Additionally, the dose-dependent pile-up effects and indentation size effects can be corrected by correlating the characteristic correction parameters and hardnesses. For micro-pillar compression, we show that ion irradiation not only affects the pillar size effects that may cause the arbitrariness on irradiation hardening evaluation, but also may result in deformation concentration within a small fraction of pillars even though the taper angles are minimized. Furthermore, in order to improve test repeatability, we introduce a refined pre-crack fabrication procedure to better control crack geometry for micro-cantilever bending tests, and recommend a specimen shape for micro-tensile tests that can effectively mitigate stress concentration.

Tentative

P2-125

Irradiation Induced Defects in W-Re Alloys Studied by Positron Annihilation Spectroscopy

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Tungsten is applied as a plasma-facing material in fusion reactors due to its superior material properties such as high melting point, high thermal conductivity and low erosion under reactor operating conditions. In addition to radiation damage, 14 MeV neutron irradiation of tungsten creates transmutation elements, mainly rhenium.

In this study, we investigated the effect of rhenium on the defect characteristics. Defects were created by ion irradiation in W-Re alloys with 1, 3, and 5 at.% Re content and results were compared with those of pure tungsten. Room temperature and 1350 K tungsten self-irradiation and rhenium irradiation were performed with an ion energy of 2 MeV. A fluence of 10E20 ions/m^2 resulted in about 50 displacements per atom (dpa). The maximum concentration of 1% Re is reached at a penetration depth of about 150 nm. We used depth-resolved positron annihilation spectroscopy (PAS) to observe the irradiation-induced creation of vacancies and vacancy-clusters up to a depth of 400 nm corresponding to a mean positron implantation energy of 30 keV. Coincidence Doppler broadening spectroscopy (CDBS) of the 511 keV annihilation line is used to perform element-sensitive measurements of vacancy-type defects. In particular, CDBS has allowed us to investigate the chemical surrounding of vacancies and hence to test theoretical predictions such as the possible stabilization of rhenium-vacancy complexes in tungsten or rhenium precipitation around voids.

Tentative

P2-126

Retarded recrystallization of tungsten by helium irradiation: simulation and experiment

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Recrystallization has been identified as an important issue which may reduce the strength of tungsten (W) and deteriorate its thermal shock resistance in ITER. It has been recognized helium (He) in W could retard recrystallization kinetics and improve the damaging threshold under thermal shock. In this work, a combined experimental and simulation approach is used to investigate the necessary conditions for retarded recrystallization to occur and identify the mechanisms at play.

By annealing He-irradiated W up to 1900 K, in rolled W samples exposed to a He plasma with a fluence of 1.4×10^26 m^-2 at 500 K an increase of recrystallization temperature of ~50 K was observed. Much stronger retarding effects were found after 40 keV He ion beam irradiation with doses ranging from 5×10^21 m^-2 to 1×10^23 m^-2 at 673 K, with a maximum increase of recrystallization temperature of ~300 K. The He concentration dependence of recovery was measured using quasi in-situ electron scanning microscope for the first time. New phenomena including large protrusions termed “thermal ridges” near grain boundaries and abnormal oversized elliptical bubbles (> 200 nm in diameter) in {110} grains were observed. Molecular dynamics and phase field simulation were carried out to understand the microstructure evolution.

This study clearly demonstrates that helium leads to significant changes in the kinetics of both recovery and recrystallization in W at elevated temperatures expected in the ITER. Implications of plasma and ion induced retarded recrystallization on the operational budget of the ITER divertor will be discussed.

Invited

P2-127

Plasma facing materials in inertial and magnetic confinement: peculiarities of radiation environment and radiation-induced damage

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The development of plasma facing martials (PFMs) able to withstand the harsh conditions (large thermal loads and radiation-induced damage) is one of the key parameters for nuclear fusion to upscale to commercial power plants both in the inertial (ICF) and in the magnetic (MCF) fusion approaches.

It is well known that the behaviour of materials under irradiation strongly depends on radiation-environment (temperature, flux, nature of radiation, pulsed or continuous...).

In this contribution the radiation environment that PFMs will withstand under the MCF and ICF, both in the indirect (ID) and direct (DD) drive approaches, will be illustrated, specially focussing on the differences between them [1,2]. Subsequently, the peculiarities of the radiation-induced damage associated to the radiation conditions will be shown. Finally, the capabilities and limitations of existing experimental facilities to mimic radiation environment [3], together with the need for validating and improving computer codes to predict material behaviour under those extreme conditions, up to date unreachable in experimental facilities, will be discussed [4-6].

The goal of this work is to push a discussion on the potential short-time scheduled experiments by using the available technology to accelerate responses to the design problem along the present roadmap of private-public development.

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Invited

P2-128

Anomalous Exothermic Behavior of Bubbles in Metals Induced by Electron Irradiation

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Anomalous exothermic phenomenon (AEP) under bubble irradiation in solids has captured widespread interest in the energy field. AEP was discovered during the transmission electron microscope (TEM) observation of deuterium-implanted aluminum. Under the irradiation of TEM electron beam, deuterium (D) bubbles led to AEP, causing the surrounding single-crystal aluminum to instantaneously transform into polycrystalline. The released heat was estimated as about 160 MeV. AEP has been discovered in different gas bubbles such as hydrogen, helium, argon, and neon. The influencing factors of AEP can be summarized as the bubble morphology and pressure, gas type and density, electron beam energy, and irradiation time. The factor interrelationships and the AEP threshold are discussed. In conjunction with experimental phenomena and simulation, the mechanism of AEP is speculated to be the plasma formation within the gas bubble. The generation of this substantial energy only requires the irradiation of bubbles with a beam of energetic particle. Such convenient, efficient, and controllable energy source is attractive. AEP may provide a potent new energy form.

Tentative

P2-129

Constitutive Modelling of Ratcheting Behaviour of Eurofer97 at High Temperatures

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Ratcheting, the progressive accumulation of plastic strain under cyclic loading, is a critical failure mode influencing the long-term structural integrity and lifespan of components. Its consideration in the design process of in particular pressurized components for high-temperature applications is crucial for their reliable operation. This study numerically investigates the ratcheting behaviour of Eurofer97 steel at room temperature, 450 °C, and 550°C under various loading conditions, including stress magnitude, stress ratio, and stress rate. Using experimental low-cycle fatigue (LCF) and ratcheting datasets, a constitutive model is developed and calibrated to accurately capture mechanical behaviour considering ratcheting effects. A series of numerical simulations through MATLAB code are conducted to evaluate the influence of stress amplitude, mean stress, and stress ratio on ratcheting behaviour. The results show that symmetrical cyclic loading with minimal mean stress significantly reduces ratcheting, whereas asymmetric loading with nonzero mean stress promotes higher ratcheting effect. Predictably, higher stress amplitudes accelerate ratcheting, while lower amplitudes and stable cyclic conditions lead to negligible accumulation. Additionally, the relative influence of each of those stress parameters on ratcheting behaviour can be quantified and compared through sensitivity analysis using Sobol' indices. This study provides valuable insights into ratcheting mechanisms and influencing factors. The findings contribute to optimizing structure and designing loading conditions that minimize ratcheting, ultimately enhancing the long-term mechanical performance and service life of Eurofer97 steel components.

Invitation

P2-130

Optimization and feasibility study of radiation shielding materials for fusion reactors by radiation transport simulation

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In stellarators or compact spherical tokamaks, installation of radiation shielding materials (RMSs) is essential to ensure the functionality of superconducting coils, coil cassettes and sensor components that contain highly neutron-reactive elements, such as silicon, nickel, and molybdenum. Although neutron fluence and energy deposition have been calculated in previous studies, the durability and lifespan of the RSMs under neutron irradiation conditions remain insufficiently explored. Furthermore, prior numerical analyses focused on a limited range of materials and lacked comparative evaluations.

In this study, we investigated candidate RSMs, assessed their shielding performance, and evaluated their lifespan in terms of nuclear functionality. Radiation transport simulations were conducted using PHITS with a simulation geometry based on the spherical tokamak. Evaluations were performed near the central coil position and the breeding position of tritium. As previously established, materials containing hydrogen, carbon and boron were selected for calculation (e.g., B_4C, ZrH_2, WB, WC). Each neutron flux at multiple positions was converted to radiation transparency, heat generation due to radiation, neutron-material interactions, displacement damage, and transmutation products. Additionally, to assess the impact on reactor performance, the tritium breeding ratio was calculated. As the preliminary results, WB and WC showed highly shielding performance because of the reaction with thermal neutrons by boron and carbon, and scattering reaction with fast neutrons by tungsten. Energy deposition at the coil position also attenuated because tungsten shields gamma-rays effectively. Our findings contribute to the selection of optimal shielding materials for compact fusion reactors, considering both radiation protection and operational feasibility.

Tentative

P2-131

Advanced Design and Integrated Manufacturing of Low-Activation Steel/Tungsten Functionally Graded Materials for Plasma-Facing Component Applications

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In China Fusion Engineering Test Reactor (CFETR) and future fusion demonstration reactors (DEMO), tungsten (W) serves as the primary plasma-facing material (PFM), while reduced activation ferritic/martensitic (RAFM) steel has been identified as the preferred structural material. The material integration between PFMs and structural materials in plasma-facing components (PFCs) has consequently emerged as a pivotal challenge in fusion energy development. Although conventional high-temperature diffusion bonding techniques have achieved physical connections between PFMs and RAFM steel, it is significant challenges persist. The substantial thermal expansion coefficient mismatch between tungsten (4.5×10^{-6}/K) and RAFM steel (~12×10^{-6}/K) induces severe interfacial thermal stresses under high heat flux conditions (typically 10-20 MW/m). This differential expansion leads to interfacial deformation mismatch and potential delamination, critically compromising structural integrity and operational longevity.

To address the critical issue of brittle intermetallic phase formation at W/RAFM interfaces and its detrimental effects on component lifetime, this study proposes an innovative graded material system: RAFM steel-V-W functionally graded materials (FGMs). Through advanced additive manufacturing techniques with optimized printing strategies and parameter control (e.g., laser power 200-400 W, scan speed 0.8-1.2 m/s), we successfully fabricate components demonstrating both mechanical robustness and composition gradients. Novel metallurgical bonding with excellent bond strength is achieved at all interfaces and no brittle phase generation is observed at the interfaces. We also establish fundamental structure-property relationships, elucidate interfacial evolution mechanisms under operational conditions (cyclic thermal loading), and ultimately enable monolithic fabrication of PFCs. Those findings are expected to provide crucial technical support for PFC development in CFETR and DEMO projects.

Tentative

P2-132

Modelling fracture properties of single-crystal Beryllium as a function of crystal orientation

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Tentative

Beryllium (Be) plays a critical role in nuclear fusion reactors, being essential as part of the fuel cycle. ITER incorporates a Helium-Cooled Ceramic Breeder Blanket with alternating layers of beryllium and lithium, cooled by helium at 600-900°C. The heat variation during its operation results in cycles of expansion and contraction inducing high incompatibility stresses that can lead to fracture or dust inside the reactor. Consequently, understanding the fracture behaviour is essential for maintaining the structural integrity of blanket modules.

The fracture of Be is affected by slip, which varies with crystal orientation. Experiments conducted by UKAEA, involving the loading of single crystal notched cantilevers at various crystal orientations, revealed distinct fracture responses depending on orientation: brittle cracking, plastic deformation, and some intermediate case. Moreover, crack propagation was observed to follow the direction of the most active slip system.

For the first time, the Crystal Plasticity Oxford-UMAT code, originally developed for analyzing STEP reactor materials, has been integrated with the Extended Finite Element Method (XFEM). This integration facilitates the examination of the role of plasticity and strain hardening in fracture propagation, analyzing its properties in relation to dislocation densities. We also investigate the independence of fracture properties from the active slip systems.

The findings will aid in predicting fracture initiation and propagation in Be polycrystals, facilitating the optimization of crystal orientation. This optimization can reduce fracture speed, thereby enhancing the integrity of the Be components.

P2-133

Advanced deep learning framework for multi-scale prediction of mechanical properties from microstructural features in polycrystalline materials

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The intricate relationship between a material's microstructure and its mechanical properties remains a significant challenge in the field of materials science. This study introduces a novel deep learning framework designed to predict mechanical performance from both global and local perspectives. Using the dual-phase Ti-6Al-4V alloy as an example, we first predict stress-strain curves and yield strength under complex microstructural conditions to characterize global mechanical behavior, followed by analyzing the distribution of local stress fields and stress concentration phenomena. To achieve this, we employ an Improved Graph Attention Network (IGAT) to effectively capture complex intergranular relationships, enabling precise global performance predictions through node features and graph structure information. Simultaneously, a 3D Conditional Denoising Diffusion Probabilistic Model (3D-cDDPM) is developed for local stress field analysis, generating detailed stress distributions through iterative denoising while identifying stress concentrations in critical microstructural regions. In fatigue prediction, this model accurately pinpoints crack initiation sites (e.g., stress concentration zones at α/β phase interfaces). The results demonstrate that this framework can effectively predict multiscale mechanical responses across diverse microstructural configurations. This work advances methodologies for material design and performance optimization, providing critical insights into the integration of computational modeling and materials science.

Tentative

P2-134

Development of an open platform for hydrogen isotope properties data

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The properties of hydrogen isotopes and isotopologues, such as the thermodynamic and transport properties of solid, liquid, and gaseous hydrogen and the diffusivity, solubility, and permeability of hydrogen in materials, are crucial in nuclear fusion engineering for designing efficient fuel cycle processes, ensuring high radiation safety in tritium handling, and maintaining the integrity of structural and functional materials. However, researchers often struggle to find reliable data, leading to inconsistencies in the hydrogen properties used in previous studies. Moreover, there remains a significant lack of tritium properties data.

To address this issue, we have initiated the development of a comprehensive hydrogen properties database with a web interface that allows easy access. Users can also contribute and validate new data, ensuring a continuous expansion of the database and a progressive improvement of its reliability. Furthermore, missing properties can be estimated through built-in fitting tools, including machine-learning regression models, by integrating experimental, theoretical, and computational data, thereby enhancing the applicability of the database in research and engineering activities. In this presentation, we will explain the current development status, showcase the key features of our platform, and demonstrate its practical use. Future development plans will also be discussed.

Tentative

P2-135

Measurement of water activation with collimated germanium detectors at the JSI KATANA facility

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Invited

Water is considered a promising coolant material for fusion reactor components, such as breeding blankets. As a coolant, water will be exposed to an intense flux of neutrons at DT peak energies which induce radio-active isotopes of nitrogen and oxygen. Although they have short half-lives of the order of seconds, they may lead to significant dose rate levels downstream of the cooling loops. Therefore it is desirable to predict such dose rate levels, but this is a complex task that requires coupling radiation transport codes with inventory and computational fluid dynamics codes. KIT participates in a collaboration funded by EUROfusion which aims at activation measurements at the KATANA water loop constructed at the TRIGA reactor of JSI in Ljubljana to provide experimental data for checking and verification of multi-physics numerical methods for the calculation of dose rate maps around water loops. KIT has done gamma-ray measurements at KATANA with collimated high-purity germanium detectors. An important part of data processing is numerical simulations of the detection geometry. We do these simulations with the code PHITS (Particle and Heavy Ion Transport code System). We used the accelerator-based DT neutron generator at TU Dresden, which can produce up to 1E12 DT neutrons per second, to gain additional experimental data on well-characterized irradiation geometries used to validate the PHITS simulations. Here we will present the results of our measurements at the KATANA loop and

with the neutron generator as well as the numerical simulation methodology of gamma field and detector.

P2-137

MatDB4Fusion: A new initiative to collect, merge, and leverage fusion material properties data

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Tentative

Major scientific breakthroughs and increasing investments have led to a drastic acceleration in fusion energy research. However, one of the largest factors severely limiting the anticipated pace to realize fusion on the grid is the development of a set of basic fusion materials, qualified and validated in expensive irradiation campaigns. Entities from several nations went through this process for decades and accumulated large, but partial, datasets of irradiated/non-irradiated material properties. Although required for global scientific advancements and design approaches, collecting and combining these datasets is time-consuming or even impossible due to intellectual property constraints.

Initiated by an international working group, the “Material Database for Fusion” (MatDB4Fusion) aims to collect all kinds of material properties data which are relevant for fusion device design, supported by many public and private entities from different countries. While it should primarily host data for open access to benefit the scientific community the most, it may also serve as a secure area for restricted data under intellectual property rights to allow for secure scientific exchange and/or analysis. Neutrality and internationality are considered as key-aspects, embodied by a steering committee and the cooperation with OECD-NEA as database host. MatDB4Fusion shall act as a single reference for all fusion device design approaches, assess data gaps and reduce duplication of effort by improving focus of experimental campaigns, accelerate creation of codes and standards and provide sufficient data for machine learning solutions. The talk will focus on its current development state and ways to support our efforts.

P2-138

A comprehensive ab initio database of radiation-induced point defect for metals across the Periodic Table

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Many structural components in a fusion device are made of lightly alloyed body centred cubic (bcc) and face centred cubic (fcc) metals, with hexagonal close packed (hcp) metals also under consideration [1]. In such components, vacancies and self-interstitial atoms are the fundamental defects produced by high-energy neutron irradiation. The spatially fluctuating deformations that the accumulation of such defects produce is determined entirely by the defects' relaxation volume tensors [2], providing input to larger length scale models predicting deformations and stress as a function of dose [3].

We generalise the analysis of defects in bcc [4] and fcc metals [5] and compile a comprehensive ab-initio database of point defects in all the close-packed metals, including the 16 hcp metals. The database now spans all the technologically significant metals in the Periodic Table, save the lanthanides and actinides. We find that the total relaxation volumes of vacancies and any interstitial configuration are -35% and 146% of the metal's atomic volume to a good approximation. Our work provides a foundation for predicting radiation-induced microscopic and macroscopic deformations in metals in fusion and other irradiation environments.

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Tentative

P2-139

Correlation of material properties and monitoring of dissolved impurities via electrochemical methods: detecting hydrogen in liquid lithium loops of IFMIF-DONES

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The chemical reactivity of lithium and its potential flammability are highly critical aspects for a safe handling in any application of this alkaline metal. It is the chemical element with the lowest EC standard potential, therefore the strongest of reducing agents and very reactive with water, oxygen and nitrogen. Furthermore, impurity loaded lithium melts strengthen corrosion rates by factors of >> 10. These high rates are very important for structural materials, which count otherwise to be stable when affected by pure liquid lithium.

IFMIF-DONES, a DEMO-oriented neutron source, currently under construction near Granada/Spain, is using liquid lithium in high quantitative amounts, as 14m3 (i.e. 7 tons) under critical conditions (temperature, impurities); particular attention is therefore dedicated to the lithium impurity management to avoid any system malfunction. This is addressed by setting up an Impurity Control System (ICS), which supervises and controls the quantities of the gaseous impurities and includes suitable removal techniques, under which hydrogen is the most critical issue. Especially for hydrogen detection possibility of monitoring by the direct correlation between material properties and physical monitoring output via electrochemical sensor (EC) systems exists.

The contribution will report about the current status, further developments and progresses of the EC sensor and its foreseen applications under Li-test loops as LITEC, under the view of operational parameters (temperatures), system parameters (adjustment of sensors within the current ENS program) and the electrochemical control of different impurity measures, as also detecting isotope effects.

Invitation

P2-140

Research on Low-Cost Preparation and Radiation Resistance of Textured MAX Phases

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MnAXn+1 phases exhibit theoretically superior radiation resistance and corrosion resistance owing to their unique layered structures, showing promising potential for applications in nuclear fusion reactors. However, two critical challenges remain: (1) The cost-effective preparation of MAX phases as structural materials has been extensively studied but requires further optimization. (2) Within the operational temperature range of 300?500°C, MAX phases exhibit a significant degradation in radiation resistance accompanied by surface cracking. To address these issues, we achieved low-cost preparation of textured MAX phase structural materials using pre-alloyed TiAl powders and enhanced their crack resistance by tailoring the irradiation surface orientation. Through combined experimental and theoretical investigations, this work elucidates the microscopic mechanisms underlying the low-cost synthesis of MAX phases via pre-alloyed powders and clarifies the radiation resistance mechanisms of textured MAX phases. Furthermore, our findings reveal that the optimized texture orientation not only improves radiation resistance but also demonstrates exceptional oxidation resistance, offering new insights for designing high-performance MAX phases in extreme environments.

Tentative

P2-141

Enhanced resistance to radiation by periodic pattern of dislocation networks at a W/Fe interface

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A tungsten-iron interface with strong resistance to radiation damage has been suggested through simulation study. Calculations of interfacial cohesion energies through molecular dynamics method indicate that semi-coherent W(100)/Fe(100) and W(110)/Fe(110) interfaces have stronger thermal stability than the incoherent W(100)/Fe(110) and W(110)/Fe(100) interfaces. The rapid decrease of the number of defects and W-Fe atomic mixing at the interface after cascades have also been found. The enhanced defect recombination and recovery is facilitated by the formation of a regular square and an irregular hexagonal dislocation network at these two semi-coherent interfaces respectively. Furthermore, it is observed for the first time that the strong resistance to shear deformation of the W(100)/Fe(100) interface before and after cascades is due to a wave-like atomic displacement induced by the square pattern, which changes the activation model of slip systems from the interface. In addition, the effect of radiation defects on the magnetic storage property of the W(100)/Fe(100) system is investigated by ab initio method. Formation of small defects and W-Fe atomic mixing decreases the magnetic storage property, as implied by the magnetic anisotropy energy (MAE) calculations. All these results indicate that even though a W(100)/Fe(100) interface exhibits good stability and mechanical property against radiation damage, a radiation protection material is needed to conserve its magnetic property for its application in magnetic memory devices in nuclear facilities and space.

Tentative

P2-142

Unraveling the Atomic Mechanism of the Crystalline Phase-Dependent Structural Features and Special Spectral Design of α-, β-, and ε-Ga_2O_3

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Understanding the atomic mechanisms governing phase-dependent structural transformations and corresponding spectral characteristics is crucial for tailoring the semiconductor material properties for specific applications. Phase-dependent thermodynamic responses in α-, β-, and ε-Ga_2O_3 revealed distinct lattice energy dissipation behaviors, where critical melting energies (0.22 eV atom-1 ~ 0.34 eV atom-1) were correlated with specific electron-phonon (e-ph) coupling coefficient, driving preferential metastable phase interconversion via domain fragmentation between α/ε-Ga_2O_3, while β-Ga_2O_3 undergoes localized β → κ/δ transitions mediated by defect clustering. Specifically, the competitive effect between antiphase boundaries (APBs) and twin domain boundaries (TDBs) in α/ε-Ga_2O_3, driven by domain-boundary-mediated energy deposition for defect accumulation, were correlated with the phase-configurational dominance and non-recoverable structural properties. The spatially segregated β → δ (surface Ga-aggregated microdomains) and β → κ (latent track-confined) phase transformations in monoclinic β-Ga_2O_3 arise from defect configuration-dependent stress gradients, where surface-nucleated δ-phase appears to stabilize transient states while internal-nucleated κ-phase exhibits recoverable lattice distortions through reversible cation reordering. Responding to intense electronic excitation, the β-Ga_2O_3 forms nanohillocks attributed to GaO_6 octahedral framework (high atomic density/strong Ga-O covalency), while α/ε-Ga_2O_3 develop crater-like nanopores driven by structural looseness (tetrahedral Ga sites with reduced bonding energy), revealing phase-dependent surface dynamics governed by atomic packing density and bonding anisotropy. Under the influence of radiative and non-radiative recombination regulated by dominant defects, the suppressed visible photoluminescence in α-/β-Ga_2O_3 contrasts with enhanced radiative efficiency via significant bandgap narrowing (ΔE: 0.30 eV) in ε-Ga_2O_3, enabling tunable visible emission through phase-dependent defect-carrier interplay and facilitating the application of metastable-phase engineering.

Invitation

P2-143

Advances in rapid radiation damage-resistant alloy selection

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In this project, we present a new, rapid workflow to discover and down-select materials resistant to radiation damage via in situ irradiation coupled with transient grating spectroscopy (TGS) measurement of thermophysical properties during irradiation.

Traditional methods for identifying suitable materials for plasma-facing components require labor-intensive testing pre- and post-irradiation for a limited set of compositions. In contrast, our approach enables the simultaneous evaluation of property evolution through short-time self-ion radiation exposures across multiple compositions, delivering fast and comprehensive insights about radiation damage resistance as a function of elemental composition. We can obtain this information from short irradiations with negligible displacements per atom (DPA). Specifically, we seek the initial response to radiation damage right out of the gate, as a diminished change in thermophysical properties upon initial application of ion irradiation correlates strongly to sustained damage resistance. To arrive at the result, we have used the prompt drop of the surface acoustic wave frequency when the ion beam turns on as a proxy for the equilibrium vacancy concentration, and then correlated this value with the number of voids visualized on transmission electron microscopy images from irradiated samples up to 20 DPA.

This work has significant implications for accelerating materials development for fusion applications, in this case, RF antennas. By identifying an optimal composition within the CuCrTa system, we introduced a strong candidate material that is Nb-free for less long-lived radioactive waste, and is well-suited for RF antennas, contributing to the broader advancement of fusion energy.

Tentative

P2-144

Computationally accelerated design of multi-principle-element alloys for plasma facing components

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Plasma-facing components (PFCs), such as divertors and first walls, are among the most challenging components to develop in a fusion energy device. They must withstand extreme heat, neutron damage, and surface degradation caused by plasma-material interactions. Typically, PFCs operate at temperatures exceeding 600°C and require key properties such as high mechanical strength, a high melting point, excellent thermal conductivity, low erosion rates, irradiation resistance, and minimal tritium retention. Multi-principal-element alloys (MPEAs) offer a vast chemical space for optimizing material properties, including low activation, high ductility, high melting points, and resistance to secondary phase precipitation. In this work, we focus on the elements W, Ti, Cr, and V, as they meet the low activation criteria, with tungsten (W) already being a leading candidate for PFCs. With a 1% atomic composition increment, we can generate 175,171 unique binary, ternary, and quaternary alloys using these four elements. Experimentally exploring such a vast chemical space is impractical. To address this challenge, we utilize a combination of electronic and structural descriptors - calculated via first-principles methods - to accurately predict the intrinsic ductility of MPEAs. Additionally, we employ machine learning models trained on computational thermodynamics data to predict the operational temperature limits of MPEAs under fusion-relevant conditions. Accurate machine learning models have been successfully developed. We will discuss the development and application of these models to efficiently screen the large chemical space and identify promising MPEAs for potential use as PFCs in fusion environments.

Tentative

P2-146

Radiation Shielding Effectiveness of Aluminum Alloy 6061 with CoCrFeNi thin film for P-type Si(100) against 230 MeV Proton

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The operational environment of low Earth orbit (LEO) satellite is filled with high-energy cosmic radiation. More than 90% radiation around LEO are high energy proton which will pose significant radiation damage to satellite components. Therefore, continuous breakthroughs in radiation-resistant and/or radiation shielding technologies are essential to extend the reliability and lifetime of LEO satellite. Several recent studies have reported the attractive radiation resistance properties of high-entropy alloys (HEAs). By utilizing vacuum coating technology to fabricate HEA thin films, the stringent weight constraints of satellite components can be made more flexible. In this study, the magnetron sputtering system has been used to coat a quaternary high-entropy alloy CoCrFeNi thin films on aluminum alloy 6061 as radiation shielding material against 230 MeV proton for a P-type Si(100) dummy sample. Post irradiation examination about the electrical resistivity of the dummy sample has been applied to evaluate the radiation shielding effectiveness against high energy proton. Comparing to case of the naked aluminum alloy, aluminum alloy with the HEA coating bring benefits on the mitigation of proton irradiation to the dummy sample.

Tentative

P2-147

Raytracing Monte Carlo model of electronic thermal conductivity in single crystal W and W-Re alloys for Fusion Environments

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Tungsten (W) displays a high thermal conductivity, which, among other properties, makes it a good candidate for plasma-facing applications. However, the thermal conductivity can degrade severely under fusion operation conditions leading to transmutation, recrystallization, and defect accumulation. Thus, it is important to have methods that can assess the thermal conductivity drop during microstructural evolution in fusion operation. In this work, a Monte Carlo-based ray tracing model has been developed for predicting the electronic thermal conductivity in single crystal W and W-Re alloys. By treating electrons as rays, which stochastically collide with other electrons, phonons and impurities, their overall movement can be tracked and a net heat flux can be measured, from which the thermal conductivity can be inferred from Fourier's Law. In this manner, this important property can be calculated with relatively high accuracy, eliminating the need for costly experiments, and enabling 'in-service' dynamic changes to the conductivity during fusion reactor operation.

Tentative

P2-148

Modeling Non-Equilibrium Ion Dynamics in Tungsten Alloys: A Machine Learning Approach to Radiation Damage

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The durability of plasma-facing materials under extreme radiation is a primary obstacle for viable fusion reactors. Modeling the cumulative effects of radiation damage is highly complex due to the ultrafast dynamics following energetic collisions with the lattice. Consequently, our scientific understanding of degradation in plasma-facing materials is insufficient, presenting a major barrier to the advancement of commercial fusion energy production. This work aims to address these challenges by employing advanced computational techniques to model radiation damage mechanisms in tungsten alloys.

We present a new class of highly accurate machine learned interatomic potentials (ML-IAP) used in Molecular Dynamics (MD) that directly capture the electronic structure changes from ground state to warm-dense matter conditions. ML-IAP are enabled by a diverse training set of first-principles density functional theory (DFT) and time-dependent DFT (TD-DFT) defect calculations. Additionally, these multiscale linkages between electronic and atomic simulations are trained on the altered electronic state during the thermal spike of high energy neutron collisions. This multiscale approach excels in capturing ultrafast dynamics in radiation degradation thanks to its ability to operate at extremely short timescales, retain DFT-level accuracy, and model non-equilibrium conditions effectively. This work exemplifies progress in utilizing machine learning for materials development, with potential applications extending beyond fusion energy to predict material behavior under various extreme conditions.

Invitation

P2-149

Ion energy dependence on irradiation-induced defect stabilization effect in tungsten

Tomi Vuoriheimo[1], Fredric Granberg[1], Kenichiro Mizohata[1], Kalle Heinola[2]

[1]University of Helsinki, [2]International Atomic Energy Agency

Irradiation-induced damage to plasma-facing materials can significantly influence their performance in future fusion devices. Tungsten, the primary candidate for plasma-facing components in fusion reactors, has been selected for ITER under the updated baseline. Consequently, understanding the mechanisms of defect formation is critical for predicting material behavior under the extreme conditions of fusion environments.

Hydrogen isotopes in tungsten lattice have been observed to increase defect concentrations during irradiation, a phenomenon known as the defect stabilization effect. Experimental and computational studies conducted in recent years suggest two primary mechanisms: (1) hydrogen isotopes occupying vacancies, thereby preventing Frenkel-pair annihilation, and (2) hydrogen isotopes hindering the migration of interstitial tungsten atoms, leading to an increase in the vacancy concentration. Despite these advancements, certain aspects of the defect stabilization effect remain poorly understood, such as why do low energy impacts seem to cause more irradiation-induced damage within the material.

My research investigates the role of deuterium ion energy in the defect stabilization effect within deuterium-decorated tungsten. Deuterium ions were implanted into deuterium-decorated tungsten samples across a range of energies, from 1 keV to 100 keV, and the resulting depth profiles were analyzed to compare the increase in defect concentrations between high and low energy impacts. Computational simulations performed by molecular dynamics are combined with the experimental data to provide atomistic insights into the mechanisms of the defect stabilization effect. This integrated approach aims to deepen our understanding of this phenomenon and its implications for tungsten performance in fusion reactors.

Invitation

P2-150 "Presenting Author: Blanca Biel"

Beyond DFT: Development of a Fe-C fusion ready potential within the NEP approach

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Tentative

In recent years, there has been a flourishing in the development of new techniques to study materials. One of these techniques is the use of machine learning interatomic potentials (MLIAPs), instead of traditional interatomic potentials, to perform molecular dynamics simulations (MD). The current classical potentials lack the accuracy of quantum approaches such as Density Functional Theory (DFT) but MLIAPs combines the precision of the quantum approaches with the speed and efficiency of classical potentials.

In the field of fusion, MD is useful to obtain key parameters for other methods such as Montecarlo simulations, as well as characterizing relative low energy neutron induced defect cascades with the primary knock-on atom approach. The use of accurate potentials is thus key to correctly describe the multiscale simulations needed in the characterization of irradiated materials. These MLIAPs will be useful to compare with experimental data obtained in facilities such as IFMIF-DONES.

In this work we developed and tested a potential for Fe-C systems based on the Neuroevolutional Potential (NEP) [1] method due to its high accuracy and efficiency. We employed a database of 30000 Fe-C structures for the fitting of the potential and added a ZBL potential [3] region to correctly characterize the short interaction in cascades. With this MLIAP we study common defective structures found on irradiated steels and compare the results with already existing classical potentials.

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P2-151

Sputtering of High Entropy Alloys

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Due to an increasing population and energy consumption, it is vital to develop big scale energy generation technologies considering climatic constraints. Fusion energy is a good opportunity to generate low CO2 emission energy at a big scale. The harsh environment of materials inside a fusion reactor makes it difficult to find an optimum material capable of withstanding high temperature and irradiation damage. That is why we need to study potential candidates. We studied how tungsten based refractory high entropy alloys behave under Ar irradiation using molecular dynamic simulations. We focused mainly on sputtering yield of equiatomic alloys in the sub-keV regime and compared to pure tungsten. We additionally studied how the surface layers behave during irradiation, to determine if there is any composition change and preferential sputtering happening. The sputtering yield of the high entropy alloys tends to be higher than tungsten, due to lower mass atoms that sputter more than higher mass atoms. We found that there is a W enrichment of the alloys on the surface up to tens of nanometers. This study enabled us to understand how novel and complex materials behave under irradiation and potentially find a more suitable candidate than pure tungsten for plasma facing materials.

Tentative

P2-152

Large-Area Measurement of Ion Irradiation Induced Voids Without TEM

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Ion irradiation studies can be used to screen structural materials for resistance to fusion-like radiation damage. However, ion irradiation studies often involve a large parameter space, and transmission electron microscopy (TEM) post-irradiation examination requires expensive and difficult sample preparation. Additionally, small analysis volumes in TEM can miss micron-scale spatial heterogeneity in damage microstructures.

Backscatter electron (BSE) microscopy has been used to survey void swelling in neutron-irradiated samples. In this study, we show BSE microscopy can be used to study void swelling in an ion irradiated sample. A nickel sample, self-ion irradiated to 50 displacements per atom (peak damage), is used. Sample preparation is described, and the contrast of voids as a function of void depth and size, and electron accelerating voltage are explored. Void swelling statistics compiled via BSE images are compared to the results of 3D reconstructed FIB serial-sectioning images to assess the accuracy of BSE imaging.

Backscatter electron imaging enables large-area examination of ion irradiation induced void swelling microstructure, and can be used in the alloy design feedback loop as a fast method to obtain reliable swelling statistics for voids larger than 10 nm. To the extent that void swelling is a concern in fusion reactors, SEM alleviates analysis volume issues with TEM, and enables accurate analysis of void swelling across spatial heterogeneity. In the future, other SEM techniques could be used in tandem with BSE imaging to further characterize radiation effects with SEM.

Tentative

P2-153

An MD-Driven Bayesian Model for Void Evolution in Irradiated and Gas-Implanted Tungsten

Anirvinya Samanyu Tirumala[1], Daniel Mason[2], Felix Hofmann[3], Max Boleininger[2]

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An accurate simulation of plasma-facing components under fusion irradiation conditions presently remains out of reach of conventional atomistic simulations due to inherent limitations of the method. While coarse-grained models such as rate theory offer predictions of vacancy and He & H-isotope concentrations over a wide range of irradiation conditions [1], these do not account for the effect of the evolving microstructure on the primary damage generation process itself. Experiments involving sequential and simultaneous light-gas implantation [2], as well as experiments involving extreme levels of He implantation [3], demonstrate that the implantation of light gases leads to highly non-linear changes in the microstructural evolution under irradiation even at low temperatures.

Here, we present a generative probabilistic model for the microstructural evolution of tungsten subjected to simultaneous hydrogen and helium implantation and irradiation at athermal conditions. The model is rigorously calibrated against molecular dynamics (MD) simulations using a Bayesian inference framework. Our model predicts incremental changes in void content as a function of damage energy, pre-existing vacancy concentration, and gas concentrations. The model allows for systematic extrapolation towards the high dose overlap cascade regime, enabling rapid exploration of long-term microstructural evolution under irradiation and implantation conditions where thermal vacancy diffusion is suppressed.

[1] Watanabe et al. (2023) Fusion Engingeering and Design 194: 113899

[2] Markelj et al. (2020) Nuclear Fusion 60: 106029

[3] Balooch et al. (2022) JNM 559: 153436

Tentative

P2-154

Revealing the temperature-dependent hardening mechanisms of non-uniformly distributed defects for ion-irradiated metals

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[1]Peking University

Within contemporary nuclear reactor infrastructures, the deployment of metal-based alloys as structural components is noteworthy. Despite the extensive insights provided by the literatures on irradiation-induced hardening of structural materials, the temperature effect is not adequately included in the post-irradiation mechanical measurements, leading to the results that insufficiently reflect the material's performance under operational conditions. Here, an integrated approach including experimental examinations and theoretical modeling was employed to examine the temperature-dependent hardening of two prominent structural materials, namely 9Cr F/M steel and Zr-1.0Sn-1.0Nb-0.1Fe. Experimental procedures involved the characterization of grain morphology and size distribution by EBSD. Irradiations were conducted utilizing Au-ions at elevated temperatures, with TEM employed for observing irradiation-induced defects. Moreover, mechanical properties were assessed through high-temperature nano-indentation tests, revealing the concurrent occurrence of irradiation hardening and high-temperature softening in these materials. For theoretical modeling, a mechanistic model was developed to address the depth-dependent hardness. This model innovatively characterized the hardening mechanisms resulting from non-uniformly distributed defects in the irradiated layer while simultaneously incorporating the temperature effect. Calibration of the model demonstrated its capability to elucidate the hardness-depth relationships at different irradiation doses and testing temperatures. Further theoretical analyses indicated distinct evolution laws in irradiation hardening behavior with varying depths, while the increment in temperature accelerated the expansion of the plastic zone, resulting in a moderate indentation size effect. With increasing indentation depth, the hardening contributions of geometrically necessary dislocations and irradiation-induced defects gradually diminished. At substantial indentation depths, the dominant hardening mechanism was then controlled by statistically stored dislocations.

Invitation

P2-155

Model for the mechanical response of a tensile loaded heavily irradiated material

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The mechanical response of a heavily irradiated material under tensile loading is one of the key inputs to the qualification of candidate structural materials for fusion reactors. Given that the evolution of irradiation-induced defects governs changes in mechanical properties, it is essential to relate the microstructure formed under various irradiation and loading conditions to the resulting mechanical response. Existing models often contain a large number of adjustable parameters and oversimplify microstructural environments, limiting their range of validity and predictive power.

This study aims to develop a parameter-free model for predicting the stress-strain response of a heavily irradiated material under tensile loading conditions. The model combines an atomistic approach to the treatment of deformation of a heavily irradiated microstructure with dislocation analysis. The high dose configurations generated by atomistic simulations used in the analysis have been validated against experimental studies. We focus on the behaviour of irradiation-induced dislocation loops, which reduce the system's total strain energy in response to loading by tilting their habit planes or by changing their dislocation nature, resulting in stress relaxation. We discuss the mechanisms of the dislocation loop behaviour and derive a surrogate model for the stress?strain relationship of the damaged microstructure under tensile loading, providing input for finite element simulations of tensile test experiments.

Template

P2-156

Effect of oxide particles on swelling behaviour in Cu and Cu alloys under in-situ electron irradiations.

Qianfei FENG[1], Wentuo HAN[2], Farong WAN[2], Hiroshi OKA[3], Shigehito ISOBE[3], Naoyuki HASHIMOTO[3], Zimo GAO[4], Diancheng GENG[4], Ryuta KASADA[4]

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Copper (Cu) alloys, as promising candidate materials for divertors in International Thermonuclear Experimental Reactor (ITER), are typically required with sufficient high-temperature mechanical properties, excellent thermal conductivity as well as the capability of resisting high-flux neutron irradiation. One of the most challenging issues is radiation-induced swelling. Numerous methodologies, for instance, interface engineering, grain boundary engineering, etc., have been developed to introduce artificial defects to mitigate swelling. Although certain progresses have been made in the past few decades, principles behind which crystal defects contribute to resistance of radiation swelling remains unclear. This study focuses on the interaction of oxide particles with radiation-induced defects based on Cu/Cu alloys via high voltage electron microscope (HVEM). Alloying powders for pure Cu, oxide-dispersion strengthened (ODS) Cu and zirconium (Zr) additioning ODS-Cu samples were mechanically alloyed via a planetary ball mill. The prepared powders were prepared and sintered respectively for pure Cu, ODS-Cu and ODS-Cu(Zr) using spark plasma sintering (SPS) at 880°C. Specimens for electron irradiation were fabricated using focused ion beam. In-situ electron irradiation and imaging were carried out by the HVEM (JEM-ARM1300) at Hokkaido University.Massive void swelling was determined in all three specimens with damage dose reaching 3.13 dpa at 573K. However, after similar doses of electron irradiation, the volumetric swelling, void sizes and distributions in three individual samples differ due to the introduction of ODS particles and the addition of Zr. Largest swelling (0.63%) with uneven distribution of void was observed in ODS-Cu while Zr-additioning ODS-Cu exhibited the lowest swelling of 0.06%.

Invitation

P2-157

Effect of low-temperature irradiation on hardness and microstructural evolution in Fe-Cr binary alloys

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Recently, we investigated dose dependence of Vickers hardness and phase separation behavior of Fe-xCr binary model alloys (x = 0, 9, 15, 30, 45, 50, 70, 85, 90, 100 in wt.%) neutron-irradiated at around 100°C using LIBERTY in BR2, which can be pulled up during reactor operation (to be submitted). As a result, we reported that Fe-xCr alloys (x = 30, 45, 50 in wt.%) showed an overage-like irradiation hardening behavior while Fe-xCr alloys (x = 0, 9, 15, 70, 85, 90, 100 in wt.%) showed monotonous irradiation hardening with increasing displacement damage. The 3D atom probe method also revealed that the Cr concentration was the most modulated at the peak hardening, indicating that the overage-like irradiation hardening was caused by the re-distribution of modulated Cr after the peak hardening. However, the mechanism of the overage-like irradiation hardening behavior is still unclear. Therefore, this study aims to investigate phase stability and more detailed dose dependence of irradiation hardening under 100 °C irradiation in Fe-Cr alloys. We conducted 2 MeV proton irradiation to Fe-Cr binary model alloys at around 100 °C using the Dynamitron accelerator at FNL in Tohoku University, investigating hardness by nanoindentation tests. The hardening behavior of proton-irradiated coupons is compared with that of neutron-irradiated coupons, and the displacement rate dependence on the hardening will be discussed at the conference.

Tentative

P2-158

Object kinetic Monte Carlo simulations on the difference between fission neutron and heavy ion irradiation induced void evolution in Fe-Cr alloys

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Experimental results show significant difference between neutrons and ions irradiation of alloys, while the underlying reasons remain unclear. Herein, we performed OKMC simulations on void evolution in Fe-Cr alloys under neutron and ion irradiations, focusing on the effects of dose rate, temperature and irradiation particle type. Binary Collision Approximation and Molecular Dynamics are applied to obtain the cascade morphology of ion irradiation to study the effect of spatial correlation of cascades along the ion track, which is considered as a significant difference between the neutron and ion irradiations. Systematic OKMC simulations were performed at a wide range of dose rate from 10^{-7} to 10^{-3} dpa/s and temperature from 300 to 500°C. Simulation results show that both a higher dose rate and a lower temperature can lead to a higher density and a smaller average size of voids. High dose rate greatly promotes the interaction frequency between small defects and inhibit the absorption of vacancies by one vacancy cluster, thus enhancing the nucleation of vacancy clusters. High temperature enhances the migration of small defects and the absorption of vacancies by vacancy clusters, and thus enlarge the vacancy clusters. The impact of irradiation particle types that has influence on the PKA spectrum and cascade morphology is less important to void evolution compared with dose rate and irradiation temperature. This work provides fundamental insights into the difference in void evolution between fission neutron and heavy ion irradiations that are important to the application of ion irradiations in study irradiation effects of nuclear materials.

Invitation

P2-159

Towards a standardised methodology of radiation damage defect distributions for microstructure evolution models

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We have developed a model that integrates molecular dynamics (MD) and binary collision approximation (BCA) results to reconstruct the irradiation damage caused by ions of any energy in materials, being designed as a standardized tool for introducing defects into microstructure evolution models, such as object kinetic Monte Carlo or cluster dynamics. Our approach combines primary knock-on atom (PKA) energy and position data from SRIM with the CascadesDB database of MD cascade debris. For energies below the database energy limit, the fer-arc-dpa model [1] is employed to estimate Frenkel pair production; and for energies above it, SRIM is run iteratively. The code also incorporates methods for defect and cluster identification, facilitating detailed analysis of the MD database and the reconstructed damage. We demonstrate its application by comparing defect production to displacement per atoms models and analysing cluster distributions for self-ion irradiation at 1 and 5 MeV in Fe and W. While focused on ion irradiation, the framework is adaptable to neutron irradiation, provided PKA data is available; and it can be extended to other MD databases or BCA codes.

[1] Q. Yang and P. Olsson, Full energy range primary radiation damage model, Phys. Rev. Mater. 5 (2021) 073602

Tentative

P2-160

Irradiation Defect Evolution in FeCr alloys and Tungsten for Fusion Application: The Role of Temperature and Dose Gradients

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Tentative

Understanding irradiation-induced defect microstructures is crucial for developing radiation-resistant materials for future fusion reactors. This study characterizes defect evolution in Fe_3Cr and Fe_5Cr alloys implanted with Fe-ions under a temperature gradient and pure tungsten implanted with He-ions under both dose and temperature gradients. The first part of the talk will introduce a novel experimental setup designed for ion implantation under controlled gradients of temperature and dose. The second part will demonstrate how a combination of complementary experimental techniques provides a comprehensive understanding of irradiation effects. This approach will be illustrated using FeCr alloys as a prototype system. Synchrotron Laue diffraction captures total defect-induced lattice swelling, including defects undetectable by TEM, with a measurement sensitivity of approximately 10^-4. Results indicate that lower chromium content leads to increased swelling, particularly below 300°C. Swelling initially decreases with temperature up to 300°C, aligning with un-implanted samples, but beyond this temperature, swelling increases, making chromium content differences less significant. Nanoindentation and atomic force microscopy reveal irradiation-induced changes in deformation behavior. The Fe_3Cr sample exhibits greater hardness than Fe_5Cr, with 300°C marking a transition point: below this, hardness decreases with temperature, while above it, hardness increases. However, strain localization, a common irradiation-induced feature, remains largely unaffected by chromium content and temperature.

Meanwhile, tungsten samples were irradiated at temperatures ranging from 135 to 485°C, with an average dose of 6×10^15 ions/cm^2, and a minimum and maximum dose of 1.807×10^15 ions/cm^2 and 9.493×10^15 ions/cm^2, respectively. Necessary characterization is ongoing to assess the achieved dose gradient.

P2-161

Lattice Strain and Damage Accumulation in Self-Ion Irradiated Tungsten: Insights from PAS and XRD Analysis

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Tentative

Thermonuclear fusion offers a promising solution for reducing greenhouse gas emissions. Tungsten has been selected for ITER's divertor and is a key candidate for the first walls of the Demonstration power plant (DEMO). These components must withstand extreme conditions, including high heat flux and intense irradiation, which degrade material properties. Accurate predictions of these effects are critical for optimizing the design of reactor components.

A major challenge is establishing a link between microstructural evolution under irradiation and macroscopic property changes, which requires quantitative multi-scale analysis. To address this question, we performed 20 MeV self-ion irradiations on single-crystalline (SC) and polycrystalline (PC) samples at room temperature (RT), accumulating damage from 0.01 to 1 dpa. Positron Annihilation Spectroscopy (PAS) was used to quantify vacancy concentration and size, showing increased damage accumulation. This growth saturated between 0.1 and 1 dpa. At 500°C and 0.05 dpa, the vacancy distribution in SC and PC differed from that observed at RT. When damage reached 0.4 dpa, the vacancy distribution stabilized.

The evolution of deformation was analyzed using X-ray diffraction. We have developed an approach that accurately quantifies strain levels, even for PC samples with large disoriented grains. Both types of sample irradiated at RT exhibited similar increasing strain up to 0.1 dpa, followed by a decrease, becoming negative above 1 dpa, suggesting lattice contraction. Notably, peak strain coincided with the onset of vacancy cluster, suggesting a link between strain evolution and microstructural changes under irradiation. The mechanisms behind this correlation will be discussed at the conference.

P2-162

High dose irradiation of tungsten with realistic dose rate and atomistic resolution

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Tungsten is one of the plasma-facing materials in proposed nuclear fusion reactors, due to its high melting point and high thermal conductivity. Irradiation of tungsten has been studied for decades, both computationally and experimentally. From the computational perspective, Molecular dynamics simulations can precisely give us the atomic-scale view, which is widely used to study the mechanisms behind defect production and evolution during prolonged irradiation.

Recent studies explored the viability of using cascade simulations [1] and integrating an accelerated method [2] to achieve high doses. Furthermore, energetic cascades can significantly alter defect morphology, highlighting the need to account for overlap effects-an aspect often neglected in many large-scale simulation approaches. To address these challenges, our methodology combines molecular dynamics simulations with larger-scale models, specifically OKMC simulations. This hybrid approach aims to preserve atomic-level accuracy while achieving experimental dose rates. By comparing our results with conventional high-dose molecular dynamics simulations and experimental studies, we gain a more precise understanding of high-dose irradiation at realistic dose rates. Our developed workflow might solve the problems with existing workflows, where atomistic reactions affect the outcome at timescales far beyond the reach of the atomistic model itself.

[1] F. Granberg, J. Byggmastar, K. Nordlund, Journal of Nuclear Materials 556 (2021) 153158

[2] F. Granberg, D. R. Mason, J. Byggmastar. Computational Materials Science 217 (2023) 111902.

Tentative

P2-163

The influence of grain boundary migration and grain size on the irradiation damage behavior of tungsten

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Tentative

Tungsten (W), a critical plasma-facing material for fusion reactors, suffers irradiation-induced degradation under high-energy particle bombardment. This study investigates the irradiation resistance of nanocrystalline (NC-W), ultrafine-grained (UFG-W), and coarse-grained tungsten (CG-W) under fusion-relevant conditions (W^{4+} ion irradiation at 750°C, doses: 0.05-2 dpa). Microstructural and mechanical responses were analyzed via X-ray diffraction (XRD), transmission electron microscopy (TEM), and nanoindentation to clarify the role of grain boundaries (GBs) in defect mitigation.

NC-W, characterized by high GB density, exhibited superior irradiation resistance. At low doses (0.05-0.2 dpa), NC-W showed suppressed void formation and lower dislocation loop densities than UFG-W and CG-W, as GBs effectively trapped irradiation-induced defects. At 2 dpa, GB migration in NC-W enhanced defect annihilation and self-healing, while CG-W accumulated significant defects due to limited GB availability. XRD analysis confirmed minimal lattice distortion in NC-W post-irradiation, indicating structural stability. TEM observations revealed that GB migration in NC-W facilitated defect recombination and microstructure recovery. Nanoindentation results demonstrated that irradiation hardening in NC-W was governed by a balance between defect accumulation and irradiation-assisted grain growth.

This work highlights the importance of GB density and dynamics in irradiation resistance. The synergy between high GB density and active GB migration in NC-W significantly enhances defect tolerance, providing a strategic approach to designing durable plasma-facing materials. These findings advance the understanding of defect-GB interactions in irradiated nanostructured materials and offer practical guidance for optimizing tungsten-based alloys in fusion environments.

P2-164

Modeling Radiolysis Effects in FLiNaK and FLiBe Molten Fluoride Salt

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Molten fluoride salts, such as FLiNaK and FLiBe, are used as fuel and/or coolants and breeding blankets in nuclear fission and fusion reactors. In this study, we show that radiation can impact the chemical properties of these salts, by breaking complex fluorine structures. Measuring fluoroacidity, the amount of free fluoride ions available in the salt, reveals these impacts. Previous experiments have shown that fluoroacidity affects the corrosion rate of metals in contact with these salts. However, the mechanistic details of this effect are not fully understood. Here we use molecular dynamics to emulate radiolysis in FLiNaK and FLiBe. We analyzed radial distribution functions to measure the change in fluoroacidity due to irradiation. The results for both FLiNaK and FLiBe should indicate a larger increase in fluoroacidity (based on the pF scale) for FLiBe, compared to FLiNaK. We anticipate that this analysis will serve as a foundational assessment to understand the effect of radiolysis in both FLiNaK and FLiBe corrosion.

Tentative

P2-165

Data driven service life prediction of nuclear silicon carbide composite materials

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Continuous SiC fiber-reinforced SiC based composite materials (SiCf/SiC) have good application prospects in the liquid cladding of fusion reactors due to their excellent neutron economy, high-temperature mechanical properties, chemical inertness, and radiation resistance. China's CFETR has also listed SiCf/SiC composite materials as candidate structural materials. Aiming at the factors of irradiation field, temperature field, and stress field in the working environment, a cross scale multi physics field prediction model for the force heat radiation coupling of nuclear SiCf/SiC composite materials was constructed through deep learning combined with finite element method. The main focus was on studying the influence of fiber weaving method, porosity, and different interfacial layers on the mechanical and thermal properties of nuclear SiCf/SiC composite materials. Predicting the performance degradation and damage evolution process of SiCf/SiC composite materials for nuclear use in service environments, as well as their influencing factors, mainly including stress-strain, creep and fatigue fracture properties, and temperature field changes, and their relationship with service environments. Studying the failure modes of the matrix, fibers, and interfacial layer in the composite material process can provide support for optimizing the design and preparation of SiCf/SiC composite materials for nuclear use and accurately predicting their service life in service environments.

Tentative

P2-167

Microstructural Evolution of Irradiation-Induced Defects in Tungsten: cluster dynamics study

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The hostile conditions in a fusion tokamak reactor pose a significant challenge to the development and design of in-vessel components. Neutron irradiation leads to degradation in the mechanical and thermo-physical properties of materials due to the formation of irradiation-induced microstructural defects, such as dislocation loops, and vacancy clusters and due to generation of transmutation products. This study adopts an integrated computational-experimental approach to investigate the microstructural evolution of dislocation loops and voids in tungsten (W), a key material for fusion reactor applications. W is widely recognized for its properties, making it an best candidate for armour application. However, its susceptibility to irradiation-induced defects under high-temperature conditions requires in-depth analysis. By employing cluster dynamics (CD) model, simulations are performed on irradiated W specimens subjected to recent experimental testing [1]. This approach bridges experimental observations and computational predictions, enabling a comprehensive understanding of defect behavior.In this investigation, the displacement per atom (dpa) rate, derived from irradiation experiments, is integrated into the CD model to simulate the defect evolution in W. Using the dislocation elastic theory method, the formation energies of two prominent dislocation loop types, 1/2<111> and <100>, are calculated across a range of temperatures (600°C-1200°C). These calculations provide critical insights into the stability and relative population of these loop types as a function of temperature. The CD simulations predict the temperature-dependent evolution of dislocation loops and voids up to a dose of 0.2 dpa, offering a detailed perspective on the interplay between defect dynamics and material response under fusion-relevant conditions.

Tentative

P2-168

Evaluating neutron activation to determine the value of RAFM steels in fusion, thermal, and fast reactors to accelerate collaborative development

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Invitation

Reduced-Activation Ferritic Martensitic (RAFM) steels are developed for structural components in fusion reactors. The main purpose of RAFM steels is to reduce the generation of long-living radioisotopes under neutron irradiation compared to conventional ferritic-martensitic steels, which are already used in fission reactors. The development of RAFM steels is spearheaded by the fusion community, but these alloys also offer benefits for fission reactors. The further development of RAFM steels would surely be accelerated if the fusion and fission reactor communities worked cooperatively on alloy development, properties measurement, irradiation testing, and manufacturing research. Neutron irradiations require special attention to ensure that the neutron energy spectra are appropriate for the data objectives.

We will discuss possible collaborative neutron irradiation experiments needed to mature RAFM steels for fusion and fission reactors. We present the results of activation evaluation for EUROFER97 (three different batches) and industrial scale Castable Nanostructured Alloys (CNAs). High-fidelity Monte Carlo neutronic models are used to predict relevant neutron energy spectra for the different reactor types operating in different flux spectra to capture energy-dependent effects of neutron interactions. The case studies include a tokamak-based fusion reactor, a liquid metal-cooled fast reactor, and an epithermal/thermal range heat-pipe cooled microreactor. The inclusion of activation specimens tailored for the neutron energies of interest is planned to complement the calculations. Accurate measurement of the neutron spectra provides inputs to the neutronics calculations to predict the radioactive waste category and can evaluate the possibility of recycling irradiated RAFM materials.

P2-169

Microscope and Ion Accelerators for Materials Investigations (MIAMI-2): Status Update

Jonathan A. Hinks[1], Graeme Greaves[1]

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The Microscope and Ion Accelerators for Materials Investigations (MIAMI-2) facility combines a transmission electron microscope (TEM) and dual ion beamlines to explore the dynamic evolution of microstructures under in-situ irradiation. The system is currently housed in a purpose-built two-storey laboratory at the University of Huddersfield. However, it is scheduled to move to new accommodation elsewhere in the UK starting in 2025 due to the relocation of the Electron Microscopy and Materials Analysis (EMMA) research group which operates MIAMI-2 for internal projects and external users through the UK National Ion Beam Centre (UKNIBC).

An overview of the facility including its capabilities, examples of experimental results and access routes for users will be presented. Alongside this, an update on progress on construction of the new building will be given as well as the timeline for the forthcoming move. Advantage is being taken of this transitional period to implement new features and upgrades to the ion accelerator and dosimetry systems which will also be covered.

MIAMI-2 is one of around ten TEMs with in-situ ion irradiation across the world which form a network of unique facilities. Each of these offers different possibilities for examining the response of materials at the nanoscale under a range of extreme conditions relevant to nuclear fusion energy systems including combinations of elevated temperatures, inducement of atomic displacements and light-gas accumulation. A mini-review of these instruments will also be presented covering not only the research possibilities but also the various approaches to interfacing ion beams with a TEM column.

Invitation

P2-170

Thermodynamic and Kinetic Properties of Defects in Concentrated Alloys

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Changes in materials microstructure under fusion environments originate from the evolution of irradiation-created defects. Hence, understanding the thermodynamic and kinetic properties of defects is paramount to assess the suitability of materials under irradiation. In this work, we present a general Langmuir-like model for defect concentration in an arbitrarily complex solid solution and apply this model to generate expressions for concentrations of vacancies and small interstitial atoms. We then calculate the vacancy concentration as a function of temperature in two model alloys, the equiatomic CoNiCrFeMn and FeAl, showing there is no clear correlation between vacancy thermodynamics and chemical ordering in the CoNiCrFeMn alloy but clear systematic patterns for FeAl. This work provides future avenues to the prediction of thermal interstitials and vacancies in concentrated alloys, which is necessary for models of non-equilibrium behavior of solid solutions. Furthermore, concentrated alloys generate rough energy landscapes for defects to migrate. Unfortunately, an analytical expression for the diffusivity of a particle in a rough potential for dimensions higher than one has been elusive to the community since Zwanzig's seminal work in 1988 [1]. In this work, we develop an analytical model for the diffusivity in a noisy solid solution in an arbitrary number of dimensions. Our results agree with Zwanzig in 1D and kinetic Monte Carlo results, and show that diffusivity is always sluggish provided that the average of the distribution does not change, with the limit for infinite dimensions equal to the diffusivity in a delta function migration barrier distribution with the same average.

Tentative

P2-171

Fe^{+} irradiation effects on the microstructure and magnetic properties of Fe-10at%Cr films

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Fe-Cr binary alloys serve as a model system for ferritic-martensitic steels, which are candidates for blanket materials in nuclear fusion reactors. The calculation of the electromagnetic forces acting on the blanket requires precise values of their magnetic properties. Therefore, it is essential to understand how high-dose irradiation affects magnetism. Two research groups from Japan and Greece jointly investigated the microstructure and magnetic properties of Fe+ irradiated Fe-Cr films.

Using the sputtering method, Fe-10at%Cr films with a thickness of 70 nm were fabricated on a silicon oxide substrate with a 5 nm Cr cap layer. Fe+ ions were irradiated at an acceleration voltage of 490 keV at both room temperature and 300°C. The damage level was set to 5 dpa. The structure was examined using XRD, EBSD and STEM-EDS. The magnetization curve was investigated using VSM.

Microstructural observations of the cross-section showed that the film consists of an aggregate of nanocrystals of the order of several tens of nanometers. After irradiation at 300°C, the coercivity decreased, while the saturation magnetization increased by 3%. In the material irradiated at 300°C, Cr segregated at the grain boundaries, whereas its concentration within the grains decreased. This redistribution of chromium is considered one of the factors contributing to the increase in saturation magnetization.

Tentative

P2-172

Thermal and irradiation creep: A thermodynamic theory of coupling between point-defect diffusion and dislocation plasticity

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We develop a unified, thermodynamically consistent model for irradiation creep, that combines dislocation glide and the stress-induced preferential absorption (SIPA) of point defects at dislocations. Central to the model is the premise that the dynamics of point defects such as interstitials and vacancies is controlled by temperature, while the dynamics of extended defects, among which dislocations are of prime relevance, is controlled by an effective temperature that pertains to the configurational degrees of freedom which evolve on a much slower time scale than the kinetic-vibrational degrees of freedom and, as such, fall out of equilibrium with the former. Results for thermal and irradiation creep in copper and aluminum suggest that conventional SIPA mechanisms are inadequate to explain the pronounced dependence of the irradiation creep rates on stress and temperature, necessitating nontrivial corrections to the SIPA dislocation climb rates. These results provide important insights into the stress and temperature dependencies of dislocation climb, and illuminate the modeling of irradiation creep in ferritic-martensitic steels for reactor applications.

Tentative

P2-173

Hardening and microstructure evolution of FeCrAl ODS steel under high-energy heavy ion irradiation

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Four kinds of FeCrAl oxide dispersion strengthened (ODS) steels (15Cr-4Al-0.1Ti, 16Cr-4Al-0.1Ti, a Zr added 15Cr-4Al-0.1Ti-0.6Zr and an Al-free 16Cr-0.1Ti steel) and one conventional ferritic/martensitic steel (HT9) were irradiated at 623 K with high energy 56Fe ions provided by a cyclotron to generate a quasi uniform atomic displacement damage plateau within a depth range of about 25 μm. The Vickers micro-hardness test at room temperature showed that different ODS steels exhibited different hardening phenomena after irradiation, in which 15Cr-4Al-0.1Ti-0.6Zr and 16Cr-0.1Ti steel have significantly lower irradiation hardening than the other two ODS steels, while the non ODS steel HT9 have the highest irradiation hardening. The TEM observation revealed that the average size and number density of defect clusters induced by irradiation in ODS steel are smaller than those in HT9 under the same conditions. The smaller grain size, higher the number density of oxide dispersions with smaller average diameters contribute to the resistance to irradiation hardening by providing sinks and/or traps for irradiation-induced point defects, which effectively suppresses the nucleation and growth of dislocation loops. A power law correlation was established between the sink strength of oxide dispersoids and the irradiation hardening. The improved irradiation resistance is primarily attributed to the increased sink strength. Assuming that interstitial dislocation loops are the main reason for irradiation hardening, it was found that the relationship between irradiation hardening given by TEM defect data combined with dispersion barrier hardening model and sink strength is consistent with experimental results.

Tentative

P2-174

In-situ irradiation effects and Ti(CON) stability in baseline and precipitation-strengthened V-4Cr-4Ti alloys

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Invitation

V-4%Cr-4%Ti (V44) alloy system is the leading structural material candidate for Fusion liquid lithium breeder blanket designs envisaged for operations in the range of ~400-700°C. In these materials, there is a poor understanding of the mechanisms controlling low temperature hardening/embrittlement (LTHE), which impose the lower temperature limit. Historically, LTHE in vanadium alloys is attributed to dislocation loops and radiation-induced precipitation of Ti-rich phases [1]. However, a detailed understanding of their synergistic interactions and formation mechanisms is not well established. To address this gap, we performed in-situ ion irradiations at JANNuS-Orsay MOSAIC facility from 400 to 700°C and 0.5 to >10 dpa to quantify the radiation induced microstructure evolution in both baseline V44 and a novel precipitation-strengthened vanadium alloy variant showing superior high-temperature properties.

Here, we will reveal the dose-temperature dependent dislocation loop microstructure development; the relative populations of a<100> and a<111> loops, and radiation induced segregation of alloying elements on the loops. The results will also show whether titanium oxycarbonitride (Ti-CON) nanoprecipitates formed during alloy processing are stable against susceptibility to ballistic dissolution. The combined results will be used to guide alloy design for future tuning of irradiated properties. Research jointly sponsored by Tokamak Energy, University of Birmingham and 2023 UKAEA's Fusion Skills Voucher programme.

[1] T.Muroga, Vanadium for Nuclear systems, Comprehensive Nuclear Materials, 2012, p391-406

P3-001

Helium transmutation damage studies on EUROFER97 using ion implantation

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The longevity of fusion power plant components remains a key challenge for the realization of sustainable fusion energy. Emphasis is on the breeding blanket (BB), particularly the First Wall (FW), which is exposed to an intense 14.1 MeV neutron flux. While atomic displacement damage is mitigated beyond 350-400°C, helium transmutation reactions are likely the primary limiting factor for FW durability. However, in the absence of a dedicated fusion neutron source, the critical concentration of helium for the baseline structural material EUROFER97 remains underdetermined.

This work explores High Temperature Helium Embrittlement (HTHE) in EUROFER97 through helium ion implantations of 0.8-1.2 MeV that were conducted at room temperature (RT) and 550°C. Helium concentrations of 200 (end of lifetime of a starter blanket) up to 1200 appm - corresponding to FW damage of 20-120 dpa - were achieved to examine transmutation damage effects on future BB designs.

Focused Ion Beam (FIB) and Transmission Electron Microscopy (TEM) were employed for helium bubble formation and evolution in-depth analysis on samples with different helium concentrations and implantation temperatures. Electron Energy Loss Spectroscopy (EELS) confirmed the post-implantation formation of helium bubbles.

Ongoing work explores the impact of helium implantation of 0.4-1.8 MeV to quantify the isolated effect of helium on the toughness of EUROFER97. Hot Isostatic Pressing (HIP) is utilized to bond two implanted surfaces to fabricate macroscopic Charpy impact and tensile specimens for subsequent mechanical characterization. These generated results contribute to the validation of EUROFER97 and to explore the limits of fusion reactor components.

Tentative

P3-002

Advancing the understanding of radiation damage in reduced activation CCAs by atom probe tomography

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Plasma-facing materials are subjected to extreme conditions, including high temperatures, high magnetic loads, and neutron fluxes with energy levels far exceeding those in conventional fission reactors. These challenges call for the development of advanced structural materials. Compositionally complex alloys (CCAs) are promising candidates due to their high-temperature strength, structural stability, and resistance to wear, corrosion, and oxidation. Another critical factor in material selection is the absence of long-lived radionuclides formed through transmutation, allowing components to be classified as low-level waste upon disposal.

This study examines the radiation response of a body-centred cubic CCA made exclusively from reduced activation elements, following heavy ion implantation at 500°C up to 15 dpa, conducted at the Ion Beam Centre HZDR in Germany. In addition to assessing the homogenised condition, the study explores the potential formation of a second phase during aging to act as recombination sites and mitigate radiation damage. SEM, X-ray diffraction, and atom probe tomography (APT) confirmed the presence of the Laves phase, while nanoindentation measurements indicated improved irradiation damage resistance after aging. APT provided insights into damage evolution, accumulation in different phases, and segregation behaviour variations based on radiation-induced defect types. Understanding defect formation and the role of second phases in delaying damage accumulation is crucial for the potential use of CCAs in nuclear applications.

Invitation

P3-003

High temperature and fatigue behavior of RAFM microstructures optimized for DEMO Water Cooled Breeding Blanket

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The contribution of ENEA together with Rina-CSM to the Eurofusion programme “WPMAT" deals with the development of innovative RAFM steels suitable to be adopted as structural materials for the blanket of the first DEMO reactor. While the main focus, so far, has been the improvement of toughness and the prevention of the low temperature irradiation embrittlement, the residual mechanical properties at high temperatures are also a subject of interest. Indeed, recent CFD calculations showing that several spots of the Breeding Blanket undergo markedly higher temperatures (550°C) compared to the main operating one (300°C).

In recent years a particular TMT has been developed to induce recrystallization of the laths substructure and obtain an equiaxed nano-grain structure. This so-called “recrystallization” treatment has proven successful in refining the equiaxed grain size by an order of magnitude thus implying benefits on the out of pile toughness and expected neutron irradiation embrittlement. However maintaining this nano-grain microstructure stable at 550°C is still a challenge. According to aging tests the hardness values remain stable at temperatures below the final tempering one, however tensile and creep tests show a marked collapse of the properties when the threshold of 400°C is exceeded. Fatigue behavior also deserves attention, since the effectiveness of this microstructure seems to be different in the case of Low Cycle Fatigue with respect to High Cycle Fatigue. In this paper, a thorough mechanical and microstructural characterization is reported in order to describe the evolution with temperature of this particular nano-grained microstructure

Tentative

P3-004

Strategies to lower the DBTT of RAFM steels acting on chemical composition and thermomechanical treatments

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ENEA and its industrial partner RinaConsulting-CSM contribute to the Eurofusion programme in the development of innovative RAFM steels able to withstand the critical low temperature (280-300°C in the “water-cooling” option) foreseen for the blanket of the first DEMO reactor. Therefore the target is the development of tougher alloys, suitable to prevent the low temperature irradiation embrittlement. The DBTT shift under irradiation is affected by Helium production in addition to dpa damage.

Concerning alloy design, an overall indication deriving from the last neutron irradiation experiment is to increase Vanadium and Nitrogen at the expense of Carbon and Manganese. Another choice concerns the reduction in Tantalum content similar to other alloys (EM10) proving limited irradiation embrittlement. The ingot (alloy 3460) has been cast according to the specifications concerning RAFM steels for fusion environment.

Alternative thermo-mechanical treatments have then been applied on this RAFM steel. On the one hand the choices on the chemical composition result in a quite low DBTT after a standard double normalization and tempering treatment; -139°C on KLST specimens. Tempering temperature results from a trade-off in order to maintain sufficient Yield Strength; 435MPa at 300°C. On the other hand the decrease in grain size (less than one micron), achieved by means of recrystallization treatment, leads to a transition curve that doesn’t show a sharp drop in the absorbed energy, but instead a more gradual transition from USE to LSE, further decreasing the DBTT. The tensile properties in the recrystallized condition result markedly higher at 300°C than after conventional treatment.

Invited

P3-005

Investigation of the Nano-Mechanical Plastic Deformation Behavior of Heavy-Ion Irradiated RAFM Steel

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The Nano-indentation deformation behavior of the heavy ion irradiated RAFMs is investigated in the present work. Specimens of the CLF-1 steel were irradiated with 123.4 MeV 20Ne ion at both -50 °C and 400 °C, respectively, to achieve a mean displacement damage level of 0.15 dpa. A quasi-uniform distribution of the displacement damage was produced in the specimens by utilizing an energy degrader at the irradiation chamber. The results of nano-indentation experiments indicate that during the loading stage, the irradiated materials exhibit a lower strain rate, but during the holding stage after reaching the maximum load, the irradiated materials show a higher strain rate than the un-irradiated material. Transmission electron microscopy observation showed that the low-temperature irradiation produced a high density of irradiation-induced dislocation loops, which impeded the glide of dislocation lines during the loading stage. However, as the load increases gradually to the holding stage, the dislocation lines can overcome the pinning by the dislocation loops. The strain that did not occur due to the pinned effect of the dislocation loops in the loading stage was released during the holding stage, resulting in a higher strain rate in the irradiated specimen during the holding stage. Based on the Peierls mechanism, a quantitative relationship between the number density of the irradiation-induced dislocation loops and the strain rate during the holding stage was established. The calculated results agree well with the experimentally obtained strain rate during the holding stage.

Tentative

P3-006

Physical and mechanical properties of EUROFER97 after 20 dpa neutron irradiation

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Reduced-activation ferritic/martensitic (RAFM) steels are the benchmark structural materials for in-vessel components of fusion reactors, such as the first wall and blanket structures; however, little is known about the irradiated thermophysical performance of these 8-9%Cr steels, while much more is known about the irradiated mechanical performance. A previous study reported that the steel varieties such as F82H and EUROFER97 generally perform similarly to Grade91 steel in terms of thermal properties. The temperature-dependent thermal conductivity of unirradiated RAFM steels appears to be higher than Grade91, particularly at temperatures below 400°C. The electrical resistivity of unirradiated RAFM steels increases with test temperature but decreases after neutron irradiation (~5 dpa), leading to an increased induction current in the blanket. Low temperature hardening and embrittlement (LTHE) initiates at very low dose and saturates at ~15-20 dpa, with most pronounced LTHE at <350°C. In RAFM steels, LTHE causes an increase in strength, a loss in ductility and a decrease in fracture toughness. This is a particular challenge for water cooled blanket DEMO designs. EUROFER97 were tested after neutron irradiation at ~20 dpa at various irradiation temperatures between 200-450°C. Thermal diffusivity, electrical resistivity, microhardness, and tensile data were obtained through baseline and post-irradiation testing. The neutron irradiation was conducted at the High Flux Isotope Reactor at Oak Ridge National Laboratory. The discussion in this presentation will cover the irradiation experiment and post-irradiation characterization methods, the effects of radiation on thermophysical and mechanical properties, the measured irradiation temperatures, and the correlations among these properties.

Tentative

P3-007

Evaluation and prediction of Creep-fatigue lifetime of F82H for Water-Cooled Ceramic Breeder Blanket Application

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A reduced activation ferritic martensitic (RAFM) steel known as F82H is the primary candidate for structural materials in the water-cooled ceramic breeder test blanket module (WCCB TBM) for ITER. Given that the WCCB TBM experiences cyclic loads from cooling water pressure, heat load, and electromagnetic forces within the temperature range where F82H is susceptible to creep, a quantitative investigation of the creep-fatigue (CF) lifetime for this material is essential. For a new material like F82H, developing a CF lifetime curve requires substantial time and extensive data. This study aims to evaluate and predict the CF lifetime of F82H by combining limited CF experimental results with a strain-range partitioning method. Additionally, a novel approach is proposed based on these findings and similar steel grades, such as 9Cr-1Mo-V, to predict the CF lifetime of F82H. The experimental work involves a round-bar CF specimen with a gauge section measuring φ5 mm x 12 mm, as specified in ASTM E2714-13. Tension-comparison tests are conducted at 550°C in air, featuring a stress ratio of R = -1 and a strain rate of 10^-1 %/s. The total strain ranges from 0.75% to 1%, with a holding duration of 180 seconds per cycle at the peak compressive stress. Using the experimental data and the strain-range partitioning method, constitutive relations for evaluating the CF lifetime of F82H are formulated. The predicted lifetime aligns with experimental results by a factor of two. The specifics of the CF lifetime evaluation and prediction for F82H will be presented in detail.

Invitation

P3-008

Investigation of Neutron Irradiation Effects on T91 Ferritic/Martensitic Steel for Fusion Reactor Applications

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Ferritic/martensitic steels, particularly T91, are prime candidates for structural components in fusion reactors due to their favorable mechanical properties and resistance to radiation-induced swelling. Understanding the effects of neutron irradiation on these materials is crucial for their application in future fusion energy systems. This study examines the tensile properties and microstructural evolution of T91 steel subjected to neutron irradiation in the Advanced Test Reactor (ATR) at doses ranging from 0.01 to 10 displacements per atom (dpa) and temperatures of 300°C, 450°C, and 550°C. Room-temperature tensile tests were conducted to assess changes in mechanical behavior, complemented by nanoindentation to evaluate hardness variations. Advanced microscopy techniques, including scanning electron microscopy (SEM), and transmission electron microscopy (TEM), were employed to characterize irradiation-induced defects such as Cr-rich α' phase precipitates, G-phase formation, and dislocation loop development. Additionally, Fe-9Cr, Fe-12Cr, Fe-9Cr-0.1C and Fe-12Cr-0.2C model alloys, analogous in composition to T91 and HT9 respectively, were investigated to isolate the roles of chromium and carbon in irradiation responses. The findings provide insights into the degradation mechanisms of T91 under neutron irradiation, informing the development and qualification of structural materials for DEMO and beyond.

Tentative

P3-009

Optimization of Normalizing and Tempering Conditions for Microstructure and Mechanical Properties of ARAA

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Advanced Reduced-Activation Alloy (ARAA) has been developed as a Korean RAFM steel for fusion reactors. To validate its fabrication process and establish a material property database, an 18-ton scale production of ARAA is underway. This study examines the influence of normalizing and tempering conditions on the microstructure and mechanical properties of ARAA. Specimens from forged and rolled plates underwent heat treatments, followed by microstructure analysis and mechanical testing, including tensile tests at room and 550°C, Charpy impact tests at -20°C, and hardness measurements. Normalizing was conducted at 900°C, 1000°C, 1100°C, and 1200°C for 40 minutes, followed by air cooling. Both 900°C and 1000°C produced an ASTM grain size number of 9; however, the grain size coarsened sharply at 1100°C and 1200°C, yielding numbers of 4.5 and 3, respectively. Moreover, the aligned ferrite formed at 900°C may adversely affect mechanical properties, leading to the selection of 1000°C as optimal. Tempering at 750°C for 70 minutes, followed by air cooling, yielded a hardness of 200?220 Hv, consistent with a tempered martensitic microstructure. Additional tempering tests at 1, 2, and 4 hours revealed gradual reduction in yield and tensile strengths, confirming 70 minutes as optimal. Based on these findings, the optimal heat treatment for ARAA is normalizing at 1000°C for 40 minutes with air cooling, followed by tempering at 750°C for 70 minutes with air cooling. These conditions are applied to the production of ARAA plates of various thicknesses, with further evaluations planned for microstructure analysis, mechanical property evaluation, and ultrasonic testing.

Tentative

P3-010

Post Irradiation Examination for Chinese Fusion Reactor Structural Steels Irradiated in STIP-VIII

Lei Peng[1], Guoping Yang[2], Yong Dai[3], Jingyi Shi[1], Hongbin Liao[2], Long Zhang[2], Peng Bi[3], Roger Brun[3], Yiyin Shan[4], Yanfen Li[4], Jiarong Zhang[4]

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In China, RAFM and ODS steels were developed as the primary candidate structural materials for fusion reactor design. To test the radiation resistance of these materials, we have joined the international cooperative irradiation program, i.e., SINQ target irradiation program (STIP). After 4 irradiation programs (STIP V-VIII), we have obtained many irradiated samples of domestic structural materials. After carrying out a series of post irradiation examination (PIE), we have accumulated some irradiation properties data of the structural steels, including mechanical properties data tested by tensile and small punch test. This report introduces recent PIE results mainly derived from the CLF-1 steel and 9Cr/14Cr-ODS steel irradiated in STIP VIII. The tensile test results show that CLF-1 has similar mechanical behaviours after irradiation as other RAFM steels, mainly manifested as obvious increasing in yield strength and decreasing in elongation, especially direct necking without strain hardening. The 9Cr/14Cr ODS steel still retains a certain uniform elongation at the low irradiation dose, however, it exhibits brittle fracture without necking tested at room temperature at high irradiation dose. The SPT test results show that the DBTT shift of CLF-1 steel increases linearly with the irradiation dose. This is in marked contrast to the results of RAFM steels irradiated in STIP I and II, where a significant acceleration in the DBTT shift occurs after doses higher than 7 dpa. This may be attributed to the significantly lower He/dpa ratio in STIP VIII, which more closely resembles the irradiation environment of fusion reactors, compared to the earlier STIP.

Tentative

P3-011

Synergistic effects of displacement damage and helium on the deuterium permeation in the oxidized RAFM steel under dual-beam irradiation

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Developing barrier coatings with high resistance to hydrogen isotope permeation in RAFM steels, which serve as structural materials in fusion blankets, is a critical challenge for fusion reactors. We previously proposed a novel approach to address this challenge by optimizing the chemical composition of RAFM steel and developing an appropriate oxidation process to create an oxide layer with a permeation reduction factor (PRF) as high as 10000.

In this work, we aim to evaluate the stability of the PRF and the microstructure of the oxide layer under irradiation. The oxidized RAFM steel samples were irradiated by Fe ions and Helium ions at doses of 0.1 dpa/1.5 ppm, 1dpa/15 ppm and 10 dpa/ 150 ppm at two temperatures(450 °C and 550 °C) in the dual-beam ions irradiation facility at Los Alamos National Laboratory. Microstructural evolution, including the formation of helium bubbles and local amorphization, was characterized using transmission electron microscopy. Additionally, positron lifetime spectroscopy was employed to identify vacancy-type defects on the material's surface. Gas-driven permeation experiments were conducted at temperatures ranging from 350°C to 650°C, and the PRFs of the irradiated samples were compared to that of the original sample. To understand the underlying mechanism behind the evolution of the PRF, a first-principle study was conducted to investigate the influence of irradiation-induced point defects on the dissolution and diffusion properties of hydrogen in the oxide layer.

Tentative

P3-012

Effect of thermal aging on microstructure, mechanical properties and corrosion resistance of reduced activated ferritic-martensitic and oxide dispersion strengthened steel

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Reduced activation ferritic/martensitic(RAFM) and oxide dispersion strengthened (ODS) steels have been considered structural material in fusion reactor. The change of material properties reduced the service life is an innegligible issue at elevated temperature for long time. Thermal aging properties of RAFM and ODS steel were researched in this study. The studied steel received in the normalized and tempered (N & T) condition was thermally aged at 873 K for 0-8000 h. The changed of mechanical properties (hardness, tensile properties, small punch properties), microstructure and corrosion resistance properties were explored. Hardness and tensile strength decreased, while the ductility increased with increasing aging time. The variation of strength and the minimum value of total elongation with increasing test temperature could be attributed to dynamic strain aging (DSA). The prevalence of DSA activity was found more on aged material. Shift in ductile to brittle transition temperature (DBTT) and decrease in upper shelf energy (USE) of the studied steel were observed on aged material. The microscopic investigation of the studied steel revealed the coarsening of martensitic sub-boundaries, $M_{23}C_6$ carbides, and the precipitation of Laves phase during thermal aging. The oxide film thickness of studied material also increased with increasing aging time due to the fraction of CSLBs along PAGBs decreased and coarsening of Cr-rich $M_{23}C_6$.

Tentative

P3-014 "Presenting Author: Jason R Trelewicz"

Mechanical properties of castable and sintered nanostructured low-activation ferritic-martensitic steels

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In this work, we present recent microstructural and mechanical properties of two reduced activation ferritic-martensitic steels of identical starting compositions formed by different fabrication routes: castable nanostructured alloy (CNA) and a sintered nanostructured alloy (SNA). We combine synchrotron based x-ray diffraction and electron microscopy (STEM-EDS) to characterize the base microstructure and nanoprecipitate properties of both alloys. The tensile properties were determined from RT to 650, and the creep properties were determined for 100-300 MPa loads and at fixed temperatures of 600 and 650 °C. The fine dispersed MX precipitates were found to have positive effects on both the tensile and creep properties with gains in elongation, yield and creep compliance, compared to other RAFM variants. In addition to the traditional mechanical properties, nanoindentation experiments spanning temperature ranges of 25 °C to 650 °C, with a maximum load of 100 mN, were performed to investigate the creep properties. The nanoindentation properties were found to be remarkably similar. The creep stress exponent (n) for CNA and SNA were found to be in the range of 8-35 and the activation volume was ~14-42b^3, underscoring a dominance of dislocation-mediated mechanisms in both alloys. Notably, we observed a decline in the creep stress exponent with increasing temperature, attributable to the heightened influence of thermally activated dislocations. This phenomenon suggests a potential transition in the deformation mechanism towards a thermally activated dislocation climb process, significantly impacting the observed creep behavior.

Tentative

P3-015

Mechanical Property and Irradiation Hardening in Low-activation Dual-phase Fe-Mn-Cr-Al-V-C Alloy

Kazuyuki Furuya[1], Koichi Tsuchiya[2], Taku Ishida[3], Eiichi Wakai[4], Satoshi Emura[2]

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Microstructures, mechanical properties and irradiation hardening were studied of Fe-20Mn-15Cr-5.0Al-1.5V-1.0C (at%) developed for applications to advanced-nuclear reactors (including fusion nuclear reactors) and high-power accelerator-target systems. The alloy composition is free of highly radio-activation elements such as Co, Ni and Mo. High Cr concentration of 15 at % was adapted to improve corrosion resistance. The alloy sample was processed by vacuum induction melting and hot rolling. The final heat treatment was done at 1473 K for 1.0 h, followed by water-quench. The sample consisted of BCC phase (about 78 %) and FCC phase (about 20 %), and small amount of $M_{23}C_6$. The sample had good mechanical properties with tensile strength of 620 MPa and total elongation of 40 % at room temperature. The samples were irradiated at 573 K with 2.5 MeV Fe^{2+} ions up to about 6 dpa at a depth of 720 nm from the sample surface. Nano-indentation tests revealed that the sample had comparable or better irradiation hardening resistance (ΔHN: about 1.15 GPa) than a typical fusion structural material such as F82H steel and fission reactor pressure vessel steels such as A533B. These results suggest that further tuning of the phase stability of the alloys, combined with high-entropy effects, could lead to the development of alloys with improved irradiation resistance and mechanical properties. Therefore, materials developed by this new approach are expected to be useful for applications in components for fusion reactors, next-generation nuclear reactors, and high-power large accelerator facilities.

Invitation

P3-016

Screening and manipulation by segregation of dopants in grain boundary of Silicon carbide: First-principles calculations

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The effect of segregation behavior of dopants on the performance of grain boundary (GB) in SiC has been systematically investigated by first-principles calculations. The solution energies of dopant atoms greatly reduce in the GB region compared with those in the bulk, which makes the dopant atoms inside the grain tend to segregate and aggregate near the GB. The driving force of GB on dopant segregation generally decreases with the increase of distance from GB plane, and the preferential site of dopant is closely correlated with the atomic size of dopant. Next, the segregation of single dopant induced the changes in the strength and stability of GB have been explored. It is found that non-metallic dopants have the significant embrittlement effects on GB strength. However, the segregation of most metallic dopants could strengthen the GB and Mg atom has the most significant strengthening effect on the GB. The stability of GB can be greatly improved by segregation of Al and Y dopants. Besides, the aggregation of H atoms has the obvious embrittlement effect on the GB. Furthermore, the co-segregation behavior of different dopants has also been explored. Be and Mg dopants have the most significant inhibition effect on the segregation of detrimental impurities H/He/O due to the repulsive interaction between dopant atoms. The present results provide a new insight into the effect of dopant segregation on GB properties and are expected to be a useful guidance for screening the chemical composition and manipulating the performance of SiC-based ceramics.

Tentative

P3-017

Unveiling the Mechanism of Irradiation-Induced Flow in Mullite Ceramics Under Particle Irradiation

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Ceramic materials such as silicon carbide are considered nuclear applications. Since cracks initiate fractures in ceramics, controlling them is crucial for structural integrity. If cracks can be filled, the opportunity for stress concentration, which serves as the starting point of failure, is reduced. Recent studies indicate that oxide ceramics probably soften (or amorphization) only under irradiation. If such softening enables crack filling under irradiation, fracture strength might be improved. This study reports the softening and moving phenomena of mullite ($3Al_2O_3 \cdot 2SiO_2$) under irradiation, which we interpret as viscous-flow.High-purity mullite were used to examine the extent of crack-closure under irradiation. The crack-closure may indicate that viscous-flow has occurred under irradiation. Fine-short (FS) cracks and wide-long (WL) cracks were introduced on polished surfaces using micro-Vickers tester at two different loads. Then, proton irradiation on the surface was conducted at 2.0 MeV, reaching doses of ~0.2 dpa. It was confirmed that almost all FS cracks were closed after irradiation. In contrast, the width of WL cracks decreased by an average of approximately 40% after irradiation, but the reduction was limited to approximately 20% in regions near the indentation. The cracks near the indentation were initially wider, suggesting that the residual stress was relatively low. In conclusion, viscous-flow occurred under irradiation, which contributes to crack width reduction, is likely stress-dependent, as evidenced by the smaller degree of crack width reduction in regions with lower residual stress. A more detailed discussion on the relationship between stress and flow will be presented at the conference.

Tentative

P3-018

Silicon-carbon bond formation and deuterium retention on a silicon crystal surface in the DIII-D divertor

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We report a formation of silicon (Si) - carbon (C) bonds on a Si crystal surface exposed to plasma fluxes in the DIII-D divertor, showing evidence from thermal desorption spectroscopy (TDS) and X-ray photoelectron spectroscopy (XPS). In-situ real-time growth of silicon carbide (SiC) layers on plasma-facing components would be an attractive option for the sustainable operation of fusion reactors, as it has low-to-medium-Z, low hydrogenic diffusivity, high-temperature strength, and mechanical resilience to neutron damage. Hence, understanding the in-situ SiC formation process is important. We exposed a Si crystal sample to H-mode deuterium (D) plasmas in DIII-D via the Divertor Material Evaluation System (DiMES). DIII-D employs graphite walls, so 1-3 % of the ion flux to the divertor is typically C. A post-mortem analysis by XPS showed a formation of a Si layer containing C ~ 10% for ~5 nm depth on the Si crystal surface, likely by C ion implantation. XPS spectra of the C1s peak showed a 283 eV peak besides a 285 eV peak corresponding to a C-C bond. The 283 eV peak is well explained by the existence of a Si-C bond. Deuterium retention analysis by TDS showed a D desorption peak at 900 K, as previously reported for D retained in SiC. A D retention of 1.3×10^20 D/m^2 was observed from the 900 K peak, giving a retention ratio of Si:C:D ~ 10:2:5.

Work supported by US DOE under DE-AC02-09CH11466, DE-FC02-04ER54698, DE-SC0022528, and DESC0023378.

Tentative

P3-019

Double walled heat exchanger tube design and fatigue cracking for Water Cooled Lithium Lead breeder module

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Various blanket concepts are foreseen for the DEMOnstration Fusion Power Reactor (DEMO). Among them, the Water-Cooled Lithium Lead (WCLL) concept achieves power extraction by a 15 MPa and 350 °C water loop. In each breeder module, water is driven through 8mm circular tubes immersed in molten PbLi. Any leakage form this pressurized water tube in molten metals may have dramatic consequences on the module and reactor safety. Therefore, this component must achieve a strong reliability of the tightness function. In this study, potential leakages are assumed to be linked to cracks initiating on manufacturing defects and propagating under fatigue loading. In order to improve the overall component reliability the design is based on tightness redundancy. Two co-axial tubes, each one able to withstand the water pressure, are assembled together to form a Double Walled Tube (DWT) foreseen to be built of EUROFER97 grade.

This study focuses on DWTs mechanical characterizations. Both tubes constituting the DWT are here linked together by an interphase that must fulfill various goals like good thermal conductivity and moderate tritium permeability and is therefore designed and manufactured expecting a beneficial contribution to the DWT fatigue behavior. Present study investigates the ability of the interphase layer to blunt or deviate fatigue cracks. A mechanical characterization of representative samples is carried out by tracking the crack path under fatigue loading of such an assembly from initiation to failure thanks to Digital Image Correlation. This study leads to a first overview of the fatigue behavior, driving further DWT design.

Tentative

P3-020

A Microstructure-based Viscoplastic Model of First Wall & Blanket Fusion Materials

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Understanding the behavior of fusion reactor first-wall and blanket materials, which evolve under extreme conditions, requires models that account for microstructural changes. To address this, we have developed a computational multiscale, microstructure-based viscoplastic model for polycrystalline materials. The model incorporates thermally activated dislocation motion to capture rate-dependent deformation mechanisms and is formulated within a component-scale thermo-elastoplastic framework, enabling predictions of mechanical behavior under coupled thermomechanical loads. Moreover, the model is implemented on MOOSE framework, a multiphysics object-oriented simulation environment, to facilitate large-scale simulations and material response evaluations. Model validation was performed using experimental data from unirradiated tungsten and reduced-activation ferritic-martensitic (RAFM) steel under monotonic, cyclic, and creep deformation. The results demonstrate its ability to capture key deformation features, including strain-rate sensitivity and temperature-dependent flow stress, thus establishing confidence in its applicability for fusion reactor first-wall and blanket materials. In addition, the model provides a basis for predicting the thermomechanical response of these materials under fusion conditions, an essential feature for accurately representing the evolving microstructure in high-temperature and irradiation environments.

Tentative

P3-022

Investigation of neutron irradiated nuclear structural materials using Small Punch Test

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The operating environments of advanced fusion reactors pose significant challenges for structural materials due to extreme temperatures and irradiation field. To address these challenges, innovative structural materials such as High-Entropy Alloys (HEAs) and Oxide Dispersion Strengthened (ODS) steels have been developed to overcome these severe conditions. However, their response to neutron irradiation must be thoroughly evaluated beforehand. Given the limited availability of irradiated materials, Small Punch Test (SPT) has been recognized as a promising tool for the investigation of neutron irradiation induced hardening and embrittlement with its miniaturized dimensions and the ability to acquire robust tensile property results.

In this study, Small Punch Tests have been carried out on selected neutron irradiated nuclear structural materials, including baseline nuclear structural material (Eurofer97) and innovative structural materials (HEA and ODS steel). These materials were first exposed to neutron irradiation to 1 dpa in the Belgian Reactor 2 (BR2) nuclear reactor at SCK CEN (Belgian Nuclear Research Centre). The followed SPT experiments are conducted in accordance with the EN 10371:2020 standard. With the tensile properties deduced, i.e. yield stress and ultimate tensile strength, a comparison among the discussed materials in terms of their resistance to neutron irradiation is provided.

Tentative

P3-024

The effect of nanoparticle/matrix interface sink strength on the irradiation response of Cu-Y_2O_3 under He/D sequential irradiation

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Oxide dispersion-strengthened (ODS) copper exhibits exceptional electrical and thermal conductivity, making it a leading candidate material for divertor heat sinks in future fusion reactors. previous studies attribute the well radiation resistance of ODS Cu to two mechanisms: 1) direct defect trapping at nanoparticle/matrix interfaces and 2) indirect effects from grain refinement. However, the dominant mechanism governing radiation resistance of ODS Cu remains unclear. To isolate nanoparticle/matrix interface effects, we conducted a controlled study on Cu-Y_2O_3 alloy with comparable grain sizes (about 1 μm) but varying nanoparticle sizes (10-500 nm).

Cu-Y_2O_3 alloys with controlled nanoparticle distributions (10-500 nm) were synthesized via mechanical alloying (MA) and spark plasma sintering (SPS). Samples underwent sequential irradiation: first with 180 keV He^+ ions (1×10^{17} ions/cm^2) at RT, then with 110 keV D^+ ions (1×10^{17} ions/cm2) at 300°C, using the BNU-400 ion implanter at Beijing Normal University. Microstructural analysis revealed no significant correlation between nanoparticle size (10-500 nm) and either bubble/dislocation loop densities or size distributions. This suggests limited direct interface contributions to defect evolution when nanoparticles exceed 10 nm. The difference in the proportion of ∑3 grain boundary and small angle grain boundaries was observed with different nanoparticle sizes, which suggests that the indirect effects on the sink strength caused by changes in the nature of grain boundary cannot be ignored, especially for the large particles.

Tentative

P3-025

Low-cyclic fatigue behaviour of CuCrZr alloy for the DEMO divertor heat sink

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Copper-chromium-zirconium alloy (CuCrZr) is the main candidate material for the heat sink of the European DEMO fusion reactor. Characterization of its mechanical properties is of paramount importance and a prerequisite for reliable design of DEMO. To expand the knowledge of the mechanical behavior of CuCrCz, experimental low-cycle fatigue (LCF) tests were performed at different temperatures. In more detail, the test matrix for the characterization program includes fatigue tests at different strain amplitudes with a fixed strain rate. LCF tests were carried out at Laboratory of Nuclear Material Qualification of Brasimone Research Center (ENEA) in Italy. The strain-controlled uniaxial fatigue tests were performed in accordance with ASTM E606/E606M with a triangular waveform and constant strain rate using an axial strain gauge. The geometries of the specimens are conformed to ASTM E606/E606M. The experimental results were analyzed and discussed showing that cyclic loading of CuCrZr induces hardening at an early stage of the cycle. The results obtained will contribute to the creation of a database related to the EUROfusion characterization program of CuCrZr alloy for the DEMO divertor heat sink.

Tentative

P3-026

Multiscale Mechanistic Study of Creep Behaviour in Copper-Chromium-Zirconium Alloy Heat Sinks for Fusion Devices Using Advanced Experimental Methods

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With heat fluxes of 15 MW/m^2, 300 °C temperature and 200 MPa, heat sinks used in fusion devices are required to endure extreme conditions. While copper has high thermal conductivity, its lack of structural stability makes it unsuitable for fusion applications. Copper-Chromium-Zirconium alloys (CuCrZr) are expected to exhibit higher creep strength and thermal conductivity. The study investigated Grain Average Misorientation (GAM) and grain rotation distribution across 5 %, 10 %, and 20 % creep lives in a CuCrZr alloy tested at 285 MPa and 300 °C. A 50-hour test, representing 20 % creep life was conducted using an hourglass-shaped sample. Regions at 20 %, 10 %, and at 5 % creep life were analysed using Electron Backscatter Diffraction (EBSD) and Large Area Mapping (ATLAS-SEM) enabled by a 5 nm TiAg coating on one side and high-temperature paint on the other for tracking macrostructural strains. At 5 % creep life, minimal rotation and uniform strain distribution were observed, indicating early-stage deformation. By 10 %, strain localisation intensified, with intermediate rotation and increased misorientation, marking a transition phase. At 20 %, substantial rotation and slightly increased GAM values highlighted significant plastic deformation and heterogeneous strain distribution, indicative of advanced microstructural damage. These findings provide critical insights into strain localisation, inter and intragranular creep damage evolution, under prolonged stress. It also emphasizes the importance of GAM and rotation in characterising creep deformation, contributing to the design and evaluation of creep-resistant materials

Tentative

P3-027

Plasma facing components based on advanced high temperature heat pipes-concepts and design

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Tentative

For future fusion reactors, the integrity and efficiency of plasma-facing components (PFCs) are crucial, especially for the divertor target. One of the primary challenges lies in managing the asymmetric heat loads that are predominantly concentrated on the plasma-facing side. Conventional cooling solutions, such as the mono-block chain design with embedded cooling channels in the tungsten armor, are limited by their small heat transfer area. This leads to extremely high local heat flux densities, low heat exhaust efficiency, and increased material constraints. For future fusion power plants with even higher thermal management requirements, an alternative heat exhaust solution with high efficiency utilizing heat radiation would be necessary.

Heat pipes are heat transfer devices with excellent heat transfer rates (up to 104 times higher compared to solid conduction), utilizing the latent heat of the phase transition from the working fluid. Their application in PFCs enables rapid heat extraction and redistribution over a larger area, reducing local hot spots and decreasing the heat flux density to the coolant. This heat radiator concept could significantly improve the heat exhaust efficiency and potentially reduce the material challenge.

In this contribution, we introduce the design concept of heat pipe-based PFCs (HPPFCs). The proposed component consists of three key units: a plasma-facing armor, a liquid metal heat pipe acting as a heat radiator, and a secondary coolant. An optimized HPPFC would incorporate with also advanced tungsten-based composites or alloys as envelope materials. The feasibility, design considerations, and expected performance benefits of HPPFCs will be discussed.

P3-028

Three-point bending behavior of W-Ta alloys after BR-2 irradiation at 290°C

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Tungsten(W) is considered a promising candidate material for the divertor of a fusion reactor. On the other hand, W has a problem of brittleness at low temperatures, to overcome this problem W-alloys and composite materials are being researched. One factor that induces the low-temperature brittleness is grain coarsening by recrystallization in a high-temperature environment. To increase the recrystallization temperature and suppress the grain coarsening at high temperatures, W-Ta alloys has been reported that the recrystallization temperature rises to about 1600°C with an addition of about 5%Ta, but the properties after neutron irradiation have not yet been investigated. The purpose of this work is to investigate the mechanical properties after irradiation of the W-Ta alloys.

The specimens used in this work were pure W, W-1%Ta and 3%Ta alloys, which were prepared by powder metallurgical process and hot/worm rolling, and were stress-relieved. Three-point bending test specimens measuring 1x1x15mm were cut out with the long axis parallel to the rolling direction for mechanical property evaluation, and irradiated at 290°C and neutron dose of $1.23 \times 10^{23} n/m^2$ (En>0.1MeV) in the BR-2 reactor using LIVERTY irradiation device. After irradiation, the specimens were shipped to Japan and hardness tests were performed at room temperature and three-point bending tests were performed at temperatures from room temperature to 700°C. Test temperature dependence of the flexural strength and fracture mode will be presented.

Tentative

P3-029

Development of W_2C-reinforced W as high heat flux material

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Template

Tungsten is favoured for fusion power plant divertors due to its excellent thermo-physical properties, but its mechanical strength deteriorates at high temperatures due to recrystallization. This research aims to enhance tungsten's properties to withstand the extreme thermal loads during operation.

Reinforcing fusion-grade tungsten with W_2C particles at grain boundaries effectively removes oxygen without requiring high-temperature hydrogen treatment [1]. This enhancement improves densification, strengthens the microstructure, and increases flexural strength compared to pure tungsten. While higher concentrations of W_2C particles lead to refined grain sizes and higher hardness [2], low concentrations of W_2C particles dispersed in isotropic W matrix led to a DBTT below 400 °C. The prolonged thermal treatment at temperatures above 1250 °C confirmed that the presence of W_2C particles at W grain boundaries prevents the abnormal growth of W grains. The examination of D retention revealed that the composite with 4 wt% of W_2C, retained (78.7 ± 2.9) 10^19 D/m^2 and (148.4 ± 1.5) 10^19 D/m^2 of D for a D exposure temperature of 370 K and 523 K [3].

High thermal stability coupled with good mechanical properties, comparable or even better thermal shock behaviour than W, and reasonable thermal transport properties (thermal conductivity > 100 W/mK @ 1000°C) makes the W-W_2C composites a very interesting and competitive material for the DEMO divertor application.

[1]Šestan et al., J. Nucl. Mater., 524, 2019.

[2]Novak et al., Mater. Sci. Eng. A, 772, 2019.

[3]P. Jenuš et al, J. Nucl. Mater., 581, 2023.

P3-030

Development of reduced activation ODS steels by standard methods and additive manufacturing

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Thanks to their mechanical properties and their ability to trap homogenously helium from transmutation, Reduced Activation ODS steels offer many advantages for fusion reactors. The aim of this presentation is to show the latest results obtained on ferritic and martensitic ODS steels using the conventional powder metallurgical route, but also the progress made in obtaining ODS by additive manufacturing.

The production of ferritic or martensitic ODS steels using the conventional route makes it possible to produce that Reduced Activation ODS steels on a large scale. This is illustrated by the latest ODS production carried out within the EUROfusion framework.

At the same time, new additive manufacturing processes are being evaluated. Laser Additive Manufacturing techniques allow to obtain dense materials. however, nano-oxide precipitation mechanisms seem to limit the possibility of obtaining high reinforcement densities. In contrast, our latest work on the cold spray technique shows that it enables a high density of nano-oxides to be obtained in a ferritic Fe-14Cr ODS steel. After cold spray deposition of ODS steel powders and a cycle of Hot Isostatic Pressing, the tensile and impact properties are particularly interesting.

The main directions for ODS steel development are presented and discussed in the light of the latest results obtained.

Textwo

P3-032

Enhancing the performance of plasma-facing materials by laser powder-bed fusion

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Tungsten is the leading plasma facing material (PFM) candidate for fusion reactors due to its attractive thermo-physical properties, resistance to sputtering, and chemical compatibility with tritium. As a PFM, tungsten will need to be fabricated into functional components with intricate geometric features, but the limited ductility of tungsten at room temperature makes it challenging to forge and machine parts more complex than sheets and rods. Laser-based additive manufacturing (AM) techniques are an attractive to reliably fabricate complex geometries without expensive tooling. However, it is challenging to fabricate crack-free AM tungsten due to high cooling rates and its low fracture toughness. In this talk, we will present a successful CALPHAD-based alloy design strategy to fabricate crack-free AM tungsten materials. Starting with >104 alloy compositions, potential formulations were down-selected by balancing: hot-cracking susceptibility, solidus temperature, thermal conductivity, and coefficient of thermal expansion. These alloys were then fabricated and characterized for both their thermal properties and mechanical properties via compression testing and three-point bend tests. We will compare these results with pure AM W and discuss the relationship of these properties with microstructural features and processing-related defects. This work demonstrates a feasible strategy for fabricating tungsten-based materials in complex geometries without substantially sacrificing the excellent properties of pure tungsten.

Tentative

P3-033

Radiation-Induced Effects in Tungsten-Based High-Entropy Alloys for Plasma-Facing Components in Fusion Devices

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Tungsten (W) is proposed as the primary material for plasma-facing components in the future DEMO reactor and has also been approved for use in the first wall and divertor area of ITER. To improve the performance of plasma-facing components various advanced concepts of W materials are being developed. In particular, tungsten-based refractory high-entropy alloys (HEA) comprising the WTaVCrTi system are considered as promising candidates due to their excellent radiation resistance and strength at high temperatures. In this work, the microstructure and properties of radiation damaged samples of pure W and HEA alloys were compared. To simulate damage morphology in samples typical of neutron irradiation, 15 MeV W ion irradiation was conducted at the UK Dalton Cumbria Facility. The microstructure and nanomechanical properties of irradiated samples were characterised at Imperial. The phase segregation analysis was performed using XRD. Nanoindentation and SEM were used to investigate the irradiation hardening and defect structures in the materials.

Tentative

P3-034

Deuterium behavior in damaged tungsten-rhenium alloy

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As a first-wall material in fusion reactors, tungsten (W) undergoes transmutation reactions under neutron irradiation, producing rhenium (Re), tantalum and other transmutation elements. At the same time, high-energy neutron irradiation and high flux deuterium plasma exposure may cause surface blistering and property degradation. Understanding the combined effects of neutron and deuterium plasma irradiation on tungsten-based materials is crucial.In this study, heavy ion damage and deuterium plasma exposure experiments were carried out on W-Re alloy samples successively. Various techniques, including SEM, NRA, and TDS, were taken to analyze the damage defects and deuterium behaviors in W-Re alloy.Surface morphology analysis shows that heavy ion damage slightly increases the size of deuterium-induced blisters on the surface, but reduces their number and coverage. The deuterium retention in damaged sample is about 40% higher than that in undamaged sample. However, under the experimental conditions of this study, only a small part of the increased retention comes from the damaged layer (0-300 nm), and the majority comes from the areas beyond the blister layer (300 nm-1 μm). Heavy ion damage introduces numerous defects in the shallow surface layerl. In the early stage of plasma exposure, the deuterium filling of defects in the damaged layer inhibits the diffusion of deuterium toward the bulk and effectively delays the blistering. Once the blisters protrude, the cracks generated also impede the deuterium diffusion. As a result, the inhibition of surface blistering in the damaged sample allows more deuterium to diffuse into the bulk, resulting in higher deuterium retention.

Tentative

P3-035

Study of ion sputtering and degradation of Platinum- and Rhodium-coated first mirrors for optical diagnostics in fusion devices

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The development of robust and efficient metal first mirrors (FMs) is essential for optical and infrared plasma diagnostics in fusion devices, where extreme conditions demand optical materials capable of resisting degradation over a 20-year operational lifetime. This study focuses on the characterization of several FM prototypes featuring micrometer-thick platinum (Pt) and rhodium (Rh) coatings and includes a comparison with single-crystal (SC) molybdenum mirrors. These prototypes were tested in a high-frequency (40.68 MHz) discharge, which generated ion fluxes with energies ranging from 100 to 200 eV in hydrogen and helium at 5 Pa. Said coatings were designed to retain high reflectivity under ion doses of 10^{23}-10^{24} ions・m^{-2}. Using advanced ion sputtering techniques, the coatings were deposited onto stainless steel substrates to ensure uniformity and strong adhesion. Their performance was assessed through experiments simulating exposure to charge-exchange neutral beam fluxes and material redeposition effects. Optical reflectivity measurements revealed degradation in the Pt and Rh coatings compared to uncoated substrates, particularly in the visible and near-infrared spectral ranges. Changes in the coating morphology and redeposition were observed in the mirrors when examined with scanning electron microscopy and energy-dispersive X-ray spectroscopy (SEM-EDX). Mirror cleaning using radiofrequency (RF) plasma was also evaluated as a potential in-situ technique to remove contamination. Prolonged exposure uncovered degradation mechanisms affecting the coatings' long-term reflectivity. Sputtering rates in the range of tens of nanometers per hour were measured. The findings emphasize the critical role of the coating materials in extending the lifetime of FMs.

Tentative

P3-036

Ex-situ and in-situ laser cleaning of divertor tungsten surface for tokamak

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During experiments in a tokamak, plasma interaction with the tungsten (W) surface causes material to migrate from areas of erosion, particularly at the strike points, to areas of deposition within the scrapeoff layer. These deposits, which can accumulate up to tens of microns, are porous and trap light impurities, leading to plasma perturbations when released. Additionally, deposits are fragile and can flake off during operation, often resulting in plasma disruptions. A high fluence campaign at the WEST tokamak highlighted this issue, with 686 flaking events over 3 hours of plasma operation and 384 discharges. These events caused 133 disruptions and early interruptions. Notably, 82% of flaking events were triggered by prior disruptions, creating a feedback loop that exacerbates the problem. A project has been initiated to clean the divertor surface, as managing deposits is crucial for long pulse operations. The project uses a commercial 1mJ/1064nm pulsed fiber laser for cleaning. Initial studies determined laser parameters not damaging bulk tungsten, confirmed through optical and confocal microscopy. Subsequent ex-situ trials on components with deposits showed that the laser was able to remove all deposits, validated by methods like confocal microscopy, X-ray fluorescence, and metallography. The generated dust was pumped away simultaneously to prevent redeposition. During the 2024 summer shutdown, two 30° sectors of the divertor were cleaned, with a cleaning speed of about 2h/m^2, and will be monitored during the following campaign. Full divertor will be cleaned in-situ during the 2025 summer shutdown to improve plasma performance for long pulse operations.

Invited

P3-037

Low high-temperature hardness: A drawback of current ODS copper applied in monoblock-type components fabricated by HIP

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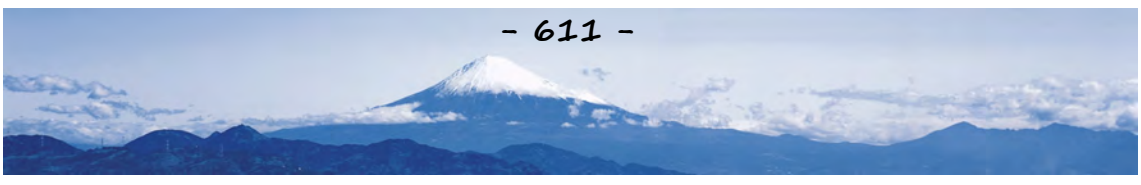
Copper-based alloys, including precipitation hardened copper (PH-Cu) and oxide dispersion strengthened copper (ODS-Cu), are considered highly promising heat sink materials for fusion divertor. Unlike PH-Cu, ODS-Cu exhibits outstanding thermal stability, retaining its mechanical properties even after annealing at temperatures up to 950°C and exhibiting reduced sensitivity to subsequent heat treatments. This stability enhances its suitability for high-temperature joining processes making ODS-Cu a highly promising material for heat sink applications in divertor.

In this work, components comprising five tungsten monoblocks with an ODS-Cu tube were fabricated using the hot isostatic pressing (HIP) technique. Circular indentations, attributed to the HIP process,were observed at the toroidal gap on the inner surface of the tube. The W/Cu/ODS-Cu mock-up exhibited indentations averaging 88.33μm in depth, signigicantly deeper than the 35.55μm indentations observed in a comparable W/Cu/CuCrZr mock-up. This disparity is primarily due to the lower high-hardness of ODS-Cu (14.90HV at 680°C) compared to CuCrZr (45.34HV at 680°C), which inceased susceptibility to deformation during HIP. High heat flux (HHF) thermal fatigue testing at 15MW/m^2 revealed unexpected premature leakage failure at the toroidal gap of the W/Cu/ODS-Cu mock-up. The indentations compromised the structural integrity of the ODS-Cu tube, reducing local fatigue strength and promoting crack initiation under cyclic HHF loading.

Preliminary results indicate that the low high-temperature hardness of current ODS-Cu presents a limitation for its application in monoblock-type components fabricated by HIP. This finding offers new perspective on the application and performance enhancement of ODS-Cu in the manufacturing of divertor components.

Invited



P3-038

Revealing deformation behaviour of W-Cu composites via in situ electron microscopy, optical microscopy, and neutron diffraction

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Tungsten-particle reinforced copper composites (Wp-Cu) have been identified as candidate materials to address one of the foremost challenges of fusion research: developing materials that can maintain strength under the intense thermal loads and neutron fluxes experienced by divertor components. Wp-Cu composites show improved toughness and thermal conductivity over standard polycrystalline W, and, importantly, are readily available with a mature supply chain from decades of use in the welding and electronics industries. Wp-Cu has so far performed well in high temperature and post-irradiation testing, and their bulk mechanical properties are well documented, but there is limited information describing their micromechanical deformation behavior. This information is critical if these materials are to one day gain regulatory approval for use in a fusion power plant. In this work, commercially available Wp-Cu composites were subjected to tensile loading while the phase-specific response was observed by different techniques. In situ scanning electron and optical microscopy provided novel insight on load portioning, crack initiation and propagation, and shear banding effects. In situ neutron diffraction was performed at elevated temperatures to investigate the ductile-phase toughening ability of Wp-Cu at fusion relevant conditions through the analysis of lattice strain and dislocation density evolution. The results presented here provide a clearer picture of the microstructural phenomena of Wp-Cu composites under loading and further validate them for thermomechanical applications in nuclear fusion reactors.

Tentative

P3-039

Microstructural evaluation of ductile phase toughened tungsten heavy alloys

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As fusion designs come closer to fruition, there remains a pressing need for materials capable of withstanding the extremes of the reactor interior. In its present state, the current leading divertor material candidate, monolithic polycrystalline tungsten, will embrittle and produce large-scale network cracks leading to eventual component failure. To address this behavior, we propose to toughen tungsten through the addition of a ductile phase to serve as a crack-arrest layer and material compatibilizer. In prior published works, we have evaluated the microstructure, properties, and performance of a ductile phase toughened W-Ni-Fe alloy, and in this presentation will provide in-depth discussion on our recent characterization and material fabrication efforts at Pacific Northwest National Laboratory.

This talk will focus on two primary areas of research: detailed microstructural evaluation of W-Ni-Fe radiation tolerance under high temperature ion irradiation, and recent progress into the compositional design and fabrication of novel low-activation tungsten heavy alloys. Characterizing and quantifying the phase dependent radiation damage morphologies and densities in these two phases systems will be crucial in understanding material evolution during service and provides valuable experimental datapoints for service lifetime prediction. At the same time, we are pursuing novel ductile phase toughened tungsten heavy alloys with compositions tailored to provide a high fracture strength in combination with lower overall activation and compatibility with divertor components. These novel material systems are intended to overcome existing divertor component limitations and are the subject of ongoing experimental evaluation.

Tentative

P3-040

Fabrication of High-Performance Tungsten-Based High-Entropy Alloy Complex Structures via Gel Casting and Sintering Additives

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In fusion reactors, liquid lithium-based surface composites with a liquid-solid structure enhance divertor performance under extreme conditions. However, fabricating capillary porous systems (CPS) using tungsten-based high-entropy alloys (HEAs) presents significant challenges, including pore formation, high shrinkage, cost limitations, and constrained performance. Therefore, innovative preparation and processing techniques are essential. This study proposes a gel-casting method for fabricating WTaTiV HEAs with TiV as a low-melting-point sintering aid. The powder characteristics were optimized through high-energy ball milling, and a paraffin-stearic acid-polyvinylpyrrolidone binder system was employed to enhance slurry stability. During sintering, the TiV phase filled potential pores or cracks in the WTaTiV HEA, improving plasticity while maintaining strength, achieving a compressive strength of 1770 MPa and a strain rate of 13.6%. Microstructural analysis revealed uniform elemental distribution, reduced oxidation, and minimized cracking. The fabricated CPS components exhibited high structural integrity, uniform shrinkage rates, and precise capillary pore dimensions. This gel-casting method enables scalable production of complex HEA components with tailored properties. The low-melting-point phase acts as a sintering aid, enhancing the sintering activity of refractory metals and improving the mechanical performance of the final products, thereby demonstrating the feasibility of pressureless sintering for refractory metals.

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Tentative

P3-041

Preparation and characterization of large-scale potassium-doped tungsten alloy

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The potassium-doped W (W-K) is deemed an attractive potential plasma-facing materials (PFMs) for future fusion reactors. In the present work, a newly developed large-scale W-K plate was prepared by powder metallurgy plus hot rolling. It is worth mentioning that the distribution of K bubbles with a mean size of ~68.5 nm and a number density of ~7.6 10^{18} m^{-3} is not random but presents a specific string arrangement along a certain direction, called K bubble string. In aspect of mechanics, the W-K exhibited high tensile strength and large plasticity simultaneously, e.g. 720 MPa and 28.6 % at 500 °C respectively. For high-temperature stability, the recrystallization temperature of W-K exceeds 1800 °C, which is much higher than 1300 °C of pure W (PW). In irradiation damage, the dislocation loops produced in W-K after high-energy heavy ions irradiation are ~2.5 times lower than in PW, ~3.5 10^{22} and ~8.6 10^{22} m^{-3} respectively, which results in better mitigation of deuterium blistering, retention and irradiation hardening in W-K compared to PW. For transient thermal shock, the cracking thresholds of W-K is 0.44-0.55 GW m^{-2}, which is higher than 0.22-0.44 GW m^{-2} of PW. Especially, the WK-Cu monoblock mock-ups as the divertor component were prepared and all mock-ups successfully passed the steady-state thermal shock test at 20 MW m^{-2} for 1000 cycles. Besides, the effect of K doping on the microstructure, mechanical properties, recrystallisation resistance, high-energy heavy ion damage, deuterium plasma irradiation and high thermal load was also revealed in detail.

Tentative

P3-042

Engineering Advanced Copper Alloys for Fusion Energy

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In next-generation fusion reactors, Cu-based alloys are widely recognized as important heat sink materials. The extreme fusion environment, including elevated temperatures, mixed radiation fields, and complex mechanical stresses necessitates that heat sink materials need to be designed to maintain creep resistance, mechanical strength and thermal conductivity during operation. In this study, we employ direct current sintering to fabricate precipitation strengthened Cu-Cr-Zr-Nb alloys and expose these materials to elevated temperatures to engineer the microstructural and precipitate networks to optimize physical properties (namely hardness and thermal/electrical conductivity). Specimens were fabricated using powders with Cu-2Cr1.35Nb-0.15Zr wt.% composition. The as-sintered specimens were annealed at 970 °C for 20 mins, and subsequently thermal aged at 500 °C for different time periods (20-6000 mins). The microstructure was probed using synchrotron-based XRD, SEM, and complemented by physical properties measurements including electrical resistivity and Vickers microhardness. XRD confirmed the presence of Cu (nm-sized and course precipitates), Cr and Cr_2Nb precipitates. SEM micrographs revealed multi-modal precipitate networks with uniformly dispersed Cr2Nb precipitates within the interior of Cu grains and coarse Cr precipitates at grain boundaries. The Vickers microhardness reaches a maximum at a critical aging time of 110 mins due to the formation of nm-sized Cr precipitates. The electrical resistivity increases with increasing aging time, consistent with the recovery of as-sintered defects (point defects and dislocations). These results help to define the optimal time and temperature windows leading to desirable microstructures in these important heat sink materials.

Tentative

P3-044

Sample temperature dependence of the post-plasma thermal evolution of He bubbles in tungsten and tungsten-based refractory alloys

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The thermal evolution of He bubbles in tungsten (W) and tungsten-based alloys can have important consequences for plasma-material interactions relevant to ITER and beyond. In this work, bulk W, sputtered W, W-5%(wt)Ta, W-3%(wt)Cr and W-4%(wt)Ta-3%(wt)Cr were exposed to a low-energy He plasma in the MagPIE device at a low temperature (LT) of 573 K and high temperature (HT) of 1050 K. The dependence of He bubble thermodynamics on the sample temperature was studied using Grazing-Incidence Small-Angle X-ray Scattering (GISAXS), Thermal desorption spectroscopy (TDS), and in-situ TEM annealing. TDS for all LT samples show low amounts of He desorption that disappears beyond 800 K. TDS of the HT samples show significant desorption an order higher than the LT samples and at temperatures up to ~1400 K. TEM shows that large bubbles present in HT samples, are absent from the LT samples. WTaCr shows a lower concentration of large bubbles. During in-situ TEM annealing, these large bubbles disappear at 1073 K, coinciding with a desorption peak observed in the TDS. Bubbles observed in the LT samples were stable up to 1073 K.

GISAXS on samples annealed at 773 K, 998 K and 1073 K support the TEM results. LT samples showed no change in the bubble radii distribution. A rightward shift in the bubble radii distribution was recorded in the HT samples with increasing annealing temperatures. These results help explain temperature-dependent plasma-material interactions such as recrystallisation suppression and have characterised the He behaviour of W-alloys for post-ITER considerations.

Tentative

P3-045

Effect of baseplate preheating on the Laser Powder Bed Fusion of W-W2C

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Additive manufacturing (AM) enables the realisation of formerly impossible design concepts for high heat flux components aimed at use in a fusion reactor. Laser Powder Bed Fusion (L-PBF) is an AM technique used to create three-dimensional metallic objects by selectively melting layers of powder, even of refractory metals like tungsten. However, challenges arise due to tungsten's high melting point, viscosity, excellent thermal conductivity, and high ductile-to-brittle transition temperature, leading to defects like cracks and porosity in the finished parts. To address these issues, a combined approach of using an active oxygen-getter alloying strategy and baseplate preheating has been proposed to eliminate cracks in tungsten parts [1]. The addition of carbon as such alloying element and in-situ formation of W-W2C composite material has shown potential as an armour material for the diverter [2].

This study investigates the effects of preheating the baseplate to temperatures above the ductile-to-brittle transition temperature, specifically up to 1000 °C, while adding 4 at.% carbon during L-PBF of tungsten. As carbon reacts with tungsten during solidification, it forms carbide precipitates at grain boundaries, resulting in significant grain refinement and a transition from planar to cellular solidification. Preheating to 600 °C and 1000 °C improved densification, achieving nearly full density at 1000 °C. Although microcracks were not entirely eliminated, both crack length and density were significantly reduced.

[1] B. Vranken et al. Additive Manufacturing 46 102158 (2021)

[2] S. Novak, et al. Mater. Sci. and Engin.: A, 772, 138666 (2020)

Tentative

P3-046

Comparison of deuterium retention in rolled and recrystallized tungsten with and without pre-damage

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Plasma-material interaction is a critical issue that influences the steady-state operation of fusion devices. As the most promising candidate for plasma-facing material in thermonuclear fusion devices, the performance of tungsten under plasma exposure and neutron irradiation is a critical issue for tritium self-sufficiency and operational safety of future fusion devices. In this work, the deuterium behavior of rolled (Ro) and recrystallized (Rx) tungsten with and without pre-damage by energetic iron ions was investigated. Slow positron annihilation-Doppler broadening spectroscopy, scanning electron microscopy, and thermal desorption spectra were used to characterize vacancy-type defects, deuterium-induced surface blistering, and deuterium retention of the investigated tungsten samples, respectively. The results show that the undamaged Ro and Rx tungsten have different types of vacancy defects and deuterium traps, whereas identical type vacancy defects and deuterium traps are found in the pre-damaged Ro and Rx. Changing the pre-damage distribution does not affect the type of deuterium traps in both Ro and Rx, further demonstrating that the given material state has little effect on the generation of irradiation-induced defects at room temperature. It is implicated that the impact of the material state on deuterium behavior is insignificant in future plasma-facing material where neutron irradiation will introduce numerous crystal defects.

Tentative

P3-047

Prospects for plasma-facing material research on the KTM tokamak for future fusion power plants

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[1]Institute of Atomic Energy Branch of the National Nuclear Center of the Republic of Kazakhstan

Invitation

Developing materials capable of withstanding the extreme conditions of fusion reactors is a key challenge in creating controlled fusion-based facilities. The KTM (Kazakhstan Tokamak for Material Testing), located in Kurchatov, Kazakhstan, serves as an experimental platform for testing plasma-facing materials (PFMs) under conditions expected in future industrial-scale fusion facilities (FF).

KTM's main design parameters include a plasma current of 750 kA, discharge durations of up to 5 s, a toroidal magnetic field of 1T, a plasma density of $5\times10^{19}m^{-3}$, and a divertor heat load reaching 20 MW/m^2. The facility enables the simulation of thermal and erosion effects in divertor regions and the first wall of future fusion reactors (FR), study of hydrogen isotope retention and release processes.

The research procedure includes initial sample characterization using XRD and SEM techniques, followed by exposure in the KTM divertor zone. Plasma exposure is conducted in three modes: quasi-stationary (up to 5 s to assess cumulative effects), pulsed (simulating instabilities), and with variable plasma composition (replicating real conditions in next-generation tokamaks).

Post-experimental studies include microstructural analysis to assess surface and depth changes, and thermodesorption analysis to study hydrogen isotope accumulation and desorption. This is important for predicting material durability and minimizing fuel leakage.

Thus, the proposed methodology shows that the use of the KTM tokamak to study PFM allows simulating operating conditions, which is extremely important for predicting their durability and efficiency in future FR. Planned tests include tungsten, lithium-based capillary-porous systems (CPS), and other candidate materials.

P3-048

Formation, Surface Characterization, and Stability of Boron-Containing Films Under Plasma Irradiation

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Boron films are promising plasma-facing materials for fusion reactors, offering oxygen gettering and tungsten protection. We have successfully deposited boron films using thermionic-assisted DC sputtering and characterized them ex-situ via Raman spectroscopy. Initial results highlight structural variations, motivating further optimization of deposition conditions.

To enhance film performance, we are refining plasma geometry and gas mixtures (He, Ar, He+H_2, He+D_2) to achieve (a) higher deposition rates, (b) reduced impurity incorporation, and (c) controlled formation of B, B-H, and B-H-O films. Plasma irradiation in H_2, D_2, and oxygen-admixed conditions will assess film evolution under reactor-relevant environments.

Ex-situ Raman microscopy, SEM and EDS are used to evaluate film composition and morphology, with in-situ Raman under consideration. Thermal desorption analysis (TDA) may be incorporated to assess hydrogen retention. Additionally, a pulsed laser deposition setup is in development, with preliminary results expected.

This work provides new insights into boron film formation, stability, and adaptability as a plasma-facing material. The presentation will cover deposition strategies, structural analysis, and future improvements for optimizing boron films in fusion applications.

Tentative

P3-049

Recrystallization behavior of tungsten for the ITER divertor outer vertical target

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The tungsten is the plasma-facing material of the ITER divertor. The R&D and manufacturing of the outer vertical target (OVT) of divertor are ongoing in Japan. During the operation of ITER, tungsten will be exposed to the high heat flux up to 20 MW/m^2 during transient events such as VDE (Vertical Displacement Event). Recrystallization is expected during transient events, which will cause degradation of the material properties. Therefore, understanding the recrystallization behavior is necessary for evaluating the integrity and life of the material and component. There are two potential tungsten grades for ITER divertor OVT, and the recrystallization behavior of these materials were investigated in this work. The grain structure and Vickers hardness of tungsten samples with heat treated at 1000-2400 C for 1-10 h were investigated. Both tungsten grades showed a decrease in Vickers hardness to approximately HV350 after heat treatment at > 1400 C for 1 h, and grain growth was also observed. The difference in hardness change behavior depending on tungsten grade was observed after heat treatment at below 1350 C and the details will be presented.

Regarding the manufacturing process of OVT, tungsten is subjected to the heat history such as bonding tungsten to the cooling tube. The influence of heat history during manufacturing processes of OVT on tungsten, which is evaluated based on the results of the recrystallization behavior investigation will be presented.

Tentative

P3-050

Research summary on the self-passivating tungsten alloy for the prospective application in a fusion power plant

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Tentative

Self-passivating tungsten alloys (SPTAs) is promised to be a solution for the first wall to guaranteeing the safety operation of the future fusion reactor in case of the loss-of-coolant accident and vacuum break. Considering the prospective application in future, microstructure and performance optimization, up-scaling production and high-quality joining with low activation steel were performed to qualify the SPTA concept for fusion.

The production of SPTAs bulk includes powders preparation and sintering. Here will present an optimized powder production process that can obtain nano-crystalline, homogeneous SPTAs powders using mechanical milling. Intensive works have focused on the SPS process optimization by clarifying the roles of current and pressure on the densification and microstructure evolutions of SPTAs bulk. It has confirmed that the electric current and pressure play a comparable role in enhancing the densification process and accelerating the grain growth. The local overheating is the reason results in the Cr-rich phase precipitation and abnormal grain growth. The pressure loading can mitigate these issues. The high-temperature oxidation tests was designed to study the effects of the geometric structure (surface roughness, curvature) and microstructure (pores, grain boundaries, Cr-rich phase, etc.) on the oxidation performance of SPTAs bulk. In addition, the long-term oxidation performance will be also discussed.

Towards industrial application, up-scaling production of SPTAs sample with a dimension of O105 mm× 7.5 mm and a density of ~98% was successfully produced using SPS technique. The SPTAs/CLAM joint with high strength of 295±6 MPa was manufactured by SPS following one-step preparation route using raw powders.

P3-051

Experimental and Computational Study of Hydrogen Permeation in Tungsten with Copper Coated Surface

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The hydrogen isotope transport properties-permeability, diffusion coefficient, and solubility constant-in materials are crucial to the design and operation of nuclear fusion reactors, particularly with respect to radiation safety, material integrity, and fuel cycle sustainability. While extensive research has focused on bulk materials, the effect of interfaces on hydrogen transport remains insufficiently understood. Since fusion reactor components are typically composed of multiple materials with interfacial structures to meet reactor design requirements, a thorough understanding of these effects is essential for developing accurate predictive models and optimizing material performance. In this study, we compare the hydrogen transport properties of pure tungsten (W) with those of heterogeneous metal bilayers to elucidate the influence of interfaces on hydrogen permeation. A thin copper (Cu) film of the order of 100 nm thickness was deposited on tungsten substrate to form bilayer structures. Permeability was measured using a hydrogen permeation measurement system (HPMS) based on gas-driven permeation, where hydrogen gas is applied to one side of the specimen and the permeated hydrogen is detected on the other side in vacuum conditions.

The permeation experiments indicate that the steady-state permeability was slightly decreased by the addition of the Cu film, while almost no change was observed when the W substrate side was exposed to hydrogen. These experimental results were compared with simulation results obtained by a rate model parameterized by atomistic simulations, and reasonable agreement was confirmed. The simulation results also indicate that a large amount of hydrogen is segregated at the surfaces and interface.

Tentative

P3-052

Innovative STEAM PFC for steady Ultra-High Heat Flux Tokamak Divertors

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We present an innovative STEAM heat shield aimed for (attached) tokamak divertors. Tested in air plasmatron jet slowly increasing power, after minor surface melting identifying the critical heat flux limit, it absorbed 32 kW with epicentral 43 MW/m^2 for continuous 22 minutes without further damage and yet no observed time limit. This was thanks to finned copper plate and impacting water allowed to boil. Tricky measurement of so high heat flux involved several independent thermovision, calorimetry methods and heat transfer simulations. Significant improvements of STEAM are ongoing via further geometry optimization, increasing the water speed and pressure (so far only few bars inlet). This may relieve the long-standing thermonuclear fusion devices challenge of plasma load in tokamak divertors suppressed below 20 MW/m^2.

Tentative

P3-053

Dislocation-controlled plastic deformation characteristics of W-2.0wt% Y_2O_3

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Tungsten (W) and W-based materials have broad development prospects in the application of first wall materials for fusion field, but high DBTT (ductile brittle transition temperature) is a major obstacle in their practical application. The higher DBTT of W is caused by many factors, among which dislocation movement plays a very important role, so it is of great significance to study the evolution and movement of dislocation structure.

Adding the second phase particle Y_2O_3 to W is an effective pathway to reduce the DBTT of W. In this paper, W-2.0wt % Y_2O_3 was used for annealing at 300 °C and 600 °C to study the dislocation evolution near and above DBTT temperature. In the study of dislocation morphology and distribution in W matrix, it is found that the plasticity of W at higher temperature is dominated by free dislocation movement, and the reduced resistance to dislocation movement at higher temperatures can even penetrate grain boundaries. Dislocation loops are densely distributed in W matrix, but their mechanism needs to be further investigated.

Y_2O_3 is a ceramic material with higher DBTT than W, which means it is not prone to plastic deformation. However, it is worth noting that in this research, dislocation slip lines appear in Y_2O_3 particles, that is, plastic deformation occurs. In this paper, the two phases are connected by a disordered amorphous transition region, which indicates that the amorphous transition region plays a coordinating role in the two-phase deformation.

Tentative

P3-054

Adaption of wire-based laser metal deposition for the regeneration of tungsten armored plasma-facing components

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Due to its thermophysical properties, tungsten is currently considered a suitable plasma-facing material (PFM) for fusion applications. Despite its comparatively high durability, the service life of tungsten armored plasma-facing components (PFCs) is limited due to erosion and exposure-related damage. To extend the time to failure, the PFM can be regenerated using wire-based laser metal deposition (LMD-w). Although the processability of tungsten with LMD-w has already been demonstrated, it is highly sensitive to inconsistencies in the ambient conditions. In order to utilize it in regular manufacturing and regeneration processes, the reproducibility must be ensured.

The crucial influencing factors on the manufacturing results of LMD-w of tungsten were identified and categorized into discrete and continuous parameters. The continuous parameters include laser power, machine feed rate and wire feed rate. To determine their sensitivity, a fractional factorial experimental design was applied with a commonly used LMD-w setup. The substrate geometry was chosen analogous to the expected plasma-facing surface area of divertor monoblock PFCs. Among the discrete parameters, the setup of laser optics, shielding gas and wire position were categorized. To determine the sensitivity to changes in the discrete parameters, a shortened experimental design with variable laser power was used. To evaluate the experiments, the process limits as well as the mechanical and microstructural characteristics of successful deposited tungsten coatings were examined. Based on the results, estimations for the adaption of wire-based laser metal deposition for a stable tungsten manufacturing process can be made.

Tentative

P3-055

Thermal performance of W/Cu flat-type components as the lower divertor target in EAST

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W/Cu flat-type component is one of the candidates for plasma-facing components for the future fusion devices, owing to its high heat load exhaust capacity, making it a focus of significant research interest. Since 2021, high performance W/Cu flat-type components have been tested on outer horizontal targets of the lower divertor in EAST. However, many damages, i.e. melting, cracking and exfoliation were found after cyclic plasma discharges. To evaluate the thermal performance of W/Cu flat-type components, especially the maximum parallel heat flux they can withstand, Fluent is employed to calculate the temperature distribution.

Such simulation considered actual experimental conditions including heat flux distribution, where installation misalignment was within 0.6 mm, the heat flux incident angle was no more than 5°, the incident angle was below 3.7°. For W/Cu flat-type components with chamfer angles of 3.76° and 5.19°, the parallel heat flux required to melt OFC is always lower than that for W, indicating that OFC may be melted first. However, for the component with 2.4° chamfer, OFC also melts first when the heat flux incident angle exceeds 2.5°, while W melts first when the incident angle is below 2.5°. Additionally, a linear relationship between the maximum temperatures of W and OFC was founded, and this linear function varies with the incident angle, which provide a basis for monitoring the temperature of OFC in EAST's flat-type components.

Such analysis offer key data for the plasma operation of EAST, and provide unique reference for application of W/Cu flat-type components in other devices.

Tentative

P3-056

PLASMA MATERIAL INTERACTIONS IN ADVANCED TUNGSTEN BASED MATERIALS FOR FUTURE PLASMA FACING COMPONENTS

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Tentative

Tungsten (W) is a prime candidate material for plasma facing components (PFC) in future fusion energy systems. Unfortunately, it is widely known that there are significant issues facing the use of W; hard workability, high ductile to brittle transition temperature, overall brittleness, low recrystallisation temperature to name a few. Therefore, attention has moved towards novel W-based materials including W-based alloys, composite fibre materials and the field of additive manufacturing (AM). Particularly, AM W is allowing the fabrication of complex shapes and microstructures that holds promise to open a new paradigm for PFC designing. Moreover, potassium (K) doping of W is known to improve radiation tolerance due to delaying radiation induced grain growth. A missing link with these materials is that while plasma material interactions (PMI) with pure-W is well documented, little is known regarding PMI in AM W and K-doped W. Using PSI-2 linear plasma device with state-of-the-art microstructure characterisation we screened the performance of different W materials: e-beam AM W, K-doped W and reference ITER-grade W. We will reveal the effect of deuterium plasma particle flux and sample temperature on near surface material degradation such as susceptibility to blistering, flaking, and cracking. The results will also show the effect of He/Ar seeding in the plasma on the relative performance of e-beam AM W and K-doped W over a wide temperature range (upto 800°C). These results will reveal the role of key microstructural features, such grain orientation and nanometric potassium bubbles on the PMI scenarios for this class of materials.

P3-057

Alloy development for metal additive manufacturing through a combinatory method using laser powder bed fusion

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Additive manufacturing (AM) of metallic components attracts great interests from both academy and industry. The recent development of combinatory methods using laser engineered net shaping (LENS) technology shows significant accelerated screening of alloys in terms of their printability, mechanical properties, and microstructural evolutions. However, the use of laser powder bed fusion (LPBF) technology for alloy screening is till rare in the metal AM research field. Owing to the better parts accuracy, faster heating and quenching process, and lower powder usage of LPBF compared to LENS, it's highly demanded to develop the combinatory method for LPBF. In this presentation, our recent work on rapid alloy screening through in-house developed combinatory method using LPBF will be demonstrated for several alloy systems including, light weight high strength steel, light weight high entropy alloys, and compositionally graded structures.

Tentative

P3-058

Deuterium retention in recrystallized tungsten exposed to high-flux plasma with fluences up to 1×10^{29} m^{-2}

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Plasma fluence at the divertor of a future magnetic confinement fusion device can accumulate up to ~10^{28}-10^{29} m^{-2} per year. Yet hydrogen isotope retention under such high-fluence plasma exposure has been rarely reported. To investigate deuterium (D) retention in tungsten (W) exposed to such high-fluence plasma, a series of high-flux D plasma exposures were preformed using recrystallized W samples at ~500 K in Magnum-PSI. The highest fluence achieved was ~1×10^{29} m^{-2}. Surface morphology observations indicate an initial increase in the number of blisters at the sample surface with increasing fluence, followed by saturation at ~1×10^{29} m^{-2}. Multiple bursts of blisters with open cracks or edges were observed under the two highest fluences of ~1×10^{28} m^{-2} and ~1×10^{29} m^{-2}. ^{3}He nuclear reaction analysis (NRA) shows a maximum D concentration up to 0.012 at.fr., distributed within the first 4 μm from the sample surface under the highest fluence. D retention, as measured by NRA and thermal desorption spectroscopy, tends to saturate with increasing fluence. Simulations of D_2 thermal desorption, performed using the TMAP rate equation code, show a maximum D trapping depth of ~10 μm, consistent with the defect depth profile revealed by transmission electron microscopy. D retention saturation observed in this work is attributed to the sample surface morphology modifications and the saturation of plasma-induced defects. This investigation provides a valuable reference for understanding the evolution of total hydrogen isotope retention in W under high-fluence plasma exposure in future fusion devices.

Tentative

P3-059

Performance of doped tungsten with high potassium content under high-flux helium plasma irradiation

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Doped tungsten with a high potassium content of 80 ~100 ppm has exhibited great potential in applications of future plasma facing materials with exceptional advantages including high conductivity and hardness, high recrystallization temperature and low ductile-to-brittle temperature. This study examines the effects of high-flux helium plasma irradiation on the 13.7 mm thick doped tungsten plate fabricated by powder metallurgy and hot rolling through a mass-production route in the real industry. Results show that a nanolayer forms on both doped tungsten and pure tungsten under the helium plasma energy of 50 eV, a flux of 10^{22} $m^{-2}s^{-1}$, and a fluence of about 10^{25} m^{-2}. The surface nanostructure is thinnest on the rolling-direction cross-section and thickest on the transverse-direction cross-section. Overall, the doped tungsten demonstrates superior irradiation resistance, forming a nanofilament layer of 0.23 μm that is 41% thinner than pure tungsten (0.39 μm). This enhanced resistance is attributed to the combination of grain refinement and sink effect provided by nano-structured potassium bubbles.

Tentative

P3-060

Atomistic simulations of cumulative irradiation damages in single-crystalline tungsten and analysis of secondary defects

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In this study, the irradiation defects were cumulatively generated in tungsten via invoking a series of primary knock-on atoms to mimic neutron irradiation to the plasma facing material using molecular dynamics simulations. As irradiation accumulated, it was observed that dislocation loops of several different types were created and the preferred types depend on the loop size and the defect source, either vacancies or interstitials. It was also prevalent to form complex sessile dislocation structures. This variation and population in dislocations was not observed when the sample was irradiated one time with the same amount of net excessive energy. The continuous irradiation with smaller amount of energy is more effective to generate various irradiation damages.

Tentative

P3-061

Heterogeneous recrystallization behavior of warm-rolled yttria dispersion strengthened tungsten plate along thickness direction

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Tungsten-based materials are considered to be one of the candidate materials for the first wall and diverter of future thermonuclear fusion. The plastic deformation of tungsten during rolling may lead to a heterogeneous microstructure, which will affect the recrystallization process. The main aim of this investigation is to study qualitatively the influence of the heterogeneous microstructure along the thickness direcrtion on the recrystallization behavior of the rolled W-Y_2O_3 plate. Electron backscattering diffraction (EBSD) was used to characterize the grain size, sub-grain size, geometrically necessary dislocation energy, nuclei density and texture of the surface, quarter and center layers of the tungsten plate in the rolled and recrystallized state, respectively.

Tentative

P3-062

Review on design strategy and scalable production route of tungsten-based materials for nuclear fusion application

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Tentative

Development of high-performance material and its engineering process route is critical for the operational of the future fusion reactors. Tungsten (W) is the main candidate of plasma facing materials as it is resistant to erosion, has high melting point and exhibits peaceable transmutation under neutron irradiation. However, W is intrinsically brittle and faces further embrittlement in service condition. In this contribution, internal and external toughening strategies to overcome these issues including alloying, second-phase doping, plastic processing and fibre toughening W-based material will be overviewed.

The alloying element of Rhenium aims to improve the mobility of dislocation, Vanadium is added to inhibit grain growth and achieve solution strengthening, and the second-phase particle of yttria targets additionally to achieve dispersion strengthening. Deformation processing of rolling or swaging aims to future increase the density of the W-based powder materials and increase the amount of the movable dislocation. Fibre toughening can maintain function as long as the interface exists. Thanks to these strategies, the toughness of W-based materials has been significant improved.

In summary, the process maturity is now high enough for industrial upscaling. The pilot production line of advanced W-based materials has been built, and reaching the 100 kg level. The corresponding flat-type mono-block has been produced using a joint techniques of vacuum casting and brazing. Here will show the latest test results on thermal loading and service performance at EAST.

P3-063

Current Status of High Heat Flux Testing at the HELCZA facility

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HELCZA is a High Heat Flux (HHF) test facility equipped with an electron beam gun of 800 kW power which is dedicated for testing large, full-scale mock-ups and plasma facing components for fusion reactors. Recently the HELCZA facility finished an extensive test campaign of beryllium Full-Scale Prototypes of the ITER First Wall and Alternative Design Mock-Ups. Over a period of more than two years, this campaign rigorously evaluated the performance of the HELCZA facility in conducting fatigue testing under HHF conditions, with heat fluxes reaching up to 3 MW/m2 on beryllium components. The successful outcome of this campaign demonstrates facility suitability for such demanding testing scenarios.

Due to ITER's transition from Beryllium to Tungsten First Wall, the facility encounters new challenges. To enable the HHF testing of tungsten plasma facing components like the Inner Vertical Target (IVT) of the ITER divertor, while avoiding health and safety concerns associated with Beryllium, the comprehensive decontamination of the entire facility, all the auxiliary equipment and the HELCZA site was performed in 2024. As a next step, the efforts at the HELCZA facility are now focused on the qualifications and then subsequent HHF testing campaigns of large tungsten components such as IVT test assemblies.

Disclaimer: The work leading to this publication has been funded by Fusion for Energy under contracts F4E-OPE-319-01&02&03. This publication reflects the views only of the authors, and Fusion for Energy cannot be held responsible for any use which may be made of the information contained therein.

Invitation

P3-064

Effects of disordered interfacial transition layer of W-Y_2O_3 on irradiation and mechanical properties

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W-0.5 wt.%Y_2O_3 was chosen due to the importance of the interfacial transition in enhancing tungsten's performance. This study revealed the presence of a transition layer with a disordered atomic arrangement between the two phases, unlike the bonding mode of traditional heterogeneous phases. This transition layer helps coordinate the interfacial lattice matching between Y_2O_3 particles and the matrix, leading to better compatibility between these two phases. This departure offers a new approach to second phase doping modification.

Low-energy irradiation experiments were conducted to investigate the impact of the interface between Y_2O_3 and W matrix on irradiation performance. The results revealed the presence of numerous small helium bubbles within the Y_2O_3 post-irradiation. Helium bubbles at the interface were observed to be larger in size and elongated along the interface, appearing to be concentrated in the disordered transition layer, indicating the effective capture of helium by this layer.

Additionally, unirradiated and irradiated samples were specifically selected for micro-column compression experiments to further examine the influence of the interface on mechanical properties. During micro-column compression, the interface bond proves to be robust, with the Y_2O_3 particles able to crack without causing the interface to detach, showcasing the effective force coordination role of the disordered transition layer.

Template

P3-065

Enhanced resistance to helium irradiations in nano-structural high-entropy-alloy plasma-facing materials

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Finding high performance PFMs is a challenging task for realizing commercial fusion reactors. Herein, we found the nanochannel CrMoTaWV high entropy alloy (HEA) [1], CrMoTaWV/W multilayer [1], and amorphouse TiZrHfTaW [3]films are highly resistant to low-energy and high-flux He plasma exposure comparing to bulk W. (1) The nanochannel HEA film has 20 times higher initial fluence for the formation of fuzz and a remarkable 8.9 times slower fuzz growth rate than those of W. (2) The thickness transformed to fuzz in the HEA/W multilayer films was about two-thirds of those in the CrMoTaWV film. The fuzz growth rates in multilayer films are lower than the average growth rate of bulk W and HEA films. (3) The initial fluence in TiZrHfTaW film is increased greatly to 20 times higher and the length of fuzz irradiated to the fluence of 3×10^{26} ions/m^2 is 28 times shorter than those of bulk W.

These findings highlight different promising routes for the exploration of high-performance PFMs.

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Tentative

P3-066

In-operando spectroscopic ellipsometry measurement of W surface during high-flux plasma exposure

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Plasma bombardment onto a material can lead to surface morphology changes, which can then affect material properties. For instance, tungsten (W) fuzz can reduce the thermal diffusivity, sputtering yield, and secondary electron emission, and also increase the probability of occurrence of arcing. Thus, in-operando surface monitoring during plasma discharge is desired to better understand and control plasma-material interactions.

We examined the capability of spectroscopic ellipsometry (SE) first in an ex-situ setting. In short, SE analysis yields two parameters, Psi and Delta, as a function of wavelength, in a range of 400 nm - 1000 nm in our system (J.A. Woollam: iSE), which characterize the change in polarization state of the probe light beam caused by reflection from the sample. Therefore, Psi and Delta are influenced by structures mainly on the surface for an absorbing material, such as metals. Ex-situ SE measurements of W fuzz revealed that the W fuzz thickness can be evaluated from a change in Delta with the fuzz growth. In comparison with W fuzz, where Delta monotonically decreases with increasing fuzz thickness, (Psi, Delta) plots of Cr cones behave differently; Delta first decreases and then increases with increasing cone height. This indicates that SE analysis can distinguish fuzz and cone structures.

The SE system has then been installed on the PISCES-RF linear plasma device for in-operando surface measurement during simultaneous high-flux plasma exposure and high-energy heavy ion beam bombardment. Results of in-operando SE measurements will also be presented in the conference.

Tentative

P3-067

Overview of the mechanical and thermal properties of plasma sintered $Be_{12}Ti$

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Beryllium intermetallic compounds, beryllides, are considered promising materials for neutron multipliers in fusion reactor breeding blankets due to their minimal swelling characteristics and lower reactivity with water vapor at high temperatures compared to beryllium. In the latest JA DEMO reactor design, it has been suggested to utilize beryllides in block form rather than as pebbles. Consequently, it is essential to develop a comprehensive database detailing the various properties of beryllide blocks. We investigated the changes in properties of $Be_{12}Ti$ blocks due to sintering temperature using plasma sintering. It is necessary to understand the entire nature of the device and determine the properties of plasma-sintered $Be_{12}Ti$. Thus, this study includes an overview of the properties of plasma-sintered $Be_{12}Ti$.

The beryllide blocks were prepared using the plasma sintering technique with homogenized $Be_{12}Ti$ powder. The sintering processes were carried out at different temperatures of 1050, 1100, and 1200 °C. Sintered density, Vickers hardness, and thermal conductivity were evaluated. From the results, including workability and sintered density, it was found that the optimal sintering temperature was 1200 °C. Subsequently, compression tests with samples sintered at 1200°C were carried out. Hydrogen generation concerning the reactivity with water vapor was also investigated.

Tentative

P3-068

Influence of various types of ionising radiation on the formation of radiation-induced defect centres in advanced ceramic breeder pebbles

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Lithium orthosilicate pebbles with additions of lithium metatitanate as the second phase, also referred to as the advanced ceramic breeder (ACB) material, are currently being developed as the European Union's reference material for tritium breeding in future thermonuclear fusion reactors. In addition to their principal function of tritium breeding and release, the biphasic ACB pebbles must withstand the anticipated harsh operational conditions, including high neutron flux, elevated temperatures, magnetic field, etc. In the present work, the influence of various types of ionising radiation on the formation of radiation-induced defect centres in the ACB pebbles is investigated using electron paramagnetic resonance (EPR) spectroscopy. Typically, neutron irradiation experiments are time-consuming, the handling and transportation of the neutron-irradiated samples is difficult due to the neutron activation of the material, and the experimental costs are high. Therefore, in the present work for the irradiation of the ACB pebbles, several alternative irradiation sources and types of ionising radiation were used: (1) X-rays with energies up to 45 keV, (2) 10 MeV accelerated electrons, and (3) 2.1 MeV accelerated hydrogen ions. Simulations of the experimentally obtained EPR spectra for the irradiated ACB pebbles as a superposition of several individual signals were performed using EasySpin software. Regardless of the ionising radiation type, similar EPR signals of electronic spin S = 1/2 systems with distinctive symmetries, g-factor values, and thermal properties have been separated and described.

Template

P3-069

In situ characterization of 3D deformation and damage evolution of LiH ceramic under different thermal-mechanical load via DIC and DVC

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Lithium hydride (LiH), an important ceramic material in nuclear engineering, is usually choice for the neutron shield and moderator in lightweight nuclear power plants, and tritium breeder material in a deuterium-tritium (D-T) driven hybrid reactors. Understanding the damage mechanism of LiH ceramics is the key to improve its application reliability. The damage mechanism of LiH ceramics is studied by measuring strain field because strain concentration is the starting and accumulation area of damage. Digital image correlation (DIC) and digital volume correlation (DVC) methods are used to calculate the surface and internal strain fields of LiH ceramics under thermal-mechanical load. The surface and internal strain fields under different thermal-mechanical load show that the formation and evolution of localized deformation are related to the microstructure, such as grain size, crystal orientation and defects (pores and inclusions). The results show that the creep and fatigue damage of LiH ceramics originates from the large pores at some grain boundaries where strain localization exists. In the statistical analysis of strain evolution, the elastic strain distribution follows normal distribution and the plastic strain distribution follows lognormal distribution. Interestingly, the peak width of the normal/lognormal distribution and the tail of the lognormal distribution increase with increasing stress. Crystal plastic finite element (CPFE) simulation shows that grain size and crystal orientation are the main factors affecting the strain concentration of LiH ceramics at room temperature.

Tentative

P3-070

Preparation and irradiation performance of advanced tritium breeder material: lithium lead titanate

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The tritium breeder in the blanket needs to meet the requirements of tritium breeding rate (TBR > 1), and also needs to face a harsh environment. In this study, a novel tritium breeder material, lithium lead titanate ($Li_2Pb_xTi_{1-x}O_3$), was designed and prepared by using the new idea of replacing part of the useless element Ti in Li_2TiO_3 with the neutron multiplier element Pb. It can continue to maintain the excellent performance of Li_2TiO_3; meanwhile, it is hopeful to have the effect of neutron multiplication and significantly improve the tritium breeding rate of the fusion blanket.$Li_2Pb_xTi_{1-x}O_3$ powders prepared by the ball milling method were sintered at different temperatures. The samples were irradiated with 100 keV deuterium ions at a dose of $1*10^{15}D$ / cm^2. XRD and XPS experiments were conducted on the samples before and after irradiation, and comparative analysis was carried out. The experimental results show that with the increase of sintering temperature, the grain size of Li2PbxTi1-xO_3 powder increases slightly at first, and then increases rapidly. The optimum sintering temperature is 800 °C. After irradiation, the phases of Li_2TiO_3 and $Li_2Pb_{0.7}Ti_{0.3}O_3$ did not change, but the XRD patterns shifted to the left. Similarly, the peak intensity after irradiation decreased in the XPS spectra. These results indicate that irradiation will cause lattice swelling and ion irradiation will cause damage to the grains of the ceramic samples.

Tentative

P3-071

Study on Tritium Release Behaviors of Li2TiO3 Breeder Irradiated by Fusion Neutrons

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The tritium breeding performance and release behavior of solid breeders are crucial for designing blankets in fusion reactors. The types and densities of defects formed by neutron irradiation with different energies result in different tritium release behaviors. To understand the effect of high-energy neutrons on the tritium release mechanism, tritium release experiments were conducted on Li_2TiO_3 irradiated by fusion neutron source HINEG-CAS and tritium release system. The results indicated that a small amount of tritium release was observed at room temperature, which means the defect self-healing behavior. The peak of tritium release occurred at ~673 K and tritiated water accounted for 80%. The relatively high HTO proportion may be caused by fusion neutron irradiation while not excluding the influence of water vapor. When heating to around 900 K, the water formation reaction led to partial production of HTO, slowing down the rapid decline of the HTO release rate. Further ESR results revealed that the fusion neutron irradiation introduced E'-center defects, acting as trapping sites. Compared to the pre-irradiation, the defect concentration at the left/middle/right positions of the sample increased by 50.3%, 63.9%, and 32.0% respectively after irradiation. Moreover, the tritium release was related to the annihilation of E'-center defects. These findings are beneficial for the optimization design of solid breeders.

Tentative

P3-072

Study on tritium release behavior of ceramic pebbles fabricated by Centrifugal granulation method

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At present, centrifugal granulation is an ideal method to prepare tritium breeder pebbles because of its low cost, easy mass and controllable structure. In this paper, the tritium release difference of tritium breeding ceramic pebbles fabricated by centrifugal granulation method was investigated in detail. Moreover, the tritium release temperature and amount were tested by using Thermal desorption spectroscopy and Liquid Scintillation Counter. The tritium release temperature and activation energy of ceramic pebbles were investigated, respectively.

Tentative

P3-073

Status of Lithium-6 Enrichment using Electrodialysis with Ionic Conductor Membrane

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In fusion power generation, the isotope ratio of lithium-6 in breeding blanket materials is crucial for achieving the tritium self-sufficiency. Lithium-6 is an abundant resource in nature, but it constitutes only 7.6% of natural lithium. Therefore, isotope separation technology is essential to meet the lithium-6 ratio required in a fusion reactor. The amalgam method using mercury has been the only lithium isotope separation technology put into practical use so far. However, it is desirable to use another technology considering the impact of environmental pollution.

At QST, we are conducting research and development on the extraction of high-purity lithium and the separation and enrichment of lithium-6 by electrodialysis using a lithium-ion conductor as a separator membrane. This technology utilizes the difference in diffusion rates of lithium ions in the ion conductor due to their mass. The ratio of diffusion coefficients between lithium-6 and lithium-7 is given by the square root of the inverse mass ratio, which is 1.08, and the separation factor characterizing the actual separation and enrichment performance is slightly lower than this value.

In this presentation, we will discuss how the applied voltage affects the separation factor and the migration rate of lithium, and describe the trade-off tendency between them. Furthermore, we will discuss the scale of the production facility when the cascade theory is applied to the results

Tentative

P3-074

Effect of Pb-16Li on the mechanical properties of ceramic materials after long-term exposure

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[1]Research Centre Rez

Flow channel inserts (FCI) are functional components designed to minimize the magnetohydrodynamic pressure drop and also serve as a thermal and tritium permeation barrier. In the simplified design of the FCI component, the ceramic is directly exposed to contact with Pb-16Li at elevated temperature. As the FCI must withstand mechanical stresses resulting from the thermal gradient during the demanding operation, the assessment of mechanical response of ceramic in contact with Pb-16Li is essential.

The mechanical properties of carbide and oxide ceramics are investigated using the PAMATES facility. As-received (non-exposed) samples are tested using a three-point bending test at room temperature and at 650 °C in argon. To assess the potential effect of mechanical strength degradation caused by Pb-16Li, some samples are pre-exposed to Pb-16Li at 650 °C for 1000 hours prior to mechanical testing in Pb-16Li at 650 °C. Scanning electron microscopy and energy dispersive spectroscopy are used to investigate the fracture surfaces and to document changes in the chemical composition of the tested samples. In this contribution, the mechanical response of the studied materials will be discussed with respect to the fracture morphology of the tested samples. In addition to flexural strength evaluation, the mechanical properties such as nano-/micro- hardness along with the fracture toughness of the as-received and pre-exposed samples is determined.

Template

P3-075

Preliminary Study on Tritium Release Properties of Li_2TiO_3 Pebbles Manufactured by PIM Process

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The solid type of tritium breeder is used as a pebble form due to packing phenomena, stress concentration, thermal conductance, purge gas flow, and so on. The important properties of breeder pebbles are not only physical properties but also tritium release property. This study aims at a preliminary investigation on tritium release properties of lithium metatitanate (Li_2TiO_3) pebbles manufactured by Powder Injection Molding (PIM) process.

Li_2TiO_3 pebbles with 3.43 mm in diameter were prepared by PIM process which was first try to pebble manufacturing by Korea Institute of Fusion Energy (KFE). Although Li_2TiO_3 pebbles with around 1 mm in diameter are foreseen to be used in breeding blanket, those with 3.43 mm in diameter were manufactured as a feasibility test by PIM process. Average grain size and porosity of Li_2TiO_3 pebbles were less than 1.00 μm and about 39.7 % with almost open pore type, respectively. The high-intensity D-T fusion neutron source (HINEG-CAS) was employed at the Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences (INEST, CAS) to conduct a neutron irradiation experiments. The total number of fusion neutrons was about 1.104×10^{15} n. Tritium release experiment from the irradiated pebbles was also conducted using the tritium release system from room temperature up to 800 °C. The total radioactivity of release tritium from the irradiated Li_2TiO_3 pebbles with 274.1 g in weight was about 1866.4 Bq. Tritium release behavior including the ratio between HTO and HT at elevated temperature was preliminarily investigated in this study.

Tentative

P3-076

Tritium production performances of Li_4SiO_4 solid tritium breeder prepared by wet method under reactor irradiation

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In accordance with the conceptual design and operational parameter requirements for the Helium-Cooled Ceramic Breeder Test Blanket Module (HCCB TBM) of China Fusion Reactor, high Li-6 (90%) isotope-enriched lithium orthosilicate (Li_4SiO_4) solid tritium breeder microspheres with qualified theoretical density and sphericity were prepared using a wet method. Based on the China Advanced Research Reactor (CARR), tritium production performance tests of the irradiated microspheres under high neutron flux conditions were completed, obtaining critical irradiation tritium production data such as tritium production rate, tritium release rate, and tritium release form. Additionally, online hydrogen-tritium permeation performance data of the RAFM steel surface composite tritium permeation barrier coatings under in-pile neutron irradiation conditions were acquired. Post-irradiation examinations of the tritium breeder microspheres were completed in hot cell. Important irradiation performance data including the crush load of the irradiated microspheres, Li-6 burnup, and tritium retention were obtained .

Tentative

P3-077

Simulations of tritium release behavior of single-phase and two-phase ceramic breeder materials

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Lithium ceramics, such as lithium orthosilicate (Li_4SiO_4), lithium metatitanate (Li_2TiO_3), and their compounds (xLi_2TiO_3-yLi_4SiO_4), have been studied extensively as tritium breeder materials for fusion reactor blankets. It is very important to clarify the tritium release mechanisms of ceramic breeder materials, whether for blanket design or for tritium extraction system design. Computer simulation is an important means to clarify the tritium release mechanisms, which is complementary to experimental study. On the one hand, computer simulation can help to analyze and interpret the experimental results, and provide guidance for the design of tritium release experiments. On the other hand, the experimental results provide a modeling basis for computer simulation, and the experimental data are used to validate the tritium release model and code. In this study, we have improved the tritium release model for single-phase lithium ceramics such as Li_4SiO_4 and Li_2TiO_3, and proposed a tritium release model for two-phase ceramics xLi_2TiO_3-yLi_4SiO_4. We have also developed a code named SimTRACE to perform simulations of the tritium release behavior of lithium ceramics. In general, the simulation results by SimTRACE are in good agreement with typical experimental results. The SimTRACE code will be further optimized, and our ultimate goal is to develop a design tool that can be used to analyze and evaluate the tritium retention and release behavior in solid breeder blankets.

Tentative

P3-078

Unveiling the in-depth microstructure of the Advanced Ceramic Breeder pebbles: the first comprehensive STEM exploration

Marcelo Roldan Blanco[1], Maria Gonzalez[1]

[1]CIEMAT

Within the European HCPB breeding blanket concept, ceramic breeder pebbles play a decisive role in tritium generation and retention. Despite their technological importance, a thorough microstructural characterization of advanced ceramic breeder (ACB) pebbles has remained elusive until now. This work provides the first exhaustive transmission electron microscopy (TEM) examination of ACB pebbles, both in their as-manufactured state and after being annealed at 900°C marking a level of detail not previously achieved in the fusion community.

The pebbles were studied using S/TEM combined with EDS, allowing the determination of phase composition and defect distribution. In the as-manufactured condition, the pebbles exhibit a complex dendritic architecture, where Ti-rich lithium metatitanate (LMT) dendrites coexist with a Si-rich lithium orthosilicate (LOS) matrix. A statistical clustering modeling from EDS data confirms these two main phases, with minor local variations in composition. A significant topic of this investigation is the effect of annealing on the microstructural evolution of these pebbles, which is essential for identifying possible transient states. Understanding these intermediate states is essential for refining the kinetic pathways of phase transformation and for validating the material structural stability. Both approaches are critical before defining the role of the microstructure on radiation-induced defect formation and bred tritium transport through the pebble.

So far, our systematic TEM studies have evidenced the absence of defects (as clusters of dislocations and bubbles) within as-received and annealed pebble's main structures, anticipating the radiation resistance of ceramic breeder structures perceived by means of indirect techniques.

Tentative

P3-079

Study of microstructural degradation of chromium beryllide under neutron irradiation

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This paper presents the results and analysis of microstructural changes in chromium beryllide ($CrBe_{12}$) under neutron irradiation. The study was conducted to assess the radiation resistance of the material, which is advanced for use in thermonuclear power engineering. The samples were irradiated at WWR-K reactor neutron field at different fluences and temperatures. X-ray diffraction analysis and scanning electron microscopy were used to study microstructural changes. The main features and mechanisms of radiation damage and the evolution of the microstructure of chromium beryllide were revealed. The results obtained demonstrate the level of radiation resistance of the material. The work contributes to a deeper understanding of the behavior of beryllium intermetallic under radiation irradiation and can be used in the development of new structural and functional materials for nuclear and thermonuclear reactors.

The research has been funded by the Science Committee of the Ministry of Science and Higher Education of the Republic of Kazakhstan with Program BR24993119.

Tentative

P3-080

Experimental Quantification of Anisotropy in Effective Thermal Conductivity of Compressed Pebble Beds for Fusion Blankets

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Understanding anisotropy in effective thermal conductivity (keff) is crucial for lithium-based ceramic pebble beds in fusion blankets due to their complex structural and thermal environment. These pebble beds are subjected to intense thermal gradients and neutron fluxes during reactor operation. Due to random packing, mechanical compaction, and preferential pebble orientation, the keff is inherently direction-dependent. Ignoring this anisotropy can lead to inaccurate predictions of temperature distribution, potentially causing localized overheating or underperformance of the breeding material. Moreover, anisotropic behaviour in keff affects the heat transfer efficiency, tritium release kinetics, and overall thermal management strategy within the breeder zone. Accurate characterization of this anisotropy ensures safer operation and reliable thermal modelling of fusion blankets, ultimately contributing to the long-term success of fusion energy systems.

Previous studies on keff under compressive stress have commonly assumed isotropic behaviour of ceramic pebble beds. While this may be valid for perfectly spherical, frictionless pebbles, real systems involve nearly spherical or ellipsoidal pebbles with surface roughness, leading to anisotropic thermal behaviour. To address this, an experimental setup based on the transient hot-wire technique has been developed, featuring two hot-wire placed in both axial and radial directions to accurately quantify anisotropy. Experiments are conducted on lithium titanate pebble beds under compressive stresses in typical temperature range of fusion blankets. The anisotropic keff will provide critical insights into the thermal behaviour of pebble beds under realistic fusion blanket conditions. The experimental setup, along with the results and their detailed discussion, will be presented in this poster.

Tentative

P3-081

Functionalisation of fusion reactor grade structural materials via the physical vapour deposition of ceramic coatings

Tommaso Sgrilli[1], Victor Bellido-Gonzalez[1], Patrick McCarthy[1], Robert Brown[1], Alex Azzopardi[1], Lara Maroto-Diaz[1], Jim Pickles[2], Mike Jackson[2], Sandeep Irukuvarghula[2], Samara Levine[2], Hazel Gardner[3], James Wade-Zhu[3], Emily Curtis[3]

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The selection of structural materials to be included in the design of future fusion reactors entails considerations on how these can sustain extreme environments while maintaining stringent performance requirements. The application of functional coatings has attracted interest, given the variety of properties that can be conferred to a material in this manner. Ceramic coatings are particularly researched, due to their mechanical properties, dielectric nature, and potential as barriers against the permeation of hydrogen isotopes.

In this study, we investigate the approach of applying selected ceramic coatings to fusion-grade structural materials using reactive magnetron sputtering. The regulation of the reactive gas, amongst other parameters, is crucial for obtaining coatings with reproducible characteristics throughout the deposition process. Strategies that are aimed at improving the adhesion strength of the layers include the substrate surface activation via ion bombardment, as well as the addition of tailored interlayers to promote affinity between substrate and coating materials. The incorporation of co-sputtered dopants to the selected coating chemistries is also investigated, with a particular focus on their effect on the crystallinity of the deposited films.

Finally, an attempt of transferring the magnetron sputtering technology to the deposition of coatings onto interior walls of tubes, which would find applications in the heat exchange systems of fusion reactors, is discussed, with an emphasis on the inherent challenges that this type of restricted geometry carries.

Tentative

P3-082

A high performance and dense Fe-Al/Al_2O_3 tritium permeation barrier prepared on 316LSS at low temperature

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Deuterium and tritium are easy to permeate into the fusion reactor internals, and Fe-Al/Al_2O_3 composite coating can effectively hinder the permeation behavior of hydrogen isotopes. However, the temperature in the preparation process of Fe-Al/Al_2O_3 coating in many literatures is high, which will affect the mechanical properties of 316L stainless steel. In this paper, Fe-Al/Al_2O_3 coating was prepared by pack aluminizing, heat treatment and in-situ oxidation at low temperature below 700°C. The aluminized layer gradually transforms from Fe_2Al_5 and $FeAl_3$ to FeAl and Fe3Al during heat treatment. Compared with the aluminized layer prepared at 650°C, the aluminized layer prepared at 600°C is thinner and the heat treatment time is shorter. The thickness of the oxide layer prepared at 700℃ for 10h and 24h in pure O_2 atmosphere is 87 nm and 131 nm. The oxide layer has the microstructure of polycrystalline γ-Al_2O_3 mixed with amorphous Al_2O_3. The sample aluminized at 600°C and oxidized in pure O_2 atmosphere at 700°C for 24h has the best deuterium permeation resistance, and the deuterium permeation reduction factor (PRF) in the range of 450°C -600°C is higher than 5000. The PRF increases with the increase of temperature. The possible reason is that the deuterium permeability of 316LSS substrate increases greatly with the increase of temperature, while that of the coating samples is less affected by the temperature change. The excellent deuterium permeation resistance of the coating is due to the high density of the Al_2O_3 layer.

Tentative

P3-083

Irradiation stability and tritium retention characteristics of novel ceramic coatings for fusion breeder blanket applications.

Hazel Gardner[1], Emily Curtis[1], Gyula Zilahi[1], Tommaso Sgrilli[2], Victor Bellido-Gonzalez[2], James Wade-Zhu[1]

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Ceramic oxide coatings are a candidate tritium permeation barrier for use in the breeder blanket of future fusion powerplants. These coatings must reduce tritium transport and trapping in structural materials to avoid generating unnecessary quantities of high-level radiological waste, minimise hydrogen isotope embrittlement and maximise the amount of tritium available for fuel. The stability of the coatings under irradiation is critical to understanding their permeation performance in the breeder blanket.

In this work, a strategy was developed and implemented to develop thermal and irradiation-stable oxide coatings using magnetron sputtering. The stability of the coatings produced was tested using heavy ion radiation to introduce displacement damage. X-ray diffraction and transmission electron microscopy were used to check for irradiation-induced changes in crystal structure and microstructure. Coatings that pass these irradiation screening tests were then exposed to tritium to assess retention. This was measured using thermal desorption spectroscopy (TDS) and compared to local tritium distribution through spatial mapping. Candidate coatings have also been deposited onto pipes, constituting a first step towards scaling up permeation barriers from the laboratory to component length scale.

Tentative

P3-084

Examination of degradation in electrical insulating performance of Y_2O_3 coating layer under ion beam irradiation

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Yttrium oxide (Y_2O_3) is a candidate material of electrical insulating coatings for reduction of MHD pressure drop in liquid metal blanket systems. In the present study, ion beam irradiation on Y_2O_3 coatings was performed to examine degradation in the insulating performances due to irradiation damage.

The Y_2O_3 coatings of 1050 nm in thickness were fabricated by electron beam deposition on SS430 substrates. On the coating surfaces, 100 nm thick Pt electrodes were made for measurement of electrical conductivities of the coating layers. The coating specimens were irradiated with a 1.33 MeV He+ beam up to ~0.007 dpa. During the irradiation, changes in conductivities of the coating layers were examined at 23, 400 and 500 °C with the applied electric field of 0.19 and 0.58 V/μm.

In the irradiation on the coating with the electric field of 0.19 V/μm, no obvious increase in the conductivity was observed at 23 and 400 °C. However, a continuous increase was observed in the irradiation at 500 °C. In the irradiation with 0.58 V/μm at 500 °C, the increasing rate of the conductivity was significantly higher, and it increased by ~3 x 10^{-4} S/m corresponding to the damage of 0.0015 dpa. Although irradiation data for higher damages have to be obtained, the conductivity might exceed the requirement for MHD coatings (10^{-2} S/m) at ~0.1 dpa from the increasing rate. The results would provide the minimum required coating thickness in blanket systems to suppress the electric field strength induced in the coating layers.

Tentative

P3-085

Development of anti-oxidation layer on fusion reactor material for the usage of supercritical CO2

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Supercritical carbon dioxide (sCO_2) can be a coolant of blanket system, not only for the solid breeder system but for the liquid one, due to the low MHD effect and expected easiness in tritium contamination management. To build the sCO2 blanket, compatibility of the material is indispensable though the major fusion structure material, such as reduced activation steel and vanadium alloy, is expected to be oxidized in high temperature sCO_2. In this study, corrosion protecting layers (Ni or W) were electrochemically plated on a F82H plate, and the plated samples were exposed to sCO_2 at 550 deg-C up to 1000h. The usage of Ni is usually restricted due to the activation by neutron to manage the activated waste, while it will be able to be peeled and separated chemically from the bare material as long as it is used as a plating material. In the case of Ni plated (~30μm) sample, the oxidization of F82H was clearly mitigated. A mutual diffusion layer was found in the cross-sectional analysis of the sample prepared by an acceleration experiments in vacuum (~800 deg-C, vacuum). The W plated sample also exhibited good oxidation resistance, without forming an obvious mutual diffusion layer. Based on these results, material diffusion, and radioactivity were estimated and discussed for the practical long-term usage in a fusion reactor.

Tentative

P3-086

Development of a Phenomenological Model of Beryllide Corrosion in Water Vapor

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Invitation

Beryllides of metals are promising structural and functional materials for use in fusion reactors (ITER, DEMO), nuclear power plants, and in various high-temperature and aggressive environments. Their advantages include high thermal stability, low swelling, low neutron capture cross-section, and neutron multiplication capability, making them key candidates for use in breeder blankets and other structural elements.

However, when operated under high-temperature water vapor exposure, beryllides undergo corrosion damage, which can lead to: loss of mechanical properties and strength in long-term operating cycles; formation of unstable oxide and hydroxide phases (BeO, $Be(OH)_2$), which can accelerate material degradation; change in the composition and structure of the surface, which affects tritium generation and fuel accumulation in fusion devices.

At the moment, there is no universal mathematical model describing the behavior of beryllides in a water vapor environment taking into account the kinetics of corrosion, phase transformations and structural changes.

This paper presents a phenomenological model of corrosion developed on the basis of experimental data on the corrosion of beryllium and beryllides, which allows us to describe the interaction processes of water vapor with beryllides at a water vapor pressure of 10 to 10,000 Pa and in the temperature range of 600-1100 °C.

The model is based on varying the introduced effective parameters of the model: the effective thickness of the oxide layer film and the effective porosity of the oxide layer on the surface of beryllides, and in general allows us to describe the corrosion processes observed in experiments.

P3-087

Investigation of chromium beryllide corrosion processes during high-temperature interaction with a gas mixture

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Invited

The key requirements for materials used as neutron multipliers in a fusion reactor blanket are, large reaction cross section (n,2n), low parasitic absorption by reaction (n,γ), and availability and safety of use. To date, beryllium intermetallic compounds such as chromium beryllide ($Be_{12}Cr$), vanadium beryllide ($Be_{12}V$), titanium beryllide ($Be_{12}Ti$), and zirconium beryllide ($Be_{12}Zr$) best meet these requirements. Among them, titanium beryllide ($Be_{12}Ti$) and vanadium beryllide ($Be_{12}V$) are the most extensively studied. However, an equally promising, yet poorly studied material is chromium beryllide, which also has a high tritium breeding ratio and lower swelling compared to other intermetallides

This paper presents the results of a study of corrosion processes and their effect on the X-ray phase composition of chromium beryllide ($Be_{12}Cr$) samples. Corrosion studies were conducted using a synchronous thermal analyzer TGA-DSC 3+. The X-ray phase composition (XRD) was analysed using a D8 ADVANCE X-ray diffractometer. During the experiments, the thermal stability of the samples was evaluated, and the corrosion process was investigated under high-temperature (up to 1600°C) interaction with noble gases containing chemically active impurities (H_2O vapour, CO_2, O_2). The obtained experimental data enabled the determination of kinetic parameters for the interaction rate between chromium beryllide and the oxidiser. The results of X-ray phase analysis of $Be_{12}Cr$ samples before and after corrosion tests provided an opportunity to evaluate the effect of corrosion on their phase composition.

The research has been funded by the Science Committee of the Ministry of Science and Higher Education of the Republic of Kazakhstan with Program BR24993119.

P3-088

Effects of Cationic Impurities on the Corrosion of Ni-based Alloys in Molten Fluoride Salts

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(LiF)2-BeF2 (FLiBe) molten salt has been suggested as a candidate material for fusion reactors as a dual-use tritium breeding blanket and a coolant. However, various impurities present in molten salts can cause corrosion, which compromises safety and economics of fusion reactors. Impurity-driven corrosion is critical because metals, oxygen, and moisture are easily introduced both during manufacturing or transport of salt and as corrosion or activation products during reactor operation. Considerable study has therefore been dedicated to impurity control to achieve corrosion mitigation in salts. These include chemical and electrochemical purification processes.

Here we investigate how certain metal fluorides, added at specific concentrations, can be intentionally utilized to reduce impurity-driven corrosion. We highlight the importance of cathodic half reactions by low-concentration metallic impurities that may exist despite purification in predicting corrosion behaviors. The model alloy Ni-20Cr was subject to static corrosion in (LiF)0.465-(NaF)0.115-(KF)0.42, FLiNaK, molten salt as a surrogate for FLiBe at 650°C for 4 hours. Different impurity-alloy systems were explored to elucidate why some impurities reduce corrosion while others do not.

We discovered that impurities in a certain range of electrochemical potential, and at a certain concentration range, are necessary to reduce corrosion without changing its mechanism. The reduction in corrosion depth and density observed in SEM-EDS analyses is due to selective decoration of grain boundaries by electrochemically reduced cationic impurities. This signifies that both oxidation and reduction reactions attribute. This study offers an in-depth understanding of impurity-driven corrosion at reactor-realistic concentrations, providing a safe, homogeneous, and sustainable corrosion-control method.

Tentative

P3-089

The simulation study on the corrosion behavior of Fe-based alloys in liquid lead-lithium under nuclear-fusion devices

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As a promising candidate for various breeding-blanket designs in future fusion devices, liquid lead-lithium (Pb-Li) eutectic alloy would result in the degradation of iron (Fe)-based structural materials due to the dissolution corrosion. By performing first principles calculations, we investigated the dissolution corrosion of steels in liquid Li and Pb through energetics evaluation on the adsorption of Li and Pb atoms and the escape of Fe atoms on Fe surfaces (001), (110) and (111). Our energetics results indicate that both Li and Pb atoms energetically prefer to adsorb on the considered Fe surfaces, and further accelerate the escape of surface Fe atoms. The dissolution corrosion related to the adsorption and escape processes exhibits strong dependence on surface structures, the coverage of adsorbed Li or Pb atoms, alloying elements and the temperature of working environment. Meanwhile, the effects of corroded products on the liquid Pb-Li are explored by the micro-structure of the liquid Pb-Li and as well as its diffusivity. Both metallic (Fe, Cr) and non-metallic (C, N) elements change the structure of liquid Pb-Li due to the different electronegativity between atoms, and non-metallic elements show strong effects on the diffusivity of Li. This study reveals the mechanism of the dissolution corrosion of Fe-based alloys in contact with liquid Pb-Li and the effects of the corroded products on the properties of liquid Pb-Li.

Tentative

P3-090

Study of corrosion processes of chromium beryllides and titanium beryllide with water vapor at critical vapor parameters

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Tentative

As is known, titanium and chromium beryllides are considered as one of the promising materials for fusion, and in some cases, fission reactor facilities.

The use of beryllides ($TiBe_{12}$, $CrBe_{12}$) in breeder blankets of fusion reactor is due to the high neutron multiplication factor, improved mechanical and radiation properties and effective thermal conductivity. In general, beryllides allow to increase the efficiency of tritium generation, reduce mechanical and radiation damage, and also improve the reliability of the breeder module design.

One of the tasks defining the boundaries of possible application of beryllides is the parameters of its corrosion resistance under operating conditions with high-temperature exposure to water vapor contained in the inert purge gas of the blanket cooling system. A sufficiently large number of studies have been conducted that have allowed us to determine beryllide corrosion parameters at various sample temperatures and vapor pressures, but there are a number of questions in determining the mechanisms of such processes. Obtaining new experimental data on corrosion will allow us to formulate such mechanisms.

This work presents data from corrosion experiments of titanium and chromium beryllides at previously unexplored parameters of sample temperature and water vapor pressure. Dependencies of changes in the microstructure and mass of samples after exposure to water vapor in the temperature range of 300-500 °C, at vapor pressures from 10 to 20 MPa have been obtained.

P3-091

Corrosion behavior of CLF-1 steel with different surface roughness in a magnetic field environment

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The corrosion behavior of the cooling pipe in the fusion blanket is important to evaluate the production rate of activated corrosion products and the service life of the cooling pipe. The cooling pipe of the water-cooled blanket in fusion reactor can be corroded by cooling water in a magnetic field. Previous studies have shown that magnetic fields and surface roughness can affect the corrosion process of steel. To accurately evaluate the corrosion behavior of the cooling pipe in the blanket, the corrosion behavior of the CLF-1 steel, a main candidate cooling pipe material for the Chinese water-cooled blanket, with different surface roughness in the magnetic field has been studied. In this study, mechanical and electrochemical polishing were used to obtain different surface roughness. Corrosion experiments have been performed on CLF-1 steel with different surface roughness in solutions of 1 wt% H_2O_2 + 0.1 wt% NaCl solution with 0 T, 0.1 T, 0.5 T and 1 T magnetic fields. To evaluate the corrosion behavior of the CLF-1 steel with different surface roughness in a magnetic field, the corroded surface of the CLF-1 steel has been observed using scanning electron microscopy. The corrosion products on the sample surface and in the solution were observed using scanning electron microscopy combined with energy dispersive spectroscopy. The chemical composition of the corrosion products was evaluated using X-ray spectroscopy. The influence mechanism of surface roughness of CLF-1 steel in magnetic field environment has been discussed.

Tentative

P3-092

Microstructural effects on He bubble distribution from tritium decay.

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Tritium, as an isotope of hydrogen, permeates throughout most known materials and decays over time into ^{3}He causing both hydrogen and helium embrittlement which have significantly different solubilities. Tritium embrittlement is fundamentally different to He exposure under irradiation since it is purely the transmutation of tritium into ^{3}He without an accompanying atomic displacement (dpa=0). Typically, austenitic stainless steel is used for tritium handling due to its high fracture resistance and favorable hydrogen embrittlement resistance. In this study, 304L stainless steel tubing was subjected to 6 different heat treatments to achieve different microstructures, then pre-charged with tritium gas to 3700 appm. Samples were then stored at -80°C to allow ^{3}He to accumulate while minimizing the loss of tritium to off-gassing. When helium content was estimated to be 650 appm, samples were tensile tested and imaged with transmission electron microscopy (TEM). Microstructural analysis was conducted by electron backscatter diffraction (EBSD) of heat-treated samples that were not tritium charged with grain sizes increasing slowly from the as received (11.1μm) to recrystallized (36.6μm) and a significant increase at the fully annealed condition (1201.8μm). He bubble densities and sizes remain consistent from the as received condition up to the recrystallized condition, at the fully annealed condition there is a significant increase in number density and reduction in size of He bubbles possibly related to the grain growth of the fully annealed condition.

Tentative

P3-093

UNDERSTANDING THE ROLE OF HELIUM-INDUCED CAVITIES ON TRITIUM TRAPPING IN FUSION MATERIALS

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Due to tritium scarcity, future fusion energy systems must be fuel self-sufficient. Additionally, proposed first wall structures and plasma facing components will experience unprecedentedly high neutron damage (~80 dpa end-of-life for blanket structures in DEMO). Whilst inducing defect formation, 14 MeV fusion neutrons also trans-mutate, producing significant amounts of helium (He) within the material, (~10 atomic parts per million (appm) He/dpa). The combination of irradiation damage and He results in enhanced nucleation of He-vacancy (V) clusters and He-filled cavities. Research shows hydrogen (H) isotopes stabilise cavities, strongly binding with He-V clusters. This means He-V clusters and He-stabilized cavities subsequently entrap tritium, risking self-sufficiency and presenting safety concerns. The extent of these potential effects in the fusion materials community, is poorly elaborated. Here, we present experimental results using ion-implantation and deuterium (D) charging to understand the effect of He-induced cavity formation on tritium entrapment, using D as a surrogate, for Fe-Cr binary alloys, RAFM steels, Vanadium alloys, and Tungsten.

Samples were ion-implanted at JANNUS-Orsay irradiation facility with He implantation levels ranging from 2500 to >10000 appm, over a wide temperature range, 350 to >600 °C. Following irradiations, the samples were exposed to D for varying temperature and time intervals. State-of-the-art transmission electron microscopy (TEM) and thermal desorption spectroscopy (TDS) techniques were utilized to reveal the depth distribution of He and D in the materials. The results will present the correlation of D trapping and release with the number density, and size distribution of He-filled cavities in the microstructure.

Tentative

P3-094

Identifying the He and D species trapped in selected fusion materials

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A comprehensive understanding of the processes involved in the diffusion and trapping of light ions in fusion materials is of paramount importance for the development of nuclear facilities. In particular, one commonly observed effect is that the interaction with, production and diffusion of H isotopes and He in the surrounding materials can lead to their structural and/or functional degradation. For instance, this drawback can occur in components of first wall armour by plasma radiation and the breeding blanket through the neutron transmutation arising of liquid metal contact. Consequently, an identification and characterization of light elements is required. However, the detection and qualitative analysis of deuterium and helium at comparable distributions in depth of materials is a challenging issue given their close atomic masses. Therefore, the objective here is to explore the differentiation of coincident helium and deuterium in reactor materials at various depths. To this end, implantations of He and D ions at different fluences have been conducted in tungsten, Cu alloy and ceramic oxide matrices. These irradiations were aimed at jointly implanting both ions at a certain depth, as well as individually for comparison. The selected energies provide ions distributed from ~100nm to several microns. The determination are being carried out by different techniques, including Secondary Ion Mass Spectrometry (SIMS) and the Quadrupole Residual Gas Analyzer with high resolution mass detection. Previous SIMS studies on deuterium irradiation in these materials point out that the light ion distribution definition underneath surface is provided with significant signal.

Tentative

P3-095

Study of tritium and helium generation and release processes from Ga-Li alloy under high-density neutron radiation

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The use of lithium-containing alloys in fusion reactors is one of the promising areas of development of modern energy. Lithium-containing alloys can operate at high temperatures, have low viscosity and have low saturated vapor pressure, which makes them ideal for use in nuclear and thermonuclear energy systems. One of such materials is a gallium-lithium alloy. No data on such studies were found in the public domain. Thus, there is a need to conduct reactor studies to clarify the mechanisms and develop models that can describe the processes of generation and release of tritium and helium from a liquid alloy under neutron irradiation conditions. The paper presents the results of a study of the processes of generation and release of tritium and helium from a gallium-lithium alloy under reactor irradiation conditions at temperatures of the studied sample in the temperature range of 200 °C - 700 °C. In the experiments at the IVG.1M reactor (Kurchatov), an irradiation ampoule device was used, into which 30 grams of an alloy of 64 at. % Ga - 36 at. % Li was loaded. Time dependences of tritium and helium flows from the sample were obtained at an integral neutron flux density from 2.5・10^13 to 1.5・10^14 n/(cm^2・s). The main parameters of generation and release of tritium and helium from gallium-lithium alloy were determined, such as: generation rate, pre-exponential factors of the tritium and helium generation rate coefficients in the Arrhenius dependence, solubility, and activation energies of processes.

Tentative

P3-096

Effects of helium production and radiation in advanced permeation barriers

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Tentative

The development of advanced coatings is crucial for fusion reactors, particularly in reducing tritium permeation and preventing its release into the atmosphere. These coatings must also protect against severe corrosion and mitigate the magneto-hydrodynamic pressure drop in liquid metal breeding blanket concepts. Key requirements include a high permeation reduction factor (PRF), low electrical conductivity, and compatibility with PbLi under high radiation and temperature conditions.

Validating these coatings involves assessing their properties and evaluating the synergetic effects of the PbLi environment and radiation at relevant temperatures. Previous studies have shown that lithium diffuses into alumina coatings after exposure to PbLi, leading to helium production through nuclear transmutation. This process can cause bubble formation that alters tritium transport and coating stability. The current research focuses on the interactions between hydrogen isotopes and helium, particularly He-induced tritium retention and its effects on coating integrity. Additionally, the electrical conductivity of these coatings will be evaluated before and after irradiation to determine their suitability for fusion applications.

Various advanced coatings, including newly developed SiC-based compositions, will be studied. Helium and proton irradiation at fusion-relevant conditions will be carried out using the Ciemat 60 kV Danfysik ion implanter, enabling the evaluation of ionizing and displacement effects, as well as helium production. Techniques like Transmission Electron Microscopy (TEM), Scanning Electron Microscopy (SEM), and Thermal Desorption Spectroscopy (TDS) will be used to assess the impact of helium on these coatings.

P3-097

Stress corrosion cracking on Eurofer subjected to Li ceramics

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Stress corrosion cracking (SCC) is a critical failure mechanism that occurs due to the combined effects of tensile stress and a corrosive environment. To study SCC, U-bend specimens are commonly used because they can simulate the conditions that lead to this type of failure.

An U-bend specimen is typically a strip of metal that is bent into a U-shape and then subjected to a corrosive environment while under tensile stress. This setup allows to observe the initiation and propagation of cracks under controlled conditions. The U-bend method is particularly useful for evaluating the susceptibility of different materials to SCC and for testing the effectiveness of various protective measures. By using U-bend specimens, is possible to gain valuable insights into the mechanisms of SCC and develop strategies to mitigate its impact on critical structures and components.

In the present work, Eurofer U-cylinders were manufactured and subjected to the corrosive action of pebbles containing lithium. the tests lasted for 2880 and 4300 hours under conditions relevant to the operation of the HCPB concept; i.e. H2/0.1%He gas flow to simulate purge gas and 550oC.

The stresses generated in the specimens were simulated by the finite element model and the differences between the corrosion layers produced in the different stress zones and the identified microcracks were studied by SEM-EDX. EBSD studies were also carried out to relate stress to possible microstructural changes or segregations.

Finally, the composition of the corrosion layer and the diffusion of lithium was studied by SIMS.

Template

P3-099

FC FIRE: Advancing Fuel Cycle Solutions for Fusion Energy's Future

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The Fuel Cycle (FC) Fusion Innovation Research Engine (FIRE) Collaborative addresses critical tritium management challenges for commercial fusion energy systems. Given tritium's radiological hazards and stringent confinement requirements, the FC FIRE Collaborative, led by Savannah River National Laboratory and involving national laboratories, universities, and industry partners, aims to develop integrated solutions for fuel cycle design, inventory reduction, emissions mitigation, and workforce development. The initiative focuses on five strategic thrusts:

Integrated Modeling Frameworks: Development of open-architecture process models to optimize tritium flow, inventory tracking, and system integration across magnetic and inertial fusion concepts. Process Innovation: Testing direct internal recycling systems, advanced impurity removal technologies, and isotope separation methods to reduce tritium inventory.Tritium Materials: Characterization of tritium interactions with materials and development of hydrogen isotope permeation barriers and tritium-resistant polymers to mitigate permeation and retention. Environmental Stewardship: Strategies for detritiation of byproducts and waste, including recycling pathways and repository planning to minimize environmental releases.Workforce Development: Creation of educational modules, internships, and public engagement programs to train a diverse workforce and build community trust in fusion safety.

Key innovations include pilot-scale DIR systems, high-throughput catalyst discovery using machine learning, and integration of techno-economic analysis with life-cycle assessment. Collaborative efforts with international partners align with global fusion roadmaps. The FC FIRE Collaborative aims to deliver scalable, economically viable fuel cycle solutions to accelerate fusion energy commercialization while addressing regulatory and societal acceptance challenges, supporting the U.S. Bold Decadal Vision for fusion energy.

Tentative

P3-100

Technology of synthesis of Ga_x-Li_x alloys with different lithium content and characterization of the obtained alloys by SEM, XRD and DSC methods

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Gallium-lithium alloys (Ga_x-Li_x) attract attention due to their low melting point, high thermal conductivity, self-healing ability and interaction with hydrogen isotopes and neutrons. Varying the atomic content of lithium in such alloys allows optimizing their properties for specific applications, such as liquid metal divertors, tritium breeding systems and heat exchangers. The objective of the presented work was to develop a technology for the synthesis of gallium-lithium alloys with different lithium atomic contents (from 10% to 50%) and a comprehensive characterization of their structure, phase composition and thermal properties using modern analytical methods. Test samples were obtained by melting lithium and gallium in an inert atmosphere and homogenizing the alloy with ultrasound. Then, the synthesized alloys were characterized by SEM (scanning electron microscopy), XRD (X-ray diffractometry) and DSC (differential-scanning calorimetry), which made it possible to establish the dependence of their structure and properties on the composition. The XRD results showed that Ga?-Li? alloys form solid solutions with a predominance of the gallium phase at a low lithium content, and with an increase in the lithium content, additional intermetallic phases are observed. SEM images demonstrate a homogeneous distribution of lithium in the gallium matrix, indicating high stability of the alloys. The formation of a finely dispersed structure is observed, which increases the surface area and improves the functional properties. The melting point of gallium-lithium phases and their enthalpies were determined using the DSC method. The results demonstrate the prospects of gallium-lithium alloys for use in fusion reactors.

Tentative

P3-101

Manufacturing Optimisation for Joining of CuCrZr and Eurofer-97 for the DEMO Limiter Component

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Joining of fusion-relevant materials, such as CuCrZr and reduced activation ferritic-martensitic steels, without detriment to the mechanical properties of the base material is a critical challenge faced within the fusion community. In each joining cycle a material is subjected to, the base material is evolving. Integrating the base material heat treatment cycles with appropriate joining techniques provides a promising method to address this challenge.

This research focuses on the development of a manufacturing workflow for the EUROfusion DEMO Limiter component, in which a sacrificial tungsten lattice is joined to a CuCrZr heat sink and Eurofer-97 pipe. Within this, the integration of the joining temperatures and the optimisation of base material heat treatments for Eurofer-97 and CuCrZr is evaluated. Brazing, a joining technique in which a filler metal melts and flows between two parent materials, is utilised due to its comparative freedom in joining temperature range, technological maturity, and scalability. Both similar and dissimilar material brazing using low-activation filler metals, CuGeNi and CuGe, are investigated. Manufacturing processes have been compared by microstructural characterisation and mechanical testing of the CuCrZr-CuCrZr and CuCrZr-Eurofer-97 joints, as well as characterisation of the base material with comparison to material property handbooks. This research delves into the manufacturing potential of strong, scalable joints without compromising the base material properties.

Tentative

P3-102

OpenMC-DONES source: Development and validation of IFMIF-DONES neutron source for OpenMC from McDeLicious-MCNP6 d-Li interaction results

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IFMIF-DONES will be a first-of-a-kind high-energy, high-flux neutron source dedicated to materials testing for fusion. Global interest in testing materials at DONES is growing, which necessitates accurate simulation of irradiation conditions and the design of tailored irradiation modules.

The DONES neutron source is based on neutron stripping reactions from accelerated deuterons (d) impacting a liquid lithium (Li) target. These reactions are challenging to model with existing neutronics codes. Currently, PHITS with JENDL deuteron data and MCNP with the McDeLicious extension can handle such calculations. To reduce costs, researchers often use precomputed neutron source definitions derived from detailed simulations.

OpenMC is an open-source Monte Carlo neutronics code that has recently gained traction among fusion scientists. Its open MIT license and active global development community have significantly expanded OpenMC's capabilities for fusion. However, OpenMC cannot directly model d-Li interactions without the McDeLicious extension and requires an externally defined neutron source.

This work presents the development and validation of a DONES neutron source package for OpenMC. IFMIF-DONES exhibits highly anisotropic neutron emission, with higher-energy neutrons emitted primarily in the forward direction. The neutron source data was obtained from MCNP-McDeLicious d-Li simulations performed by KIT. To account for this anisotropy, the energy distribution was defined for finite azimuthal angles of 2.5?5 degrees, while the poloidal distribution was kept isotropic. The spatial distribution was kept uniform along the beam footprint. The package was validated using neutron response estimations for a homogeneous HFTM model.

Tentative

P3-103

Impact of Bolt Penetration Size on Radiation Streaming and Dose Rates in the IFMIF-DONES Test Cell Floor

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The International Fusion Materials Irradiation Facility - DEMO Oriented NEutron Source (IFMIF-DONES) aims to provide critical insights into material performance under extreme neutron irradiation conditions. The recent design of the Test Cell (TC) floor includes mounting bolts to ensure structural stability and efficient thermal management. However, bolt penetrations and their associated gaps may introduce pathways for radiation streaming, posing challenges for shielding optimization and safety assurance. This study evaluates the impact of bolt penetration size variations on radiation streaming and dose rates within the IFMIF-DONES Test Cell floor. Using detailed Monte Carlo simulations with MCNP and the McDeLicious extension, we analyze neutron and gamma flux distributions for different penetration sizes. Advanced variance reduction techniques with ADVANTG are applied to improve statistical accuracy in critical regions. Preliminary results indicate that increasing bolt penetration size enhances neutron and gamma streaming, leading to elevated dose rates in localized regions. Conversely, reducing penetration size mitigates streaming but introduces engineering constraints. The findings highlight a critical balance between design feasibility and radiological safety. Dose rate maps and flux distributions are presented to guide design modifications and shielding enhancements. This work provides valuable guidance for optimizing engineering features in high-flux neutron facilities, ensuring compliance with safety standards while maintaining structural and operational efficiency. The methodologies and results are integral to advancing the IFMIF-DONES project and informing future fusion reactor designs.

Tentative

P3-104

Thermal-Hydraulic Analysis on New DONES -HFTM Design

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IFMIF-DONES (International Fusion Materials Irradiation Facility- DEMO-Oriented Neutron Source is an IFMIF-based neutron irradiation facility which aims at providing the irradiation data required for the construction of a DEMOnstration fusion power plant (DEMO). DONES consists of only one of the IFMIF accelerators (40 MeV and 125 mA), and utilizes the High Flux Test Module (HFTM) for the irradiation of material specimens. The HFTM is the key component to provide the material irradiation data. The irradiation is planned at several blanket-relevant irradiation temperatures and shall accumulate structural damage of up to 50 displacements per atom (dpa).

Helium flows at low pressure are used to cool the specimen capsules and the structure of HFTM. The flow path includes inlet and outlet ducts with large cross sections, but also mini-channels with gap widths less than and equal to 1 mm, where a high velocity low Reynolds number flow removes the heat introduced into specimens and HFTM structure by nuclear heating. Additionally, three-segmented heaters allow to set close to homogeneous temperatures for all specimens of a capsule.

To effectively use the available neutron flux for sample irradiation inside the tested materials, the design of the HFTM was optimized and significantly improved. To support the new design, detailed numerical thermo-hydraulic analysis of the conjugated heat transfer in the HFTM was performed. Various load cases consisting of pressure and temperature profiles were defined to demonstrate that the HFTM design iterations achieve conformance with the RCC-MRx under all possible scenarios.

Invitation

P3-105

Optimization of the electrical harnesses for the in-vessel systems at the IFMIF-DONES Test Cell

Santiago Becerril[1], Pablo Araya[1], Frederik Arbeiter[2], Jesus Castellanos[3], Tamas Dezsi[4], Jorge Maestre[1], Maria Megia[5], Francesco Saverio Nitti[6], Denes Oravecz[7], Aurora Ortega[1], Ivan Podadera[1], Juanjo Rueda[1], Urszula Wiacek[8], Andras Zsakai[7], David Jimenez[9], Philippe Cara[10], Angel Ibarra[11]

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Tentative

The Test Cell will be the cavity at IFMIF-DONES where unprecedented, continuous, high-flux fast-neutron radiation fields, combined with intense gamma fields, will be generated. These fields result from the stripping reaction between the deuteron beams (125 mA, 5 MW, 40 MeV each) and the liquid lithium target. The extreme radiation conditions create a highly demanding environment for all in-vessel systems and components housed within the Test Cell.

Each in-vessel component requires a set of diagnostics and actuators to ensure proper functionality. These elements are controlled from ancillary systems located outside the Test Cell and are routed inside through customized shielding plugs-known as PCPs (Piping and Cabling Plugs). These PCPs incorporate ducts, pipes, and connection panels linked to in-vessel electrical connection bridges.

A deeper analysis of the current baseline design has shown good margin for improvement in the network of electrical harnesses, thus the risk of failure for in-vessel systems being reduced. From a RAMI (Reliability, Availability, Maintainability, and Inspectability) perspective, this section of the facility needs to be strengthened.

This work presents an updated baseline design of the electrical infrastructure and harnesses associated with in-vessel components at the Test Cell. It also includes a critical assessment aimed at minimizing diagnostics and electrical actuators to enhance system reliability.

P3-106

Correlation study of material damage and activation inventory for EUROFER, tungsten and CuCrZr alloy in the IFMIF-DONES irradiation environment

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IFMIF-DONES is an accelerator-based neutron irradiation facility. The facility comprises a deuteron accelerator impinging on a lithium jet as neutron producing target and the High Flux Test Module (HFTM), which is utilized to irradiate material specimens. These specimens are composed of reduced activation ferritic-martensitic steels such as EUROFER, as well as tungsten and CrCrZr alloy which are considered to be used as plasma-facing material and heat sink material, respectively.

Previous work on EUROFER material damage and activation inventory in the IFMIF-DONES irradiation environment has indicated that the damage dose in terms of dpa-NRT has strong correlations with the gamma activity of EUROFER. As Mn-54 is the dominant isotope which emits more than 99% of gammas during the timespan of half-year to several years, the gamma spectrometry of EUROFER is concluded to be a useful way to determine the damage dose of each sample through semi-linear correlations.

In this work, further study on this correlation is reported, extending it with experimental validation using an accelerator-based neutron source, such as cyclotron neutron source at NPI-CAS (Czech Republic) and NFS at Ganil (France), with comparable neutron energy as DONES. EUROFER material samples are irradiated together with the activation foil sets at different positions, and the correlation dependance can be estimated. In addition, this study is extended to tungsten and CuCrZr alloy, which are more challenging as the dominant isotopes and gamma lines are much different. The conclusion on the correlations for these two materials and the recommended approaches will be drawn.

Tentative

P3-107

A blanket functional materials irradiation module in IFMIF-DONES for Helium Cooled Pebble Bed breeding blanket

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Within the EUROfusion DEMO programme, the Helium Cooled Pebble Bed Breeding Blanket (HCPB BB) is one of the candidate driver blankets for European DEMO. The HCPB BB utilizes pressurized helium as a coolant, lithium-containing ceramic pebbles as the tritium breeder, titanium beryllide blocks as the neutron multiplier, and Eurofer97 as the structural material. The current HCPB BB design is based on a fuel-breeder pin concept, where the lithium ceramic pebble bed is enclosed within a Eurofer-steel pin, with the titanium beryllide block positioned outside. The tritium breeder and neutron multiplier materials are collectively referred to as the functional materials of the HCPB BB. Extensive design and experimental studies have been conducted on these materials, focusing on single-effect phenomena such as thermal cycling behavior, tritium release properties, and crush characteristics. To investigate multiple-effect phenomena under DEMO-relevant conditions, a fusion-like neutron source with sufficient irradiation volume, a relevant neutron spectrum, and an appropriate flux is required. IFMIF-DONES (International Fusion Materials Irradiation Facility - DEMO Oriented NEutron Source) is primarily designed to test small samples. However, behind its High Flux Test Module, the neutron flux and available volume remain suitable for studying multi-effect phenomena in the breeder zone of the HCPB BB using a mockup that closely replicates DEMO conditions. For this purpose, a dedicated mockup called BLUME (BLanket fUnctional Materials irradiation modulE) has been developed. This work presents the system requirements, test objectives, preliminary design, and analyses of BLUME.

Tentative

P3-108

Neutronics analysis of the structural and functional materials within the IFMIF-DONES Test Modules

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The presented neutronics analysis serves to support the materials qualification of the EU DEMO Breeding Blanket (BB) by designing an irradiation experiment in IFMIF-DONES (International Fusion Materials Irradiation Facility - DEMO Oriented NEutron Source). In IFMIF-DONES, the beam of deuteron (d) ions is accelerated to 40 MeV with a current of 125 mA to impinge on the target of liquid lithium (Li) jet, producing on nuclear stripping reaction Li(d,xn) neutron flux of 5e14 n/cm^2/s peaking at the energy of 14-15 MeV. In addition to neutrons, prompt and secondary high-energy photons are generated by Li(d,xγ) and (n,xγ) nuclear reactions. These neutrons and photons irradiate specimens of the structural materials (EUROFER, tungsten, and CrCrZr) installed inside the High Flux Test Module (HFTM) rigs. Passing through HFTM, the neutron flux is attenuated to 8e13 n/cm^2/s, entering the Medium Flux Test Module (MFTM), where functional materials (Advanced Ceramic Breeder and titanium beryllide as neutron multiplier) of the EU DEMO Helium-Cooled Pebble Bed (HCPB) BB are supposed to be irradiated. To reproduce at MFTM in DONES the physical phenomena of the HCPB BB neutron irradiation in DEMO, a dedicated BB test module mockup called BLUME (BLanket fUnctional Materials irradiation modulE) has been developed. The neutronics computations have been performed with the McDeLicious code. Neutron and photon fluxes, neutron damage, helium production, and nuclear heating have been analyzed for the structural and functional materials. In conclusion, the results demonstrated the representativeness of the IFMIF-DONES neutron source to the EU DEMO fusion reactor conditions within the BB.

Tentative

P3-109

Neutronics analysis for irradiation candidate materials with fusion neutron source

Kentaro Ochiai[1], Saerom Kwon[1], Shogo Honda[1], Dai Hamaguchi[1], Takashi Nozawa[1], Kai Masuda[1]

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Plans are in progress to construct a materials irradiation facility using high-energy neutrons to perform engineering validations on candidate materials for fusion reactors. The high-energy neutrons are to be generated by accelerator-based deuteron beams and liquid lithium targets, which are planned as A-FNS in Japan and DONES in EU.

Candidate materials for irradiation using these facilities include reduced activation ferritic/martensitic (RAFM) steel as a structural material, tungsten as a plasma facing material, which is also being considered in ITER blanket and divertor, and copper alloys as cooling tubes for the divertor components.

The d-Li neutron source with 40 MeV deuterons produces more than 14 MeV neutrons in contrast to DT neutrons. In the case of RAFM steel, the ratio of helium production to displacement damage rate, i.e., appm-He/dpa, is known to be close to that of DT, so simulated irradiation is considered possible, but tungsten and copper alloy have not been fully investigated. In addition, tungsten is said to have an effect on material properties due to transmutation products, e.g., Re and Os, from nuclear reactions.

In the present study, the radiation transport code (MCNP) and activation calculation code (FISPACT) for the above candidate materials are used based on the well-validated d-Li neutron spectra to systematically analyze the dpa, hydrogen and helium production, nuclear reaction products, etc., and to present a comparative study of the differences from the case of DT neutrons.

Tentative

P3-110

Low level dose characterization of 12 MeV proton irradiation on Inconel 718 alloy

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The higher energy neutronics environment of proposed fusion deuterium and tritium reactors require extensive testing to understand how novel materials may react during their lifetime exposures to radiation within these reactors. Current materials testing techniques, such as high neutron flux exposure and heavy ion irradiation, produce either long lived radioisotopes or are unable to create bulk damage for macroscopic post irradiation examination. The goal of this research was to test the viability of 12 MeV proton irradiation as a method for fast, high fidelity, and tunable radiation damage testing at both the macroscopic and microscopic level for the eventual use of testing novel fusion materials. A beam of 12 MeV protons from a medical isotope cyclotron was used to irradiate Inconel 718 alloy to different doses in the expected linear buildup regime of damage with samples having doses between 0.2-3 mdpa and comparing yield strength changes and expected defect densities between similar dose neutron irradiations. Tensile tests were performed on 250 μm thick samples and defect density and size estimations were measured through the use of transmission electron microscopy, atom probe tomography, and strengthening and damage models. Similarities in yield strength changes were observed between both irradiation methods suggesting proton irradiation possesses a similar linear buildup regime as expected with other irradiation techniques.

Tentative

P3-111

Defect formation behavior of ARAA steel under He ion irradiation

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In the complex radiation environment of fusion reactors, the first wall/blanket structural materials would be degraded by high-energy neutrons. The high-energy neutron irradiation generates a large number of defects in the materials, such as dislocation loops and voids, then causing the embrittlement of the materials. Thus the fusion blanket materials must have excellent resistance to high-energy neutron irradiation and the reduced activation ferritic/martensitic (RAFM) steel have been considered as a preferred candidate which can be applied in the harsh fusion environment. For this purpose, a kind of RAFM steel, namely Advanced Reduced Activation Alloy (ARAA), was developed in South Korea and its physical and mechanical properties have been extensively studied for the application to the International Thermonuclear Experimental Reactor (ITER). As a part of the research regarding the irradiation resistance of ARAA steel, in this study, the defect formation behavior was investigated under helium (He) ion irradiation, which is an effective way to probe the relation between defect formation and performance variation of the steel. 688 keV He ions were used to irradiate ARAA steel in order to simulate neutron irradiation in fusion environment. Using transmission electron microscopy, the effects of irradiation fluence and temperature on defect formation were comprehensively investigated and the depth distribution of the irradiation-induced defects was also quantified. The observed experimental results showed the mutual consistency with the related simulation results. *This work was supported by the National Research Foundation of Korea (NRF) funded by the Korea government. (Ministry of Science and ICT) (RS-2022-00156205).

Tentative

P3-112

Compatibility of pre-oxidized ODS FeCrAl under combined neutron irradiation and high temperature liquid metal effects

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Oxide Dispersion Strengthened (ODS) FeCrAl steels are being considered as blanket structural materials based on promising static compatibility studies in high temperature eutectic PbLi and Sn, neutron irradiation resistance and ability to form an alumina scale. Alumina scales would act as electrical insulators and tritium permeation barriers and have been shown to decrease dissolution rates in eutectic PbLi and Sn below 600 °C. However, the combined effect of neutron irradiation and high temperature liquid metal exposure on the stability and microstructure of the alumina layer has not been studied in detail and could be a limitation of these protection barriers at reactor-relevant operating conditions. Therefore, in the present work, a first of a kind, liquid tin rabbit capsules containing various pre-oxidized FeCrAl alloys were irradiated in the High Flux Isotope Reactor at a target temperature of 400 °C for 10 days to start addressing potential limitations. Post-irradiation examination of the exposed specimens was conducted. The impact of neutron irradiation on room temperature tensile properties and on the stability and microstructure of the alumina layer will be discussed with the results of multi-scale microstructural characterization techniques.

Tentative

P3-113

Effect of Neutron Irradiation on the Mechanical Properties of Additively Manufactured 316L Stainless Steel and the Influence of Sampling Location

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Austenitic stainless steels are considered promising candidate materials for vacuum vessels, support structures, and cryogenic systems in fusion reactors due to their favorable mechanical properties, high corrosion resistance, and ability to withstand high temperatures and radiation exposure, as well as the extensive knowledge base derived from fission reactor experience. Modern additive manufacturing (AM) technologies, particularly laser powder bed fusion (LPBF) methods, are highly applicable to these materials. The rapid melting-solidification process and multiple heating-cooling cycles during LPBF result in a fine, metastable microstructure with significant local variability. This study investigates the dependence of the mechanical properties of AM 316L stainless steel on irradiation dose and temperature and explores the influence of sampling location on property variability. SS-J miniature tensile specimens were machined from six different locations across three AM 316L plates and neutron-irradiated to doses up to 10 dpa at 300°C and 600°C. Post-irradiation tensile tests were conducted at room temperature, 300°C, and 600°C. Compared to traditional 316L steel, AM 316L exhibited significantly higher initial strength but lower ductility. Significant irradiation hardening and prompt necking at yield, with limited uniform ductility, were observed after irradiation at 300°C, although no embrittlement was noted up to 10 dpa. At 600°C, irradiation induced noticeable ductilization, though this effect was observed only at low irradiation doses. Although neutron irradiation, particularly at 600°C, increased variability in strength and ductility, no consistent dependence on build thickness or sampling location was found, which is attributed to the unique, uniform residual stress field in the fine metastable microstructure.

Tentative

P3-114

Neutron irradiation micro-structural effects in the low activation ferritic/martensitic steel EUROFER97

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In this contribution a series of experimental results on neutron irradiation effects in EUROFER97 will be presented. Small-angle neutron scattering (SANS) has been utilized, since this tecnique provides non-destructively micro-structural information averaged over large sample volumes, therefore complementary to TEM and ATP; it also allows to investigate separately nuclear and magnetic defects. Furthermore, the problems relating to preparation and manipulation of irradiated samples for SANS measurements are considerably reduced when comparing with other micro-structural techniques. The SANS measurements have been carried out at different neutron sources (ILL-Grenoble, FRM II-Garching, SINQ-Villigen), always utilizing an external magnetic field of at least 1 T and un-irradiated EUROFER97 reference samples. EUROFER97 samples irradiated at BOR60 and at HFR Petten for similar dose and temperature have been investigated, finding considerably smaller SANS cross-sections in the BOR60 ones; this is tentativey attributed to the fact that while micro-voids dominate after mixed spectrum irradiation (HFR) the fast spectrum of BOR60 produces mostly dislocations, nearly invisible by SANS. Furthermore, B-alloyed EUROFER97 HFR neutron irradiated to 16 dpa at various temperatures have been investigated utilizing polarized SANS, finding a huge production of helium bubbles and a strong decrease of the magnetic SANS component. These results will be discussed with reference to TEM and ATP characterization of these same materials, in order to contribute to better understand the effects of the considered irradiation conditions.

Tentative

P3-115

Accelerated evaluation of swelling in F82H as a function of implanted H/He through dual gas-gradient triple-ion irradiation experiments

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Ion irradiation can emulate neutron irradiation through simultaneously injecting gas atoms with self-ion

radiation damage. Key to this is the ratio between gas implantation (appm) and irradiation dose (dpa). Rapid exploration of this parameter space (appm/dpa) is limited by the serial nature of irradiation experiments, which typically investigate one fixed H/He ratio. Recently, the Michigan Ion Beam Laboratory has developed dynamically controlled shutters to create lateral H and He implantation gradients, and thus a range of H/He ratios, within a single sample. Hundreds to thousands of H/He combinations can be made, limited only by the resolution of the analysis technique. This capability was used to conduct a dual-shuttered triple-ion irradiation of F82H-IEA ferritic-martensitic steel to 50 dpa at 500°C, with implanted gas ratios of 0-80 appm/dpa H and 0-10 appm/dpa He. Transmission electron microscopy was used to characterize the cavity size distributions and volumetric swelling as a function of implanted H/He. The results will be compared to existing experiments, from fixed H/He irradiations, that have established the relationship between H/He implantation and cavity evolution. A second irradiation was conducted on F82H-Mod3, to a dose of 9 dpa at 550°C, with the same implanted gas ratios. The results will be compared to a prior neutron irradiation that utilized in situ He implantation. These experiments demonstrate the significant increase in sample throughput that can be achieved, and can be used to determine the precise H/He implantation ratio required to accurately emulate the cavity swelling of neutron irradiations.

Tentative

P3-116

The effects of low temperature neutron irradiation on the mechanical property and microstructure evolution of Amosic-3 SiCf/SiC composites

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The effects of low temperature neutron irradiation on the mechanical property and microstructure evolution of Amosic-3 SiCf/SiC composites was studied after 0.5 displacements per atom (dpa) at approximately 40 °C. The neutron irradiation was performed at China Mianyang Research Reactor (CMRR) and the Post-Irradiation-Examination including C-ring compression experiment, Micro-Raman spectroscopy, scanning electron microscopy and transmission electron microscopy. The results of the C-ring compression experiment showed that the circumferential strength of SiC composites remain stable at about ~194MPa after neutron irradiation. The crystallinity of SiC fiber grains was preserved accompanied by partial irradiation induced amorphization after 0.5 dpa. The carbon packets underwent moderate amorphization, but still remained excellent stability in terms of carbon bonding under neutron irradiation. Only partial disorder was observed in SiC matrix which can be attributed to the SFs structure. The PyC interface basically keeps its original morphology and graphitic nature although the continuous fringes were interrupted into small segments along with amorphous islands. Moreover, there is no irradiation-induced chemical mixing at the edges of the PyC interface with neighboring SiC grains. Only minor local amorphization occurred in the fibers, matrix and PyC interface under neutron irradiation, indicating an excellent resistance to irradiation-induced amorphization for Amosic-3 SiCf/SiC composites. Both C-ring compression and microstructure results indicate that Amosic-3 SiCf/SiC composites have excellent stability under low temperature neutron irradiation.

Invitation

P3-117

Atomistic Study of Interaction and Diffusion of Solutes and Impurities in Fe-Cr

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Fe-Cr ferritic alloys are attractive structural materials for the blanket in future fusion energy devices. While Cr is the main alloying solute, the alloys employ other elements, including W, Mn, C, V, and Si, that are not typically considered in modeling of radiation damage microstructure evolution. The particle-dispersion-strengthened variants of the alloys often employ Y-Ti-O nanoparticles, where Y, Ti, and O may not be fully consolidated into the particles during fabrication (remain in the matrix), due to complex kinetics and/or off-stoichiometric compositions. Furthermore, the alloy chemistry also changes during neutron irradiation, and additional elements including Re, Ta, Os, He, and H can be produced. Irradiation studies using spallation neutron sources also produce Ca and Sc. All these elements may influence the microstructure evolution. Using density functional theory, we study the interaction of these elements with interstitial Fe atoms and vacancies, and among themselves, and the diffusion mechanism of these elements and how they affect the diffusion of Fe atoms and vacancies. Current findings are the following. All solutes attract vacancies (Ca and Sc exert the strongest attraction). The majority of solutes repel each other, including those of the same element, except Mn-Si, Ca-(Ca, Mn, Si, and Sc). Cr, Mn, and Si form a stable mixed dumbbell with Fe (FeMn is the most stable), while others do not. Thus, Cr, Mn, and Si can diffuse via interstitial, in addition to vacancy mechanism. We will discuss how different stability and diffusivity of these elements may affect secondary phase precipitation during neutron irradiation.

Tentative

P3-118

Transmutation Effects on the Local Ordering Phenomena of Tungsten-Based Plasma-Facing Materials

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Tentative

Tungsten (W) is considered a leading candidate for structural and functional materials in future fusion energy devices. The most attractive properties of tungsten for magnetic and inertial fusion energy reactors are its high melting point, high thermal conductivity, low sputtering yield, and low long-term disposal radioactive footprint. However, tungsten also presents a very low fracture toughness, primarily associated with inter-granular failure and bulk plasticity, limiting its applications.

In recent years, several families of tungsten-based alloys have been explored to overcome the aforementioned limitations of pure tungsten. These might include tungsten-based high-entropy alloys (W-HEAs) and the so-called tungsten-based “smart alloys”. Given their proximity to the plasma, it is crucial to understand how the exposure of these candidate PFMs to the neutron fluxes expected in fusion reactors impacts their material behavior over time.

Recent studies have highlighted the significant impact of short-range order (SRO) on properties such as stacking fault energy, tensile strength, resistance to fatigue, magnetization, work-hardening, corrosion resistance, and irradiation damage. Therefore, it is crucial to improve our understanding of how neutron-induced transmutation in tungsten components, which creates transmutant elements like Re, Os, and Ta, affects SRO.

In this work, we present a novel computational approach that combines inventory codes, Monte-Carlo (MC) sampling, first-principles DFT electronic structure calculations, and Machine Learning techniques to understand the neutron-induced transmutation effects on the local chemical environment of tungsten-based PFMs. In particular, we investigate how their local ordering phenomena, such as SRO and local clustering, change during the course of irradiation due to transmutation.

P3-119

Electronic energy losses and sputtering mechanisms in Fe under light ion irradiation

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In fusion reactors, plasma-facing materials are continuously exposed to energetic ions, leading to complex processes that impact material performance and longevity. A key aspect is the transfer of energy from incoming ions to target electrons, which must be accurately modeled to improve our understanding of plasma-wall interactions. This study focuses on hydrogen (H) ion interactions with iron (Fe), examining how ion-electron energy transfer affects surface processes such as sputtering and ion reflection. Using ab initio simulations, we quantify the energy lost by hydrogen ions to electrons and analyze its dependence on local electron density. We also investigate a model Fe-based alloy relevant to EUROFER97 steel, as well as defect-containing lattices, to evaluate how structural modifications impact electronic energy losses.

To bridge these quantum-level insights with larger-scale models, we derive an electron density-dependent energy dissipation function for use in molecular dynamics (MD) simulations. This enables us to examine H sputtering from Fe surfaces across a range of impact energies and compare results with standard MD models that neglect electron-ion interactions. This presentation will discuss the methodology for linking first-principles data with MD simulations, key insights into electronic stopping in Fe, and comparisons with experimentally measured electronic stopping cross sections, providing validation for our simulated data.

Tentative

P3-120

Radiation induced defects in TiB_2: atomistic modeling and experimental validation

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Ultra-high temperature ceramics (UHTCs), particularly TiB_2, have shown great potential as alternative materials for the plasma facing components in fusion energy devices. Experiments reported intergranular cracking in TiB2 as a primary issue under neutron irradiation, likely driven by the anisotropy of vacancy and interstitial defects formed and evolved under irradiation. However, structure and properties of these defects are mostly unknown. In this research we applied atomistic modeling, both ab initio and classical molecular dynamics approaches, to study structure and properties of point and extended defects in TiB_2. The considered objects and properties include energy, mobility and interaction of vacancy and interstitials point defects, extended defects, and dislocation loops. Structure of some extended defects such as interstitial dislocation loops is verified within the specifically designed experimental program. The verified structure of extended defects allows a deeper understanding of their properties via large scale atomistic modeling for obtaining mechanisms and parameters necessary for predicting TiB2 behavior under irradiation conditions.

This research is supported by the Office of Fusion Energy Sciences, U.S. Department of Energy, under LAB-24-3295 project “Development of High-Performance Materials for Enabling Fusion Energy”.

Tentative

P3-121

Post-irradiation examination of neutron-irradiated W/Cu interfaces

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Post-irradiation examination (PIE) of neutron-irradiated tungsten has found the hardening mechanisms are dominated initially by radiation defects (dislocation loops, point defects, etc.) followed at higher dose by precipitation of fine clusters of transmutant Re and Os. However, there has been limited investigation into neutron-irradiation effects at interfaces of W with other materials (Cu, SiC, and steels). Laminates of W/Cu have been found to embrittle at moderate doses similarly to bulk W, but the mechanistic understanding is limited. Ion-irradiated nanoscale laminates have shown that defects and He atoms can accumulate at W/Cu interfaces, along with intermixing of W/Cu, but the kinetics of ion-irradiation is considerably different than neutron irradiation. There is a need to understand and mitigate radiation and transmutation-induced degradation mechanisms at these interfaces where failure is most likely to occur, which are critical for magnetic fusion reactor divertors.

Here, a thorough PIE campaign is underway to study W interfaces previously irradiated in the High Flux Isotope Reactor. It combines mechanical testing through tensile tests, thermal expansion tests, hardness mapping, and fractography to analyze failure mechanisms, and detailed microscopy using analytical scanning electron microscopy, transmission electron microscopy, and atom probe tomography to study defects, swelling, hydrogen retention, and compositional/ phase/ morphology changes and gradients. The results will improve the understanding of the degradation mechanisms specifically at the interfaces of W-alloys and other materials under irradiation. These results will then be used to develop W-alloy interfaces that are more resistant to interfacial degradation.

Tentative

P3-122

An investigation of irradiation induced grain growth in iron using x-ray diffraction experiments and molecular dynamics simulations

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Nanocrystalline materials have been proposed for use in future fusion reactors, as a high grain boundary density may act as an efficient sink for material defects. To test this hypothesis, iron is selected as a prototypical, fusion-relevant material. Molecular dynamics simulations are used to study the formations of nano-crystals through high shear strain deformation. These sheared cells are irradiated through collision cascade simulations up to 5 dpa and are compared with irradiated nanocrystalline cells generated by Voronoi tessellation. Irradiated pristine cells are also presented for comparison. These simulations reveal an irradiation-induced grain coarsening process, with the cells becoming single crystalline after ~ 2.5 dpa. An increase of in-grain dislocation density is also observed for the pristine and Voronoi cells, with the initially nanocrystalline cells saturating at a lower dislocation density that the initially pristine cells. This is accompanied by an increase in microstrain. For the sheared cells, the dislocation density reduces as grain size increases, which corresponds to a decrease in microstrain. A Toy Model is created which simulates volume exchange between grains based on different hypothesised exchange mechanism. Using this, we show that the data from MD simulations is not inconsistent with random grain growth. The MD simulations are directly compared to experimental data from ion-irradiated, nanocrystalline iron discs produced by high pressure torsion. GIXRD line profiles show a clear peak narrowing with increasing irradiation dose. This is caused by a microstrain reduction process with increasing irradiation, similar to that in the sheared cells in MD.

Tentative

P3-123

From microscopic defects to a full tokamak device: the European effort on modelling radiation effects in fusion reactor materials

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An operating D-T fusion power plant is a powerful neutron source, where the high-energy neutrons and gamma quanta produced by the fusion plasma are absorbed by the materials surrounding the plasma. The neutron exposure of fusion materials is expected to be many times that characterising the current generation of fission reactors, stimulating the highly non-linear evolution of defect and dislocation microstructure, the accumulation of transmutation products and radiation-stimulated retention of deuterium and tritium in the materials. The newly developed multiscale models for the radiation-driven non-linear microstructural transformations, relating microscopic models to engineering applications, enable modelling radiation-induced deformations on the scale of a full tokamak device. The environmental operating parameters for materials are fundamentally correlated and vary as functions of spatial location, with full reactor models helping define the context for materials testing and qualification. The current effort on modelling is combined with dedicated experiments, highlighting the complex nature of radiation effects in the high dose limit, driven by the defect generation, lattice-distortion-mediated interactions, and thermally activated relaxation and diffusion. The outstanding challenges for modelling include the treatment of combined effects of radiation, chemical, magnetic, and thermal environments where a self-consistent micro-to-macroscale simulation-based analysis can aid engineering design studies.

Invitation

P3-124

Modeling the deformation and ductile damage of irradiated EUROFER97 for fusion reactor design optimization

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EUROFER97 is a Reduced-Activation-Ferritic-Martensitic Steel, selected as one of the baseline structural materials chosen for fusion reactor components due to its favorable thermo-mechanical properties and swelling resistance. However, irradiation diminishes its strain hardening and consequently its uniform elongation (UE). The loss of UE as a failure criterion in design rules is simple, but excessively conservative and potentially rendering the material unfit for use in its irradiated state. The remarkably high fracture strain of irradiated EUROFER97 suggests availability of sufficient ductility to allow for use beyond UE. Thus, a failure criterion based on reduction in area as a measure of ductility can be a viable option for determining the allowable material use while assuring safe operation. To explore this, a thermodynamically consistent finite strain coupled deformation and ductile damage model is developed within the viscoplasticity framework, incorporating elements from Aktaa-Petersen (2011) and Tvergaard-Needleman (1984). The model incorporates the influence of irradiation, strain rate, and triaxiality and is calibrated using experimental data (0-70 dpa). The model's application on high-triaxiality ($\geqq 0.33$) tensile specimens at RT and 300°C demonstrate that it can reliably and accurately describe the influence of irradiation on the fracture strain and the inverse relationship between fracture strain and triaxiality. Based on the results, this work provides a foundation for revising the current design rules and proposes true fracture strain as a metric to establish the limits of reduction in area for safe operation of EUROFER97 beyond UE, thus optimizing the use of the material in fusion reactor applications.

Tentative

P3-125 "Presenting Author: Blanca Biel"

Application of DFT and machine-learning trained potentials to study radiation-induced defects in ferrite-cementite interfaces

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Invitation

One of the materials to be investigated in IFMIF-DONES - a future neutron irradiation facility - is EUROFER97, a reduced activation ferritic/martensitic steel and the European reference for DEMO's First Wall and Breeding Blanket. Neutron irradiation-induced defects significantly impact mechanical properties, with carbon impurities playing a crucial role in defect evolution in Fe-based alloys. However, the arrangement of C atoms in a bcc-Fe lattice remains unclear.

We investigate C configurations in the presence of cementite, a common carbide in steels, using Density Functional Theory (DFT) and Molecular Dynamics (MD) with machine-learning interatomic potentials trained via the Neural Equivariant Potential framework. Our focus is on defect formation at ferrite/cementite (α/θ) interfaces, as cementite can emit or absorb C atoms, influencing defect evolution.

Following a DFT and MD study on ferrite/cementite interfaces [1], we apply the trained potential to study formation and bonding energies and migration of defects to verify that it can reproduce the results with near DFT accuracy while reducing the computational demand close to that of MD. We find that C atoms destabilize α/θ interfaces, resulting in a likely defect sink under irradiation.

Using this potential, we analyze defect formation, migration, dislocation behavior near the α/θ boundary and primary knock-on atom cascade simulations. Our results provide insights into cementite's role as a defect sink and its effect on C redistribution under irradiation.

[1] Canca, P., Fu, C. C., Ortiz, C. J., Biel, B. (2025). A Combined DFT and MD Study on Interface Stability in Ferrite-Cementite Systems. Under revision (Acta Materialia).

P3-126

Cryogenic ion irradiation of high-temperature superconductors with in situ positron spectroscopy & AC susceptibility measurements of critical temperature

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Superconducting magnets made of yttrium barium copper oxide (YBCO) tapes enable compact power plant designs, which have accelerated the commercialization of fusion energy. Efforts to understand the radiation tolerance of YBCO have culminated in a hypothesis, formulated by Eisterer, stating that critical current density (Jc) declines as radiation-induced defects accumulate and enhance Cooper pair scattering (ρ). In this model, ρ is proportional to the change in critical temperature, ΔTc. However, all experiments to date have measured either the superconducting properties or the microstructure. Moreover, the vast majority of data proceeds from irradiations conducted above 300 K; not representative of the fusion environment. We present a first attempt to measure Tc and characterize the defect landscape simultaneously. We irradiated YBCO crystals (4x 4x 0.25 mm, twin density 5% of their post-irradiation value. We present the results of the latest campaign: where ΔTc and MLT are measured in-situ, between irradiation and isochronal anneal steps, with an outlook on the possibility of testing the theoretical prediction of Eisterer. If confirmed, this model will become an essential design tool to set the minimum neutron shield thickness needed to achieve reliable, economical fusion electricity production.

Tentative

P3-127

Defect formation, diffusion, and recombination in $YBa_2Cu_3O_7$-δ in a fusion environment: A first-principles study

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In a fusion tokamak device, high temperature superconducting (HTS) magnet materials, such as $YBa_2Cu_3O_7$-δ, will be exposed to high energy neutrons that may impact the superconducting properties and ultimately lead to degradation of the magnetic field strength. First-principles density functional theory (DFT) calculations have been used to investigate the energetics of point defects in $YBa_2Cu_3O_7$-δ. The formation energies of point defects indicate oxygen defects are dominant, followed by Cu defects. Ba and Y defects are less energetically favorable. DFT combined with the nudged elastic band method has been used to evaluate the diffusion activation energy of point defects. The DFT results predict an anisotropic diffusivity of oxygen defects. Oxygen prefers to diffuse along the b-axis on basal planes via a vacancy mechanism, while oxygen diffuses along the c-axis via both interstitial and vacancy mechanisms, but at relatively high temperatures. Excess Ba and Y prefer to create antisite defects and Cu interstitials, rather than being at interstitial positions. Cu interstitials can diffuse in bulk $YBa_2Cu_3O_7$ with an energy of about 1.1 eV. Cation vacancies are much less mobile than their interstitials and oxygen vacancies. The required energies in the recombination process of oxygen Frenkel pairs indicate different recovery rates of superconducting properties of irradiated $YBa_2Cu_3O_7$ following thermal annealing with increasing temperatures. The effect of oxygen stoichiometry on defect behavior has also been investigated. Such computational studies, when coupled with cryogenic neutron irradiation, offer the potential to accelerate the qualification of HTS magnet materials for fusion applications.

Tentative

P3-128

Molecular dynamics simulations of radiation damage processes and effects near grain boundaries in tungsten

JongHyeon Park[1], Akira Hasegawa[2], Ryuta Kasada[2], Takuji Oda[1]

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In nuclear fusion reactors, materials such as tungsten for plasma-facing components undergo severe radiation damage from high-energy neutrons and atoms. The radiation defects impede dislocation motion, leading to radiation hardening and embrittlement. Grain boundaries (GBs) act as defect sinks, creating defect-free zones known as denuded zones, and modify the radiation effects. Therefore, GBs can be used to improve the radiation tolerance of materials, although the detailed processes and mechanisms involved are not fully understood.

Molecular dynamics (MD) is a widely used computational method to study the radiation damage process and the resulting changes in material properties. In this study, we simulated the radiation damage formation and evolution near GBs in tungsten and analyzed the effects of GBs on radiation hardening and embrittlement using MD. To accelerate the simulation, we utilized the Primary Radiation Damage structure generator (PRDgen) code, which was developed in our previous study to rapidly generate primary radiation damage structures using machine learning trained on MD collision simulation data. We used the PRDgen code to induce radiation damage near GBs and conducted MD simulations to investigate the following changes in damage structure and dislocation mobility. The simulation results confirm that a denuded zone is created due to efficient recovery of point defects at GBs and the dislocation motion is affected by the denuded zone. We plan to compare the simulation results with experimental results in the presentation.

Tentative

P3-129

First principles-based approach to calculation of electrical and thermal conductivity in functional and structural materials

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Here we demonstrate the first principles-based approach to calculation of electrical and thermal conductivity in structural materials. It allows for consideration of thermal transport through both electrons and lattice. Further, the formalism is used to investigate transport properties of Cu-based multicomponent alloys. Copper alloys are the primary candidate for heat exchangers within plasma facing components of future fusion reactor because they exhibit advanced mechanical and thermal transport properties. While pure Cu has very high thermal conductivity, its strength must be improved to be used as a part of fusion reactor structural component. This goal is usually achieved by alloying pure Cu with elements with a low limit of solubility which form nanoprecipitates. For development of potentially promising materials, it's necessary to consider thermal conductivity modification due to alloying. This was made within the Green-Kubo formalism. The electron scattering on chemical disorder is treated within the coherent potential approximation (CPA). The lattice part of the thermal conductivity was calculated within three phonon scattering approximations. Good agreement with available experimental data was demonstrated.

Research sponsored by the Office of Fusion Energy Sciences, U.S. Department of Energy, under
contract DE-AC05-00OR22725 with UT-Battelle LLC.

Tentative

P3-130

Understanding deuterium's effect on defect production in irradiated tungsten

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Tungsten is proposed to be used as the divertor and wall material in the ITER reactor, and would therefore be subjected to various stresses and high levels of irradiation. Another key issue concerns the build-up of impurities such as hydrogen. Understanding both the damage accumulation and the embedding of deuterium and tritium, in tungsten, is therefore crucial.

Multiple experimental studies have investigated deuterium retention in irradiated tungsten. By first irradiating the tungsten samples to high doses and then implanting the deuterium, roughly 1.7% deuterium was retained [1]. In another study, the samples were irradiated and implanted with D multiple consecutive times, which resulted in up to 4% retained D [2]. To investigate these differing amounts of retained deuterium and possibly identify the underlying mechanisms we have conducted various irradiation simulation schemes in deuterium-embedded tungsten, using molecular dynamics (MD).

We performed a comprehensive set of overlapping and single cascade simulations, on both pure tungsten and cells with varying amounts of D [3]. Additionally, we investigated the impact of PKAs on voids containing 1-5 D atoms per vacancy. Our results demonstrate that filling the vacancies with D leads to the creation of a significantly larger number of defects.

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Tentative

P3-131

Accurate, Robust and Interpretable Property Prediction Framework for Reduced-Activation High-Entropy Alloys with Small Datasets

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In recent years, machine learning (ML) has profoundly changed material design paradigms. However, the scarcity of experimental data on fusion reactor materials results in small and sparse datasets, hindering reliable property predictions and posing a crucial challenge for ML-assisted design. To address this issue, we propose a robust framework for accurate property prediction using small datasets by integrating domain adaptation and symbolic regression. By leveraging a source domain with similar and sufficient samples, the framework can identify domain-invariant features that adapt the source domain to the small dataset, i.e., the target domain, and significantly enhance the model performance. For hardness prediction of reduced-activation high-entropy alloys (RAHEAs) with only nine samples, we utilized the data from conventional high-entropy alloys (HEAs) as the source domain to adapt the RAHEA data in the target domain. Symbolic regression was used to generate and screen domain-invariant features from non-linear combinations of element parameters and compositions. The domain-invariant features can significantly enhance the accuracy of ML models by 30% compared to that of models using only element-based features. Furthermore, these features offered comprehensive physical insights into hardness in terms of crystal structure, bonding, and strengthening mechanisms from their explicit mathematical expressions. This framework can achieve accurate and robust prediction with interpretable domain-invariant features on limited and sparse data. It is adaptable to the scenarios involving small datasets, offering a pathway for efficient, accurate, and robust ML-assisted composition design for fusion reactor materials.

Tentative

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Study on the Mechanism of Nanoparticle Induced Void Nucleation in ODS Steel under Complex Stress States

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Due to its excellent high-temperature performance, creep resistance, and irradiation resistance, ODS-RAFM steel is considered one of the most promising candidate structural materials for future fusion reactors. However, under the complex stress states in the service environment of fusion reactors, the issue of ductility degradation in ODS-RAFM steel caused by 1-10 nm nanoparticle induced void nucleation, remains a critical concern. In this study, we employed molecular dynamics simulations to investigate the process and underlying mechanisms of nanoparticle-induced void nucleation in bcc iron under multiaxial tensile conditions. The effects of different stress states and nanoparticle sizes on the void nucleation process were analyzed in detail. The results reveal that nanoparticle-induced void nucleation occurs in two stages: initially, atoms debond at the cluster-matrix interface, followed by the rapid nucleation of voids associated with dislocation emission. Increasing the size of the nanoparticle results in the nucleation of voids at lower strains, suggesting that as the nanoparticle size increases, the material becomes more susceptible to ductile damage during tensile deformation. In addition, compared to uniaxial tension, voids nucleate at lower strains under the complex stress conditions of biaxial and triaxial tension. The influence of 1 nm nanoparticle on void nucleation is slight, and the void nucleation process for a bcc iron matrix with 1 nm nanoparticle is similar to that observed in pure iron.

Tentative

P3-133

Distinct ion irradiation responses in refractory multi-principal-element alloys

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Refractory multi-principal-element alloys (RMPEAs) have recently gained significant attention, not only for their outstanding high-temperature mechanical properties that meet the demands of novel reactor designs, but also for their potential to tune intrinsic defect dynamics, enabling innovative strategies to enhance irradiation resistance. In this presentation, we report unique ion irradiation responses in a series of RMPEAs that hold desired strength-ductility combinations across a wide temperature regime. First, we show that, compared with V and the conventional vanadium alloy, ternary VTaTi shows smaller loop size and lower density at both room and elevated temperatures, resulting in weaker irradiation hardening. However, simply increasing the number of elements doesn't necessarily improve irradiation resistance, as evidenced by the more severe defect aggregation observed in quaternary VNbTaTi. In contrast, HfNbZrTi shows exceptional capability in suppressing interstitial defect evolution and abnormal irradiation-softening behavior at room temperature. Combining ab initio calculations and atomic-level elemental mappings, we propose that lattice distortion and local chemical fluctuation play key roles on the defect dynamics in RMPEAs, while the evolution of short-range order is responsible to the softening phenomenon. Despite the strong suppression of interstitial defects, we further report that the sizes of helium bubbles in these RMPEAs are considerably larger than those in elemental metals, owing to both the reduced vacancy formation energies and the low-energy vacancy migration pathways. This presentation highlights the distinctiveness and complexity of irradiation behavior in RMPEAs, which may draw further research attention in this field.

Tentative

P3-134

Cluster Dynamics Simulations of Cascade Overlap Model for 800H Alloy

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The austenitic 800H alloy provides a benchmark for emulating neutron damage in face-centered cubic materials. Understanding the microstructural evolution under high-dose particle irradiation conditions is essential for accelerating the deployment of advanced nuclear reactors, and this approach can be directly adapted to fusion structural materials. We are creating a predictive tool by integrating ion irradiation data with an improved version of the cluster dynamics code, Xolotl. This code simulates the evolution of faulted and prismatic loops, as well as helium-vacancy clusters and cavities, under varying irradiation temperatures and dose levels for both ion and neutron irradiation. Our primary focus is to understand how the overlap of displacement cascades affects pre-existing defects as the dose increases. For benchmarking, Xolotl prediction of loop sizes and densities are compared with existing experimental results.

Tentative

P3-135

In-situ TEM investigation of irradiation induced defect evolution in neutron-irradiated Tungsten

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As a baseline armor material for plasma-facing components in future fusion reactors, Tungsten (W) is exposed to extreme environmental conditions, including neutron irradiation at high temperatures, which leads to microstructural changes that significantly affect its mechanical properties. Therefore, it is crucial to study the impact of temperature on the irradiation induced defects, their nature, type, morphology, evolution etc., in plasma facing W to better understand degradation of mechanical and thermo-physical properties and to further optimize materials development.

This study primarily investigates the evolution of irradiated-induced defects in ITER grade product (IGP) W. The samples prepared were irradiated to a dose of 1.06 dpa at 400 °C. In this work, in-situ transmission electron microscopy (TEM) annealing experiments were performed across a temperature range up to 1000 °C to characterize the evolving defects.

Experimental observations indicate that microstructural changes become more pronounced within a specific temperature range, while minimal changes occur at lower temperatures. At temperatures between 600 °C and 800 °C, significant modifications in defect structures were observed, suggesting the presence of a critical temperature range for defect evolution. Isothermal annealing experiments provide valuable insights into thermal stability and defect transformation, extending knowledge for optimizing material performance in fusion environment.

Tentative

P3-136

A high-fidelity machine learned interatomic potential for radiation damage in $YBa_2Cu_3O_7$

Ashley Dickson[1], Davide Gambino[2], Daniele Torsello[3], Samuel Murphy[1], Flyura Djurabekova[4], Francesco Laviano[3], Mark Gilbert[5], Duc Nguyen-Mahn[5], Niccolo Di Eugenio[3]

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High-temperature superconductors (HTS) are promising materials for next-generation fusion reactors, owing to their remarkable magnetic field strengths at elevated temperatures. However, the extent of radiation induced damage in these materials under operational conditions is poorly understood. Direct experimental observation of the damage process under operational conditions is challenging, therefore, there is an increased reliance on modelling. In this context we employ molecular dynamics (MD) simulations to predict the structural response of $YBa_2Cu_3O_7$ (YBCO) to neutron irradiation. Key to accurately predicting the material's response to damage is being able to accurately replicate the interactions between the ions in the system.

Machine-learned potentials (MLPs) offer the accuracy of an electronic structure approach, such as Density Functional Theory, at a fraction of the computational cost, making them ideal for modelling processes such as radiation damage. Here, we develop and validate a high-fidelity tabulated Gaussian Approximation Potential (tabGAP) for YBCO, trained on a diverse dataset that ensures accurate representation of atomic interactions in the highly disordered heat-spike of a collision cascade. In order to demonstrate the efficacy of our tabGAP, we perform collision cascades at a range of energies. This model enables more reliable predictions of fusion-relevant damage in HTS materials, providing important insight into the shielding requirements of future fusion devices.

Tentative

P3-137

Effect of radiation defects on grain boundary evolution under shock loading

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Tungsten Grain boundary (GB) failure under shock loading is a key factor to estimate the performance of tungsten as a plasma facing material in nuclear fusion reactors. In this work, molecular dynamics method is used to investigated the GB evolution after absorption of different radiation defect clusters under shock loading. Different to the cases without radiation defects, after absorption of interstitial dislocation loops and voids, two mechanisms have been identified for the decrease of critical shock velocity (vc) and the related GB failure. The 1st mechanism is direct void growth, accompanied by the appearance of disordered structure around voids in GB, originating from the remaining vacancy clusters that could not be annihilated by compressive stress wave. In the 2nd mechanism, activation of slip systems, the dislocation formation, motion and multiplication, as well as void nucleation and growth dominate the failure process. The higher potential energy and local stress concentration around a defect absorption region in GB are recognized for atomic motion deviations from shock loading direction, resulting in formation of disordered structure, activation of slip system and dislocation multiplication with lower vc under coupling effects of radiation damage and shock loading. These results indicate that under a high dose of radiation damage, GB failure induced by shock loading should be considered seriously to estimate the lifetime of tungsten-based plasma facing materials.

Tentative

P3-138

Evaluation of internal sink intensity of nano-particles in neutron-irradiated ODS steels

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Oxide Dispersion Strengthen (ODS) steels are considered as a promising material for fusion reactors. Main reason is anti-irradiation properties, as well as high strength. Many studies have been done by several irradiation methods, however, not focused on the role of nanoparticles (NPs) as internal sinks. Recently, we have investigated on nano-micro structures of ODS steels irradiated by electron, heavy-ion and neutron, and then confirmed coherent character of YlXmOn NPs and their high stability during irradiation, but have not studied on the tensile strength and annealing properties. The aim of this study is to clarify the effect of internal sink intensity of NPs on the radiation-hardening and its recovery at high temperature.

Two ODS steels are Fe-14Cr-7Al-0.5Ti-0.4Y (ODS1) and with 0.3Zr (ODS2). Neutron irradiation was carried out in BR2 at 290°C, where LIBERTY capsule allowed a series of irradiation at three different doses up to up to 1E24 n/m2. Additionally, post irradiation annealing was done at 510 and 800°C. PIE of tensile test, HRTEM and Positron lifetime (PA) were carried out.

The radiation-hardening developed and increased with dose. The hardening was ODS1>ODS2, and the NPs density was ODS1<ODS2, meaning the hardening was reduced by high sink intensity of NPs. Another important result was that the hardening showed “very quick recovery” by annealing at 510 and 800°C.

Those results are telling us that NPs act as internal sinks for radiation-induced defects with high efficiency. Combined with PA, we will clarify the relation among the hardening, recovery behavior and internal sink intensity.

Tentative

P3-139

Study of the influence of irradiation temperature on the destabilization mechanisms of the near-surface layer of BeO ceramics

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Beryllium ceramics, particularly beryllium oxide, can be possibly used as materials for new-generation fusion and fission reactors, due to their high thermal conductivity and low neutron absorption cross-section. Also, resistance to thermal effects, including the ability to maintain thermal stability up to temperatures of about 2000 - 2400°C, allows to consider them as materials operating in extreme conditions. The main purpose of study is to determine radiation damage effect associated with accumulation of implanted helium in the near-surface layer of BeO ceramics and changes in thermo-physical parameters, including changes in thermal conductivity and thermal resistance. As radiation damage, we considered the effect of low-energy He^{2+} ions. In experiments, it was found that the main role in the change of structural features is played by vacancy defects, the density of which has a direct dependence on the temperature at which irradiation is performed. Increase of irradiation temperature from 300 to 700 - 1000 K leads to acceleration of surface layer destruction processes associated with diffusion of point defects, their agglomeration and aggregation in the form of gas-filled inclusions in pores formed as a result of deformation distortions of damaged layer. It was determined that accumulation of amorphous fraction in damaged layer leads to its destabilization and, as a consequence, to a decrease in thermal conductivity and resistance to temperature variations, which is most evident in the case of high-dose influence by low-energy ions.

Invitation

P3-140

Study on the Influence of Alloy Elements on the Radiation Effect of Vanadium based Alloys by Positron Annihilation

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Vanadium alloys are promising candidate for the first wall structural material of fusion reactors, which must withstand high temperature and high-energy neutron irradiation during operation. In this study, V, V-4 wt.% Cr, V-4 wt.% Ti, V-4 wt.% Cr-4 wt.% Ti, and V-4 wt.% Cr-4 wt.% Ti-0.3 wt.% Y alloys were irradiated with 180 keV He ions at room temperature with a damage peak of 2.3 dpa. The positron annihilation technique was used to analyze and measure the changes in positron lifetime, S-parameter, and W-parameter of V-based alloy plates after irradiation to reflect the changes in surface defect information before and after irradiation. Doppler broadening spectrum (CBD) was also measured to reflect the influence of elements on the evolution of defects in V-based alloy plates after irradiation.

The experimental results showed that there was no significant change in the positron lifetime of V, V-4Ti, V-4Cr, and V-4Cr-4Ti alloys after irradiation, but the positron lifetime of the unirradiated V-4Cr-4Ti-0.3Y alloy increased due to the formation of Y_2O_3 nanoparticles. According to the results of the S-W parameter, all samples formed vacancy-type defects after irradiation, but the change in the S-W pattern of V-4Cr-4Ti-0.3Y was different from other samples. No precipitates were found in the unirradiated V-4Ti and V-4Cr-4Ti alloys by TEM. However, according to the analysis of CDB measurement results, it is known that there are Ti rich clusters in V-4Ti and V-4Cr-4Ti alloys, and the precipitation type of V-4Cr-4Ti-0.3Y has undergone significant changes compared to the other two Ti containing V alloys.

Tentative

P3-141

Machine learning-based multi-scale simulation approach for predicting effective thermal conductivity in tungsten monoblock

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Tungsten (W) monoblocks, as plasma-facing materials, are subjected to high heat flux and high-energy neutron irradiation. The thermal load capacity of the divertor limits the maximum operating power of fusion reactors. Therefore, identifying divertor materials that can maintain high thermal conductivity under these harsh conditions is crucial for nuclear fusion energy research. The irradiation effects on divertor materials are a typical multi-scale physical phenomenon, spanning from atomic processes, microstructural evolution, and macroscopic properties. To achieve a more accurate analysis, we integrate a phase-field approach with the finite element method (FEM) to capture how microstructural changes affect macroscopic thermal properties. The phase-field model simulates the microstructural evolution of defects at the nanoscale under irradiation and computes the corresponding effective thermal conductivity. These conductivity values are then assigned to the nodes of the FEM to derive a more accurate effective thermal conductivity for millimeter-scale W divertors. However, since FEM analysis requires extensive phase-field simulation data, directly computing the effective thermal conductivity for each microstructure is computationally prohibitive. To accelerate this process, we introduce a ResNet18-LSTM machine-learning framework that accelerates the calculation of thermal conductivity by incorporating phase-field microstructural information, significantly enhancing both computational speed and accuracy. As a result, our model enables accurate prediction of the macro-scale temperature distribution and effective thermal conductivity of W monoblocks while accounting for the microstructural evolution of irradiated-induced defects. This integrated model provides an efficient and robust multi-scale simulation approach for understanding and predicting the evolutions of irradiation-induced defects in nuclear materials.

Tentative

P3-142

In-situ Scanning Electron Microscopy Study of the Thermal Creep Bebavior in Body-Centered Cubic Metals

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One of the key factors ensuring a long-term, continuous, stable and safe operation of nuclear reactors lies in the nuclear energy materials employed in reactor cores (for fission reactors) and first-wall divertors (for fusion reactors). These materials are exposed to extreme service conditions, including high temperature, high-energy particle irradiation, and stress, etc., which leads to significant degradation of their macroscopic mechanical properties. Creep behavior is one of the key issues determining the long-term service performance of nuclear energy materials. We selected body-centered cubic metals tungsten and iron, which are commonly used in reactors, as model materials. In-situ tensile loading (with a resolution of 100 nm) in a scanning electron microscope (SEM) was carried out to investigate the thermal creep behavior of the materials under different temperatures and tensile stresses. By analyzing the measured creep curves, macroscopic mechanical properties such as the activation energy, activation volume, and stress index were obtained. Micro-structural information at different creep stages was obtained by using in-situ SEM and electron back scatter diffraction (EBSD), as well as focused ion beam (FIB) and transmission electron microscopy (TEM). Through a comparative analysis of the macroscopic and the microscopic results, we provided direct evidence for elucidating the thermal-creep mechanism.

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P3-143

Microstructural evolution study of ion irradiated Laser Direct Energy Deposition and Laser Powder Bed Fusion 316H.

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316H stainless steel is used as a structural material in a wide variety of nuclear applications due to its mechanical properties, corrosion resistance, and irradiation performance up to high operating temperatures (above 500°C). This alloy has attracted wide interest in additive manufacturing (AM) processing to print complex, rapidly fabricated structures for nuclear components. The rapid solidification and thermal gradients characteristic of AM processing result in cellular structures, internal stresses, and alloying component segregation that affect mechanical properties and radiation response. In this study, AM 316H was printed by Laser Powder Bed Fusion (LPBF) and Laser Direct Energy Deposition (LDED), then ion irradiated with 5 MeV Fe ions to 10 dpa at 400°C, 500°C, and 600°C. TEM and STEM were used to study the temperature effects on the cellular structures of the two print methods and the associated print parameters. Due to lower cooling rates, LDED has more pronounced chemical segregation than LPBF, including Cr and Mo enrichment near boundaries. Both the LDED and LPBF have unique cellular structures, but post-irradiation examination revealed that irradiation-induced defects alter the as-printed structures. The LDED samples have larger dislocation loops within the cellular structure compared to the LPBF samples at comparable doses. As the temperature increases, the dislocation loops grow due to the increase in the mobility of defects. The temperature effects on the dissolving of the cellular structures and the chemical segregation of Cr and Mo and other irradiation-induced microstructural evolution will be presented.

Tentative

P3-144

Damage saturation in low-temperature ion irradiation of Fe thin films observed by in-situ electrical resistivity measurements

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The idea that the accumulation of radiation damage in materials may reach a point of saturation at very high dose has been proposed since the early days of radiation damage physics. Recently, there have been several experimental observations and theoretical simulations in support of this hypothesis. This effect is particularly significant for the lifetime of materials operating in high radiation environments, such as those encountered in advanced fusion energy systems.

In this work, using low-temperature (~20 K) ion irradiation coupled to electrical resistivity measurements, we were able to observe in-situ the evolution of damage in Fe thin films from linear defect accumulation at low dose (10^{-5} dpa) towards damage saturation at ~1 dpa. The results are in good qualitative agreement with recent theoretical predictions.

Tentative

P3-145

Effects of cavity formation on the magnetism of iron and iron-chromium alloys

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Fe-Cr alloys are widely used as structural materials in facilities subjected to irradiation environments. The magnetic properties of these alloys can be modified by neutron irradiation during operation, which may affect the performance of the equipment. Therefore, understanding this phenomenon is essential. We are conducting ion irradiation experiments on high-quality film samples to investigate the effects of irradiation. Considering the irradiation conditions of a fusion reactor, it is important to examine the effects of cavity formation. In this study, we performed He+ irradiation on Fe and Fe-Cr films and analyzed their microstructure and magnetic properties.

Thin films of Fe, Fe-10Cr, and Fe-20Cr, each with a thickness of 200 nm, were epitaxially grown on MgO(001) using UHV electron beam deposition. Subsequently, He^+ irradiation was performed at room temperature (30 keV, 7.5×10^{20} ions/m^2). The damage level reached 5 dpa, and a significant amount of He? was retained within the films, as confirmed by SRIM calculations. The structure was examined through XRD, EBSD, and TEM techniques, while the magnetic properties were assessed using a VSM and a Kerr microscope.

From the cross-sectional TEM observations, the formation of high-density cavities was confirmed in all samples. Magnetic measurements indicated that there were no significant changes in Fe, but in Fe-Cr alloys, magnetic hardening was observed. Differences in the magnetic domain structure were also confirmed, with closure domains present in Fe and stripe-shaped domains in Fe-Cr. These differences were discussed from the perspective of cavity formation.

Tentative

P3-146

Damage by MeV ion implantation in Fe and FeCr from OKMC simulations

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Ion implantation is often used as surrogate for neutron irradiation [1]. However, there are clear differences that need to be taken into account when extrapolating from ion to neutron damage [2]. Such correlation requires of validated models for damage production and evolution.

We use object kinetic Monte Carlo (OKMC) of ion implantation at MeV energies in Fe and FeCr alloys to study changes in the microestructure. The damage produced is obtained from a model that combines the recoil spectra from binary collision approximations (SRIM) and molecular dynamics results of collision cascades (CascadesDB). Experimentally [3], three different regions are observed as a function of depth: close to the surface there is a depleted region where defect concentration is very low, a central region with low concentration of defects but large sizes and a third region with a significantly higher defect concentration of much smaller sizes. Our OKMC simulations, performed in systems sizes up to 1 micrometer, show clearly the depleted region close to the surface. A region with larger defects is observed around the damage peak followed by a region with smaller clusters, although the differences are not as strong as those observed experimentally. Several effects are studied, such as the influence of the implanted ion, temperature and dose.

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Tentative

P3-147

Computer study of sputtering in fusion-related materials: An in-depth analysis of sputtered species

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The selection of suitable plasma-facing materials is essential to ensure the safety and economic viability of fusion reactors. This selection is complicated by plasma-wall interactions, where erosion is one of the key issues. Erosion can alter the surface of wall materials, reducing their durability and thermomechanical characteristics. Moreover, eroded impurities may infiltrate the plasma, leading to fuel dilution and energy loss. Furthermore, the redeposition of these impurities plays an important role in tritium retention, presenting a major challenge in reactor development. Within the reactor environment, plasma transport processes are the determining factor in plasma impurity content. The effectiveness of these processes depends on the attributes of the sputtered species, such as their kinetic energy, emission angle, and molecular characteristics of the sputtered materials, including structure and vibrational state. This study aims to evaluate the characteristics of sputtered species employing computational techniques. Using molecular dynamics (MD) simulations, we examined beryllium and tungsten sputtering by hydrogen isotopes to obtain the kinetic energy and the polar angle distribution of the sputtered particles. In addition, we analyzed the structural and vibrational properties of sputtered molecules via MD simulations. Our study reveals that the impact energy and angle of the projectile substantially influence both the kinetic energy and the emission angle of sputtered species. Furthermore, the structural and vibrational properties of sputtered molecules are also influenced by the characteristics of the projectile.

Tentative

P3-148

Interactions of Transmutation Products with Radiation Induced Point Defects in Tungsten from First Principles and Machine Learning

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One of the most challenging aspects of fusion reactor operation is that materials will continually change due to the creation of defects under neutron irradiation. In tungsten (W), irradiation embrittlement is accelerated by the formation of transmutation products (e.g., Re, Os, and Ta) and their subsequent interaction with grain boundaries, dislocation loops, and radiation-induced point defects. In this study, the interactions of Re with point defects in W are investigated using density functional theory (DFT) to inform machine learning (ML) models in the prediction of defect formation energies, binding energies, and migration barriers. From DFT, Re is shown to reduce the vacancy formation energy while increasing the binding energy between the vacancies and Re, indicating that the barrier to form vacancies is suppressed by Re and those vacancies are attracted to the Re atoms. The distribution of binding energies to different defect cluster configurations is attributed to local ordering of Re-vacancy pairs. Vacancy mobility was also examined using Nudged Elastic Band (NEB) calculations, which reveal that migration barriers are considerably reduced with increasing Re concentration. Collectively, these results indicate that vacancies act to stabilize Re clusters while Re enhances vacancy mediated diffusion, in turn promoting Re cluster formation. DFT-informed ML models were developed and incorporated into the computational workflow to significantly reduce computational resource requirements and allow for meaningful correlations to be uncovered between point defect energetics and underlying defect configurations.

Tentative

P3-150

Migration mechanisms of mono-/di-interstitials in vanadium with alloying Ti/Cr

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Self-interstitial atoms (SIAs) are the dominant self-defects at equilibrium in metals, but many questions such as their diffusion characteristics are still open, and studies of the Ti/Cr/Fe effect on the kinetics of SIAs are very scarce in vanadium alloys. This work systemically investigated the migration mechanisms of mono/di-interstitials as well as the effect of alloying elements Ti/Cr/Fe on the kinetics of SIAs using first-principles calculations. The findings reveal that SIA exhibits three migration mechanisms, including one dimensional (1D) migration, translation-rotation and translation, corresponding energy barriers of 0.008, 0.19 and 0.44 eV. Different migration pathways are activated at high temperatures, which supports the experimental observation of 3D migration of SIAs under high temperatures. The di-SIA (two parallel SIA) migration by both successive and simultaneous jumping paths, the lowest migration barrier is 0.009 eV via successive jumping path passing a translation state. Moreover, based on three migration mechanisms of SIA, SIA migrating to Ti region possesses the lower barrier than pure vanadium, in verse the barrier is relative high and then hinders the SIA diffusion. Thus, the alloying Ti has a pinning effect on SIAs. While Cr promotes the evolution of SIA into a more stable V-Cr interstitial. This work provides valuable information for understanding the migration mechanisms of SIAs in vanadium alloys and provides accurate input parameters for large-scale simulations.

Related achievements:

[1] Mingliang Wei, Pengbo Zhang* et al. J. Nucl. Mater. 603 (2025) 155407.

[2] Xing Wang, Pengbo Zhang* et al. Fusion Eng. and Des. 202 (2024) 114402.

Invitation

P3-151

Predicting irradiation hardening and microstructural evolution in ion-irradiated Eurofer97: a nanoindentation study supported by CPFEM and TEM

Tymofii Khvan[1], Katarzyna Mulewska[1], Łukasz Kurpaska [1], Witold Chromiński[1], Michał Adam Stróżyk[1]

[1]National Centre for Nuclear Research

Reduced activation ferritic/martensitic (RAFM) steels are the main candidates for the construction of structural components in future fusion and Gen IV nuclear reactors. To ensure stable reactor employment, RAFM-based materials require efficient methods for their characterization under harsh operational conditions. Constantly enduring neutron irradiation, their mechanical properties degrade and may cause a failure of the component. However, neutron irradiation for research purposes is an expensive and long process, so it becomes a limiting factor to investigate its effect and deliver new research data. Hence, a safer, more paced, and cheaper solution of ion irradiation as a tool for surrogating the neutron damage is becoming more popular. To correctly estimate the impact of ion damage, non-uniformly distributed on the sub-surface, nanoindentation technique is widely applied. The presented study demonstrates a semi-empirical approach to effectively interconnect the ion and neutron radiation-induced hardening in Eurofer97, introduced in a range of irradiation conditions. The applied set of tools, based on nanoindentation and tensile tests, as well as their simulations using crystal plasticity finite element method, allows us to extract the irradiation effect on the material law, and accurately reproduce the experimental data. Ultimately, the analysis performed on an ion-irradiated specimen can provide the macroscale (neutron irradiated) yield stress values in a range of dpa doses, which correlate with the literature. The complementary investigations of the microstructural evolution done by focused ion milling and transmission electron microscopy are compared with their computational analogue to confirm the predictive capability of the method.

Tentative

P3-152

The mechanism of titanium suppressing the clustering behavior of vacancy in vanadium alloys

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The energetics and dynamics of vacancy clusters in presence of Ti in vanadium alloys were systemically investigated by first-principles calculations to elucidate the mechanism of Ti suppressing the clustering behavior of vacancy. The growth of vacancy clusters in pure vanadium can be explained using the crystal growth model of terrace-ledge-kink, and their formation energy and binding energy can be predicted using the Wigner-Seitz area. In vanadium alloys, Ti can reduce the vacancy formation energy and stabilize vacancy cluster energetically, and the Ti-vacancy clusters are more stable than alone vacancy clusters over 30 %, corresponding to the smaller saturated size. Kinetically, vacancy diffusing towards Ti shows a lower barrier of 0.04 eV than 0.39 eV in vanadium, and the barriers of vacancy migrating towards Ti-vacancy complexes decrease gradually and are even close to 0 eV during clustering. The migration barriers of vacancy dissociating from Ti-vacancy complexes are relatively higher, and Ti changes the migration mechanism of di-vacancy from one-step to two-step. Further, Ti significantly reduces the effective diffusivity of vacancy, and the effective diffusivity decreases with Ti concentration. The present results show that Ti additions provide more nucleation sites of vacancies, forming large numbers of stable small-size complexes and hindering vacancy diffusion, and eventually suppress the formation and growth of large vacancy clusters/voids.

Tentative

P3-153

Study of the effect of self-ions on the microstructural evolution of irradiated Fe-Cr alloys

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Ion irradiation is commonly used to study the irradiation damage mechanism of materials and to evaluate the irradiation properties of materials. Still, there are some differences in the irradiation element in simulating reactor neutron irradiation. Previous experimental and computational simulations have reported the effects of dose rate, primary knock-on atom (PKA) energies, and impurity ions on the evolution of irradiated microstructure, and elucidated the effects of different irradiation conditions; some recent experimental studies have shown that self-ions may also influence the evolution of the irradiated microstructure, especially on the suppression of the growth of solute clusters; However, the mechanism of self-ions effect on different irradiated microstructure is still poorly understood and deserves further study. In this work, an Object Kinetic Monte Carlo modeling of defect evolution in an iron-chromium alloy (model of low activation ferrite martensitic steel) was implemented. The trends of self-ions on the density and average size of different types of matrix defect clusters with different dose and dose rate are investigated. The mechanism of the influence of self-ion irradiation on the formation and growth of solute clusters is also explored. The results suggest that the effect of self-ions on the morphology of matrix defects increases with the increasing irradiation dose and dose rate, and the effect is significant above about 1 dpa. The present work clarifies the mechanism of the influence of self-ions on the irradiation effect and provides a scientific reference for improving the experimental scheme of ion irradiation in simulating neutron irradiation.

Tentative

P3-154

Machine Learning Interatomic Potential for Atomistic Simulations of Nanostructure Formation on He-Irradiated Tungsten Surfaces

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When high-flux helium (He) ions bombard tungsten (W), a candidate divertor material, the nanostructures called fuzz are formed. Extensive experimental studies have successfully identified the fuzz formation conditions. However, the formation mechanism has not been fully uncovered. In this study, we investigate the initial stage of fuzz formation using molecular dynamics (MD) simulations, which can provide atomistic insights into a complex phenomenon. The reliability of MD simulations is predominantly determined by the accuracy of the potential model used. Although several empirical potentials have been reported for W-He interactions, there is no potential model that can accurately describe all the complex structures that can occur during fuzz formation. To simulate the entire process beginning from He irradiation to surface morphology changes, we developed a machine-learning moment tensor potential (MTP). We prepared the training dataset using first-principles density functional theory (DFT) calculations for structures characteristic of fuzz formation, such as adatoms, concavities, columnar nanowires, He bubbles, and W nanoclusters.

To validate its performance, we compared the elastic properties, the stability and mobility of He atoms in W bulk, and defect formation energies calculated by MTP with experiments, DFT calculations, and empirical potential models. Most properties are in good agreement with available experimental and DFT results. We also confirmed good performance for V-He complex binding energies and adatom migration barriers. Finally, we conducted preliminary MD simulations for He irradiation on W surfaces and examined the results. We anticipate that this potential model will enable accurate atomistic simulations of fuzz formation.

Tentative

P3-155

Small punch test of deformation behavior of ultra-thin steel specimens irradiated with high-energy heavy ions

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In the present work, small punch tests of 3-mm disks of Fe-base alloys with thickness less than 0.25 mm specific for irradiation with high-energy ions were performed. Effects of the variation of specimen thickness (from 0.25 down to 0.1 mm) on the load-displacement character in SP test were studied. A finite-element-method (FEM) model for the thin disks of the Fe-base alloys was developed. Correlation between the small punch test and the tensile test was discussed based on the FEM analysis. The position of the maximum shear stress at the periphery of the bulge from FEM calculation matches well with the position of the observed circumference cracks on the SP-tested disks. It is suggested that, with the decrease of specimen thickness down to 0.1 mm, the displacement to fracture (δf) is close to the point of the initiation of cracks at the periphery of the bulge, and there is less difference in stress between the upper and lower surfaces. Based on the FEM analysis, the irradiation experiment of some high-Cr ferritic steels (T91, MA956, a 9Cr-0.1Ti ODS steel) was performed with 122 MeV Ne ions or 357 MeV Ni ions, respectively, to successively increasing doses at various temperatures. Deformation behavior of the irradiated ultra-thin steel specimens was investigated by the small punch test at either room temperature or elevated temperatures. Hardening and ductility loss due to the irradiation were evaluated, and the correlation with the microscale defects observed by TEM is studied.

Tentative

P3-156

Contrasting Irradiation Behavior of Two Phases in Titanium Alloys at Low-Temperature Due To Omega-Phase Precursors in the Beta-Phase Matrix

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Recent high-energy proton and low-energy iron beam irradiation experiments have revealed clear differences in the behavior of alpha(HCP)- and beta(BCC)-phases of Ti alloys with respect to radiation damage at low temperatures. Compared to the conventional dual-phase Ti-6Al-4V, the Ti-15V-3Cr-3Sn-3Al (Ti-15-3) alloy exhibits superior radiation damage tolerance in both solution treated (100% metastable beta-phase) and solution treated and aged (mother beta-matrix + precipitated alpha-phase 50% each) states. It should be emphasized that the beta-phase may have much superior radiation damage resistance than the alpha-phase, which can be caused either by strong sink effects expected from the homogeneous lattice disorder present at sub-nanometer sizes, or by anomalous point defect recombination due to the high mobility of vacancies. Both are considered to originate from metastable omega-phase precursors (athermal omega-phase) that are specifically formed in the beta(BCC) phase of group-4 transition metals. Two-step aging treated Ti-15-3 shows only slight hardening at least up to 10 dpa under ion irradiation at 300 degree-C. It is a prime candidate for the next generation high intensity accelerator beam window materials, and intensive research and development is underway for its production. These results obtained with Ti alloys can be beneficial to nuclear reactors and fusion reactors in many ways, including the elucidation of fundamental radiation damage mechanisms in nanoscience and the development of new engineering technologies, including optimization of thermo-mechanical processes for Ti alloys to maximize radiation damage resistance at medium temperatures.

Ref.: T.Ishida, S.Kano, E.Wakai et al, J. Alloys and Compounds 995 (2024) 174701

Tentative

P3-157

Phase diagram of the Γ2-phase: C15 Laves phase in RPV steel

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The Γ_2-phase Ni_3SiMn_2 is one possible configuration of irradiation-induced Ni-Si-Mn precipitates (a.k.a. the late blooming phase) in the rector pressure vessel (RPV) steel of light water reactors. Unlike the other possible configurations such as the G-phase $Ni_{16}Si_7Mn_6$, this intermetallic compound is crystallographically characterized as the C15 Laves phase, which is of particular interest in fundamental irradiation damage study, specifically in relation to nucleation of interstitial-type dislocation loops. Although the dislocation loop is a planar, 2D cluster, recent simulation studies have suggested that its nucleus is not 2D but 3D, crystallographucally identical to the C15 Laves phase. Hence in the modeling of microstructure evolution in the RPV steel the relation between the Γ_2-phase and the dislocation loop may be a key harnessing irradiation-induced precipitation and dislocation loop formation. We have recently succeeded in fabricating ingots of the Γ_2-phase by means of arc melting, and examined the validity of existing Ni-Si-Mn phase diagrams. It appears that both experimental (Kuz'ma 1964) and simulation (Thermocalc TCNI12) diagrams are not valid. The stoichiometric composition Ni_3SiMn_2 (=$Ni_{50}Si_{17}Mn_{33}$) was found not to be stable. Although existing phase diagrams stipulate that the range of stable non-stoichiometric composition is extended along a fixed Mn concentration (33 at.%Mn), from Ni51Si16Mn33 to $Ni_{56}Si_{11}Mn_{33}$, our experiments have revealed that the range is rather along a fixed Si concentration (14 at.%Si), from $Ni_{43}Si_{14}Mn_{43}$ to $Ni_{53}Si_{14}Mn_{33}$, though thermodynamically the most stable non-stoichiometric composition appears to be their crossing point, $Ni_{53}Si_{14}Mn_{33}$.

Tentative

P3-158

Resolving Cr segregation at grain boundaries in Fe-Cr alloys by Ising-machine accelerated Monte Carlo optimization

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The segregation of Cr at grain boundaries (GBs) critically influences the mechanical properties of high-strength steels, but it challenges the Monte Carlo (MC) optimization based on density-functional theory calculations due to high computational costs. For the 5(310) GB, we have compared the efficiency of MC with the emerging factorization-machine quantum-annealing (FMQA) algorithm using Ising-machines. We find FMQA is more effective in low-concentration scenarios, while MC retains advantages at higher Cr concentrations. By integrating FMQA steps into the MC process, we introduce an FMQA-accelerated MC (FMMC) method, which demonstrates a threefold improvement in predicting low-energy atomic configurations. Utilizing FMMC, we have investigated 3(111) and 21(541), and find Cr segregates at all GBs, but does not precipitate across them. Analyses suggest that spin polarization acts as a driving force for Cr segregation, while the high GB formation energy of pure Cr inhibits Cr atom precipitation at the GBs within Fe-Cr alloys.

Tentative

P3-159

Dynamics study of transformation of a stacking fault tetrahedron to a void in FCC alloy.

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Stacking fault tetrahedra (SFT) are formed in FCC alloys under high energy particle irradiation. Study about the formation of SFT, it is postulated that vacancy voids collapse into Frank loops which then dissociate into SFT. On the other hand, for large SFT, Kojima et al. showed that SFT change their morphology to voids under electron beam irradiation. We have also investigated the shape change of SFT to voids using high voltage transmission electron microscope. We revealed that the defect size of the morphology change from SFT to void is about 7-8 nm, which is smaller than the theoretically determined critical size. And voids are found to be more energetically stable than SFTs above a certain size. In this study, the dynamics of the morphological change from SFT to void was analyzed using molecular dynamics (MD). MD simulations were performed with the large-scale atomic/molecular massively parallel simulator, LAMMPS using the embedded atom method potential. A SFT made into a super cell and vacancies were added into it and the most stable configurations were searched. By repeating this operation, the paths that eventually deform into voids were studied. The details will be shown in the presentation.

Tentative

P3-160

A coupled Dislocation Dynamics and Cluster Dynamics model of irradiation creep and growth

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We develop a coupled Dislocation Dynamics and Cluster Dynamics model of irradiation creep and growth. The model couples spatially dependent reaction-diffusion boundary value problems (BVP) for mobile vacancy and SIA clusters to the evolution of the discrete dislocation network within the crystal, including climb and glide motion of dislocations. The framework is implemented in three-dimensional discrete dislocation dynamics (DDD) simulations within a superposition solution scheme, and it considers the effects of various bias factors including the diffusion anisotropy difference (DAD) of SIA clusters, the dislocation bias, and the production bias of defects from the radiation cascade. The framework is applied to model the high-temperature deformation of irradiated materials, with emphasis on irradiation creep and growth in both fission and fusion conditions. In irradiation growth conditions in Zr, we find that the DAD is the most critical factor influencing the kinetics of the loop evolution, while the recombination/interaction of mobile defects induces a strongly spatial dependence of the loop evolution. The method is also adopted to study the evolution of interstitial <a> and vacancy <c> dislocation loop ensembles in Zr. Our findings reveal the spatial dependence of the growth, and it determines a specific range for the anisotropy factor of SIA clusters to reproduce the co-growth of <a> and <c> loops. Finally, we verify the model in thermal creep conditions for RAFM steels, including a realistic polycrystalline microstructure, and we predict the irradiation creep behavior in fusion conditions.

Tentative

P3-161

Assessing the long-term performance of the ARC divertor

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Plasma facing components of future fusion systems will be expected to withstand years of operations. However, the long-term exposure to these extreme environments will alter the material properties and therefore performance of the components. Using an integrated plasma-material interaction model that includes descriptions for edge plasma, ion-surface interactions and the sub-surface gas dynamics, we assess the evolution of ten locations across the tungsten divertor target of ARC during ten pulses of deuterium-tritium (D-T) plasma exposure. Our simulations show no erosion due to the extremely low ion energies in these detached conditions, even with neon in the plasma. The low impact energies also lead to constant reflection coefficient, and therefore the D-T content accumulation is primarily driven by particle flux, as more D-T is implanted, and continues to increase with each pulse. The gases diffuse deeper with increasing the substrate temperature that is caused by heat fluxes. We intend to continue the simulations until D-T reach the bulk-boundary to explore whether and when the system may reach steady-state conditions. Finally, these results (depth resolved D-T concentrations and temperature profiles) will be used as input for modeling the crystal plasticity evolution of the tungsten and obtain the polycrystal evolution and grain growth dynamics under in-service conditions, from which critical stresses, temperatures, and doses can be extracted for use in device design optimization.

Tentative

P3-162

High Mn austenite steel revisited: novel alloy design concepts and material properties

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The variety of design proposals put forward by fusion startups represents a great opportunity even for materials scientists. In the current national projects including Japan, the mainstream of blanket structural materials is reduced activation ferritic steels. However, there are still some unknowns regarding the effects of ferromagnetic materials on plasma confinement and the material strength and fracture characteristics of ferromagnetic materials under strong magnetic fields. In order to mitigate such risks, until the 1980s, development of Fe-Cr-Mn-based reduced activation austenitic steels, in which the Ni in SUS316 was replaced with Mn was also underway, but research became minor from the perspective of phase stability and environmental /irradiation resistance. Recently, our group has shown that high-Mn austenitic steel with Al addition has excellent corrosion resistance to liquid metals [Wang et al. Corrosion Science, 175 (2020) 108864, 209 (2022) 110818.], and that oxide dispersion strengthening (ODS) can enhance their mechanical properties and oxidation resistance [Geng, Nuclear Materials and Energy 38 (2024) 101572.]. We consider that these novel high-Mn austenite steels are suitable for the helical-type fusion reactor proposed by Helical Fusion Co. Ltd., which has a complex magnetic field structure and advocates the liquid metal divertor concept. The chemical composition of materials is Fe-30Mn-5Al-1Si with and without Y_2O_3, leading to lower density (~90%) than SUS316. The tensile yield strength ~180 MPa of non-ODS alloy significantly increased up to 1268 MPa by ODS. In this conference, the latest results, such as materials properties, irradiation effects, and environmental effects will be presented.

Tentative

P3-163

Materials development for the ARC fusion power plant

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[1]Commonwealth Fusion Systems

Commonwealth Fusion Systems (CFS) will deploy the first ARC fusion power plant in Virginia in the early 2030s. ARC consists of a compact tokamak featuring high temperature superconducting (HTS) magnets, a molten salt breeding blanket and primary coolant, and a replaceable integrated vacuum vessel. Delivering ARC will require rapid maturation of materials and processes for operation in a range of extreme environmental conditions including high temperatures, neutron and heat fluxes in vacuum vessel components; deep cryogenic operation of magnet structures under cyclic loading and high fields; lifetime HTS performance under shielded neutron flux; and a variety of FLiBe molten salt wetted surfaces. Some components, particularly those exposed to the highest neutron fluxes, are designed to be replaceable to extend the overall asset lifetime. Others, such as the HTS and structures of superconducting magnets, must last the life of the plant. To address these challenges and others, CFS has established a materials and processing development program focusing on validating material performance in expected operational regimes where possible, conducting rapid emulation testing where necessary (ie. multi-beam ion irradiations in place of direct neutron exposure), and designing new material solutions where required. To enable early 2030s deployment, CFS is deeply integrated with current and emerging supply chains to be able to provide industrial quantities of power plant materials. The overall ARC materials program roadmap as well as the tools and capabilities required to meet this need will be described along with examples of success this playbook has enabled during SPARC deployment.

Tentative

P3-164

An overview of the STEP programme, materials strategy, and materials R&D activities.

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The UK's Spherical Tokamak for Energy Production (STEP) programme has recently completed its conceptual design phase. It is now moving into detailed engineering design with increased emphasis on technology development and validation. An overview of the status of the STEP programme will be presented, with special focus on the STEP materials strategy, materials selection in key systems, and critical R&D activities to progress the material readiness levels across the powerplant.

Tentative

P3-165

Development of advanced manufacturing technologies to produce large-scale components to enhance the competitiveness of the Spanish industrial sector in Fusion(RODAS)

REBECA HERNANDEZ PASCUAL[1], Marta Serrano Garcia[1], Edgar Leon-Gutierrez[1], Nerea Ordas Mur[2]

[1]CIEMAT, [2]CEIT-BRTA

The "RODAS" project, part of the Public-Private Partnership (PPP) Spanish National Research Initiative "Trasminisones," focuses on the development of advanced manufacturing technologies for large-scale components aimed at enhancing the competitiveness of the Spanish industrial sector in fusion energy. RODAS collaboration framework brings together key players from public research institutions, technological centers, and private enterprises to explore and implement additive manufacturing techniques for the production of components intended for fusion reactors. The main objective is to address the challenges associated with producing large, complex, and high-performance components that are crucial for the successful operation of fusion reactors. The project has defined 6 activities that will organize in a coordinated way the research for the development of design and simulation tools, obtaining raw materials specifically designed for fusion applications, advanced manufacturing processes for metals and development of advanced technologies and manufacturing and characterization equipment that allow to advance in the manufacture of components, in an efficient, sustainable and scalable way. Specifically, we will work on different approaches that will culminate in 3 representative components that will validate the RODAS developments: a representative section of a ITER First Wall Panel(FWP) component, a representative component of the first wall of a Breeding Blanket and a representative component of a Test Blanket Module(TBM). The RODAS project, therefore, is articulated around 3 fundamental pillars: Materials, Advanced Manufacturing and Advanced Testing. These innovations are expected to contribute to the advancement of fusion energy research in Spain and globally.This work is completed with the participation of “RODAS Team”

Tentative

P3-166

Design and Development of an Optical Gas Sensor for the ITER DRGA System

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Fast and accurate neutral gas measurement systems are crucial for future deuterium-tritium (D-T) fusion reactors, particularly for the closed-loop fueling cycle. These systems monitor exhaust fuel gases, mainly isotopes of hydrogen (H) and helium (He), in real time.

Quadrupole mass spectrometry (QMS) is typically used to measure gas partial pressures, but the similar mass-to-charge ratios of fusion gas species make this challenging. An alternative method, remote optical emission spectroscopy (ROES), measures gas concentrations based on their light emissions. However, ROES has challenges in fusion applications. While D and ^{4}He emissions are separated by >10 nm, isotopic emissions of He and H are closely spaced, requiring high-resolution optical spectroscopy. Additionally, the sensor must function in reactor fringing fields over 0.2T while maintaining a stable plasma within the sensor. Ionizing radiation from the fusion reactor can further complicate the sensor operation.

In this paper we present the development and design of a ROES sensor for the ITER DRGA system. The sensor is qualified to detect small (<0.1%) concentrations of H and He gas isotopes with a speed of response of less than 1 second. The optimization of the sensor's plasma for resolving closely spaced emissions from D + T and ^{3}He + ^{4}He gas mixtures is discussed. Finally, the stability of the sensor's operation in a representative fusion environment is discussed. Experimental results of sensor operation during exposure to magnetic fringing fields up to 0.3 T), gamma radiation (up to 500 kGy) and neutron bombardment up to (1E14n/cm^2) will be presented.

Tentative

P3-167

Performance Qualification of a Vacuum Pump Train with Modified Scroll Pump

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[1]United Kingdom Atomic Energy Authority, [2]Edwards Ltd

The Hydrogen-3 Advanced Technology (H3AT) facility will consist of a closed loop tritium fuel cycle simulation plant with a tritium inventory of up to 100g. The facility will have the capability to store, deliver and purify large quantities of tritium quickly, along with the necessary infrastructure and safety systems for safe operation and proper disposal of tritium waste. The vacuum pump train in most tritium operations within the fusion industry typically consists of a metal bellows, scroll, and a turbo molecular pump. This set-up enables efficient pumping from atmospheric to vacuum. Historically, the now-obsolete Normatex scroll pump with a metal tip seal demonstrated exceptional performance in this role, creating a significant challenge for the fusion industry to find an adequate replacement. Standard scroll pumps with polymer tip seals are incompatible with tritium, limiting available options. To address this challenge, the H3AT facility developed a test rig to evaluate the performance of the modified Edwards scroll pump in tritium operations, aiming to match or surpass the Normatex pump performance. This presentation will detail the testing methodology, challenges, and results, focusing on pump-down time, noise, vibration, pumping rates, and overall efficiency. It will also assess the scroll pump's compatibility and durability in a tritium environment. The findings aim to provide a clear pathway for adopting the Edwards modified scroll pump as a robust and reliable solution for vacuum pumping in tritium operations, addressing a critical gap in the fusion industry. Keywords: Scroll Pump, Vacuum Pump, Normatex

Invited

P3-168

A grain boundary engineering approach to enhance intergranular damage resistance in reduced activation ferritic martensitic steel

Ming Shang Chen[1], Lei Peng[1], he Ren Zhou[1], yi Jing Shi[1], jun Cheng Zhu[1], qiang Zhi Fan[1]

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Reduced activation ferritic/martensitic (RAFM) steels have been considered as candidate structural materials for Generation Ⅳ nuclear reactors. In this work, grain boundary engineering (GBE) method was used to improve the corrosion resistance of RAFM steel CLF-1. The thermo-mechanical treatments (TMTs), including cold-rolled to reduction ratios from 0% to 20%, followed by austenited at 1323 K for 3.6 ks, and then were subjected to tempering at 1053 K for 7.2 ks were used to as-received (AR) specimen. The effect of TMTs on prior austenite grain (PAG) size and prior austenite grain boundary character distribution (GBCD) were explored. The results show that there exists a critical deformation, 10% in this work. While the reduction ratio was less than 10%, PAG size, the fraction of Σ3n boundaries along PAGBs, and the fraction of coincidence site lattice boundaries (CSLBs, $3 \leqq \Sigma \leqq 29$) along PAGBs increased with increasing reduction ratio due to strain induced boundary migration (SIBM), otherwise, PAG size, the fraction of Σ3n boundaries along PAGBs, and the fraction of coincidence site lattice boundaries (CSLBs, $3 \leqq \Sigma \leqq 29$) along PAGBs decreased with increasing reduction ratio due to recrystallization. The PAG size is positively correlated with the fraction of Σ3n boundaries along PAGBs and the fraction of CSLBs along PAGBs. Steam oxidation tests revealed that the oxide scale thickness was negatively with the fraction of Σ3n boundaries along PAGBs and the fraction of CSLBs along PAGBs, indicating that GBE method could improve corrosion resistance of CLF-1 steel effectively.

Invitation

P3-169

Amorphous FeCrAlTiSi high entropy alloy coating prepared using vacuum ion plating

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In this work, FeCrAlTiSi high-entropy alloy coatings (HEACs) were prepared using magnetron sputtering (MS) under various substrate biases with a Ti interlayer pre-deposited through multiarc ion plating (M-AIP) to enhance the coating/substrate adhesion strength. The morphology, microstructure, elemental and phase composition, nanohardness, Young's modulus, and corrosion resistance were characterized using SEM, XRD, TEM, XPS, nanoindenter, and electrochemical workstations. Results indicated that all FeCrAlTiSi coatings were in an amorphous state. Nanohardness and Young's modulus had a strong substrate bias dependence, reaching the maximum of 16.04 GPa and 241.3 GPa, respectively, at -100 V bias. Meanwhile, FeCrAlSiTi coatings at -100 V exhibited superior corrosion resistance to that of 316L in 3.5 wt.% NaCl aqueous solution, about 60 times improvement in terms of corrosion current density owing to the combined effect of homogeneous composition distribution, amorphous structure, and chemically stable oxide film predominantly composed of Al_2O_3, TiO_2 and SiO_2.

Tentative

P3-170 "Presenting Author: Min Su Ha"

Geometric signals of PAUT in various TIG welding configurations of 316L(N)-IG

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ITER VV has mainly applied three kinds of welding configuration, i.e. Full penetration T-joint (Type III.1 welded joint in RCC-MR 2007), Full penetration Butt-joint (Type II.1) and Inclined full penetration Butt-joint (Inclined II.1), according to its box-type structure design. As a first-of-a-kind technical verification of NDT feasibility, the qualification of PAUT procedures was key challenge. ITER grade austenitic stainless steel, 316L(N)-IG, was used for the fabrication of a total 170 qualification blocks with more than 400 artificial reflectors, which have been agreed by ITER as being all the postulated defects in the volume of weld zone. However, due to high attenuation and dispersion of the ultrasonic signal in thick weld of the austenitic stainless steel, several false PAUT signals were generated by metallurgical and weld-geometry reflectors in the defect-free weld zone. This paper presents the various geometric signals from type III.1, II.1 and inclined II.1 joints. In case of type III.1, the probe wedges and corners of weld throats have generated strong geometric indications in PAUT scan data. The type II.1 joint showed strong echoes from probe wedges and weld root penetration in most cases of PAUT scan techniques. The inclined II.1 joint generated more cases of false signals in PAUT data due to the complex weld geometry. Several types of narrow oval-shaped signal were collected in high amplitude at high angle of sectorial scan images. Also, several scan techniques showed a long tail indication connected to the indication of root penetration along the line with the same time base.

Preview

P3-171

Mechanical Property Assessment of FM and ODS Steels Before and After Irradiation

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Ferritic-martensitic (FM) steels are prime candidates for fusion structural materials due to their excellent radiation resistance and thermal stability, while oxide dispersion-strengthened (ODS) ferritic steels offer enhanced strength, creep resistance, and damage tolerance, making them more suitable for high-temperature fusion reactor applications. Evaluating the mechanical properties of these steels before and after irradiation is crucial for assessing their performance in service. This study compiles mechanical property data for various FM and ODS steels, focusing on their high-temperature performance and its dependence on processing conditions. It also examines the effects of radiation and temperature on strength and toughness values that define their performance limits in fusion reactors. Particularly, a comparative assessment of Fe-Cr-based FM and ODS alloys is conducted using the temperature and strength dependencies of fracture toughness to evaluate their high-temperature structural behavior. Fracture toughness vs. yield stress plots confirm an inverse relationship between the two parameters. Some nanostructured ODS steels exhibited fracture toughness values outside the expected range for a given strength, suggesting significant changes in deformation and fracture mechanisms, whereas FM steels showed less strength dependence. Notably, at temperatures ≧ 600 °C, most ODS and FM steels lost their nanostructure advantage in high-temperature deformation and cracking resistance due to embrittlement and/or strength reduction. Nanostructured ODS steels generally demonstrated superior radiation resistance compared to FM steels, exhibiting lower radiation-induced swelling, finer helium-bubble formation, reduced irradiation creep, and less low-temperature embrittlement, although only a select few processing routes can achieve the desirable performance required for high-temperature applications.

Tentative

P3-172

A Material Production and Qualification Plan for lead-lithium eutectic Li(6) enriched nuclear grades

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Basis for a complete Material Production and Qualification Plan (MPQP) for nuclear grades of lead-lithium eutectics is discussed. Production Plan includes:
(1) the set of raw materials (Pb, Li) and LLE products handling and qualification established procedures;
(2) the possible LLE alloying routes according to technical feasibility and industrial reliability;
(3) the possible alloying routes for experimental demonstration at laboratory to be scale-up by design and demonstrated at required industrial scale.
The provided Qualification Plan provides:
(1) the specification of the material characteristics both constitutive (e.g. Li title, short-scale homogenization) and compositional (e.g.: allowable impurity limits);
(2) the material database specification inputs and allowable uncertainties according to the material uses and design functionalities;
(3) the material certification procedures, as the set of suitable experimental procedures to certify database properties.
The two main goals are:
(1) - the standardization by the generation and establishment of Norms and Standards (e.g. as UNE or ISO) in the production and qualification of lots of LLE(6) nuclear material grades;
(2) - The establishment and management of a specific network of Services for LLE(6) characterization and material standardized qualification.

Tentative

P3-173

Continuous Production of Advanced Ceramic Breeder Pebbles

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Advanced Ceramic Breeder (ACB) pebbles are regarded as the EU reference material for solid tritium breeding in the ITER TBM and solid breeder blanket concepts for DEMO. These pebbles and their production process, KALOS, have been developed at the Karlsruhe Institute of Technology to meet the wide range of requirements for use in a fusion reactor, such as a high lithium density, strong mechanical strength and good tritium release characteristics. Furthermore, in order to be able to provide the required amounts for a breeding blanket, it is essential that the process has the capacity to produce at this scale with consistent and predictable product quality.

The melt-based KALOS process offers many advantages, including a close control of the pebble sizes, the ability to reprocess used material and the scalability of the process. The facility has recently undergone a significant upgrade including a conversion from a batch process, limited by the volume of the melting crucible, to one capable of continuously melting synthesis powders and producing pebbles. The theoretical maximum capacity has been increased to 10 kg in a working day. This work presents the first results from pebbles produced using the new process configuration and confirms the reliability of the process and the quality of the pebbles.

Tentative

P3-174

Hardening of T91steel induced by self-ion irradiation after He implantation at 450 °C

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Ferritic/martensitic (F/M) steel is regarded as one of promising candidate structural materials for fusion reactor due to their favorable mechanical properties and radiation resistance. In order to understand the synergistic effect of He and displacement damage on evolution of Mechanical properties in fusion environments, self-ion irradiation after He implantation at 450 °C was carried out on T91 specimens. After irradiation, the hardening behavior of T91 steel was analyzed by using nano-indentation technology with Nano Indentor G200 in a mode of continuous stiffness measurement (CSM) and the distributions of He bubbles in the samples are shown in the TEM results. Significant irradiation hardening occurred in all the irradiated samples. As the increasing of implanted He ion concentration, the increment of hardness raised. Due to He-ion implantation, amount of He bubbles formed. As the increasing of implanted He concentration, the density of He bubbles significantly increased. When more He ions implanted into samples, the density of dislocation loops had slightly increased, and the mean size of dislocation loops firstly increased and then droped. The displacement defects induced by Fe ion can be the effective nucleation center which promote the growth of the He bubbles.

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