19-20 October 2023
Parque de las Ciencias, Granada, Spain

The 21st International Workshop on Ceramic Breeder Blanket Interactions (CBBI-21)

Organized by

GOBIERNO DE ESPAÑA MINISTERIO DE CIENCIA E INNOVACIÓN

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PROCEEDINGS OF
THE 21st INTERNATIONAL WORKSHOP ON CERAMIC BREEDER BLANKET INTERACTIONS (CBBI_21)

19-20 October 2023
Granada, Spain

Dr. María González
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WELCOME to the biannual event that brings together the community of ceramic breeder scientists and technologists: the **21st International Workshop on the Ceramic Breeder Blanket Interactions (CBBI-21)**.

Organized under the auspices of the *IEA Implementing Agreement on the Nuclear Technology of Fusion Reactors*, and in conjunction with the ***International Conference on Fusion Materials (ICFRM-21)***, the **CBBI-21** will be held from **19th to 20th October 2023**, in the beautiful and historical city of **Granada, Spain**.

In the context of the CBBI workshop, researchers, engineers and technologists, involved in the development of the fusion breeding-blanket solid concept, meet to exchange the latest progress in the design, behaviour, testing and modelling of materials and components based on lithium-ceramics.

Closer to the end of ITER, the **CBBI-21** should be the stage to show updates of TBMs development in those countries involved, and a debate forum for its exploitation towards DEMO-like fusion power plants.

The proximity of IFMIF-DONES Granada, as a large materials' testing facility, will allow our community to know first-hand how to be prepared for validating models and testing the behaviour of ceramic breeders with fusion neutrons.

As in previous editions, the **CBBI-21** should serve as a focal point for technical discussions and information exchange improvement of the ceramic breeder blanket as the system for tritium fuel production and energy extraction in a fusion energy reactor.

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The 21st International Workshop on

Ceramic Breeder Blanket Interactions (CBBI-21)

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		THURSDAY, 19th October 2023	
	9:00	***WORKSHOP OPENING SESSION***	
		Ceramic pebble production and properties. Chaired by T. Chikada	
1	9:20	***INVITED***. How to define a specification for ceramic breeder pebbles?	Regina Knitter, Milan Zmitko
2	9:40	ACB Pebble Production: Increasing the Capacity of the KALOS Process in Time for ITER	Oliver Leys, Julia Leys, Regina Knitter
3	10:00	R&D progress of the tritium breeding functional materials and pebble bed technology for the solid breeding blanket at SWIP	Baoping Gong, Juemin Yan, Hao Cheng, Long Wang, Long Zhang, Xiaoyu Wang, Yongjin Feng
4	10:20	System design to fabrication the Core-shell type tritium breeder	Young Ah Park, Young Soo Yoon, Yi-Hyun Park
5	10:40	Long-term annealing performance of Li-6 enriched biphasic Li_4SiO_4/Li_2TiO_3 pebbles	Julia Leys, Christina Odemer, Oliver Leys, Regina Knitter
	11:00	***COFFEE BREAK***	
		Ceramic pebble production and Irradiation behaviour (I). Chaired by R. Knitter	
6	11:20	***INVITED***: Status of lithium ceramic breeder materials development, characterizations and R&D activities	Paritosh Chaudhuri, Harsh Patel, Chirag Sedani, Maulick Pancha, Aroh Shrivastava
7	11:40	Development of Lithium Titanate Ceramic Pebbles By Freeze Granulation and Freeze Drying Method	Aroh Shrivastava, Paritosh Chaudhuri
8	12:00	Fabrication and characteristics of Li_2TiO_3 pebbles fabricated by using powder injection molding process	Yi-Hyun Park, Young Ah Park, Young Soo Yoon, Mu-Young Ahn
9	12:20	Results of neutron irradiation experiments with Li_4SiO_4/Li_2TiO_3 at the WWR-K research reactor	Timur Kulsartov, Zhanna Zaurbekova, Regina Knitter, Gunta Kizane, Julia Leys, Asset Shaimerdenov, Saulet Askerbekov, Magzhan Aitkulov, Yevgen Chikhray, Inesh Kenzhina
10	12:40	In-situ neutron irradiation experiment for EU reference ceramic breeder material.	Julia Leys, A. Shaimerdenov, Sh. Gizatulin, T. Kulsartov, Y. Chikhray, I. Kenzhina, M. Ionescu-Bujor, R. Knitter
	13:00	***LUNCH TIME***	
		ITER TBMs and Compatibility issues. Chaired by P. Chaudhuri	
11	14:00	***INVITED***. The HCCP Test Blanket Module: Current Status in Development and Qualification of Ceramic Breeder Material and an Overview of Open Issues	Milan Zmitko, Regina Knitter, Alessandro G. Spagnuolo
12	14:20	Progress of design and analyses of CN HCCB TBM for ITER	Xinghua Wu, Qixiang Cao, Long Zhang, Xiaoyu Wang
13	14:40	ORNL Status and Progress in Research on Ceramic Breeder Blanket Materials	Xiao-Ying Yu, Yutai Katoh, Takaaki Koyanagi, German Samolyuk, and Weicheng Zhong
14	15:00	Development of a novel breeding blanket using the solid-type PbxLiy eutectic alloy for CFETR	Kecheng Jiang, Qiuran Wu, Lei Chen, Songlin Liu
15	15:20	Corrosion of F82H in ceramic breeder pebble bed and its effect on hydrogen permeation	Keisuke Mukai, Shunsuke Kenjo, Naoto Iwamatsu, Bakr Mahmoud, Takumi Chikada, Juro Yagi, Satoshi Konishi
	15:40	***AFTERNOON BREAK***	
16	16:00	***INVITED***: Microstructural change and deuterium permeation of ZrO_2-coated steel exposed to solid tritium breeder pebbles	Takumi Chikada, Wataru Matsuura, Julia Leys, Marcin Rasinski, Suguru Nakano, Jae-Hwan Kim, Taehyun Hwang, Tsuyoshi Hoshino, Masaru Nakamichi
17	16:20	Corrosiveness of solid tritium breeders to RAFM steel considering evaporated lithium compound and irradiation.	Qiang Qi, Chi Wang, Yingchun Zhang, Haishan Zhou, Songlin Liu, Guang-Nan Luo
18	16:40	Microstructure, corrosion behavior, and mechanical properties of ARAA after compatibility test with Li_2TiO_3 pebbles	Yunsong Jung, Yi-Hyun Park, Duck Young Ku, Youngah Park, Mu-Young Ahn, Seungyon Cho
	17:00	***Round table "Ceramic breeding pebbles: production and behaviour". Chaired by M. Zmitko and K. Mukai***	
		DAY's SESSION CLOSING	

The 21st International Workshop on
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		FRIDAY, 20th October 2023	
	T Inventory and Safety issues. Chaired by M. Roldán		
1	**9:00**	***INVITED***. Design update of the European DEMO Helium Cooled Pebble Bed breeding blanket	Guangming Zhou, Anoop Retheesh, Jin Hun Park, Ion Cristescu, Francisco A. Hernández, Christina Köhly
2	**9:20**	Tritium transport model of lithium-based ceramic pebbles in Ecosimpro.	Carlos Moreno, Almudena Rueda, Jenifer Serna, Fernando R. Urgorri
3	**9:40**	Application of tritium transport model on a DEMO breeding blanket system.	Yonghee Lee, Alice Ying, Mu-Young Ahn, Hyung Gon Jin, Seong-bo Moon
4	**10:00**	Study on Hydrogen Isotope Permeation Behavior of High-Entropy Alloys Coatings.	Long Wang, Zhihao Hong, Yongjin Feng, Baoping Gong, Xiaoyu Wang, Long Zhang
5	**10:20**	Modelling transport of fragmentation and dust particles in granular tritium breeder material inside fusion reactors.	Dario Passafiume, Marc Kamlah
6	**10:40**	Development of a system-level code and application to the tritium transport for the water cooled ceramic breeder blanket.	Songlin Liu, Xueli Zhao, Lei Chen
	11:00	***COFFEE BREAK***	
	Pebble Bed Thermomechanics. Chaired by F. Sánchez		
7	**11:20**	***INVITED***: Plans for Breeding Blanket Test Facility in KFE	Mu-Young Ahn, Min Ho Chang, Yoo Lim Cheon, Seungyon Cho, Moses Chung, Hyoseong Gwon, NamIl Her, SeongHee Hong, Hyun Wook Kim, Sungjin Kwon, Jae-Uk Lee, Yonghee Lee, Youngmin Lee, Sungbo Moon, Yi-Hyun Park
8	**11:40**	Effects of vibration conditions, spatial confinement and friction on mixing and segregation characteristics of mixed pebble beds for CFETR WCCB blanket.	Yong Liu, Lei Chen, Cong Wang, Chongyang He, Songlin Liu
9	**12:00**	Thermal, uniaxial compression and flow simulations of packed beds of spherical and elliptical shaped pebbles.	Harsh Patel, Maulik Panchal, Paritosh Chaudhuri
10	**12:20**	A study of purge flow characteristics and effective thermal conductivity of pebble bed: Experiments and Simulation by ANN	Chirag Sedani, Maulik Panchal, Paritosh Chaudhuri
11	**12:40**	Simulation of Mechanical, Thermal, and Flow Characteristics of Pebble Beds for Solid-type Ceramic Breeding Blanket	Youngmin Lee, Dongwoo Sohn, Mu-Young Ahn, Yi-Hyun Park, Seungyon Cho
	13:00	***LUNCH TIME***	
	Ceramic pebble irradiation behaviour (II). Chaired by J. Leys		
12	**14:00**	***INVITED***: The scope of the future IFMIF-DONES facility and how it can be applied in breeder ceramics	Santiago Becerril, in behalf of the IFMIF-DONES España team
13	**14:20**	Fabrication and tritium release properties of advanced tritium breeder: $Li_4(Si,Ti)O_4$ ceramic pebble	Juemin Yan, Baoping Gong, Hao Cheng, Long Zhang, Xiaoyu Wang, Xiaojun Chen, Chengjian Xiao
14	**14:40**	Influence of various radiation types on radiation-induced processes in lithium orthosilicate-based ceramic breeder materials.	Arturs Zarins, Anna Ansone, Mareks Senko, Janis Cipa, Andris Antuzevics, Liga Avotina, Larisa Baumane, Gunta Kizane, Maria Gonzalez, Julia M. Leys, Regina Knitter
15	**15:00**	Secondary Ion Mass Spectrometry (SIMS) as an important tool to track compositional variations from ion-implanted and ion-damaged Advance Ceramic Breeder (ACB) compositions.	Guiomar Delgado, María González
16	**15:20**	TEM studies in support of the high radiation resistance of ion-irradiated advanced ceramic breeders.	Marcelo Roldán, María González, Fernando Sánchez
	15:40	***AFTERNOON BREAK***	
17	**16:00**	Is H-isotope effectively trapped in structural defects of ion-beam damaged ceramic breeders?	María González, Marta Malo, Alejandro Moroño, Arturs Zarins, Gunta Kizane, Marcelo Roldán, Fernando Sánchez
	16:20	***Round table "Updating HCPB BB towards ITER and DEMO". Chaired by G. Zhou and M-Y Ahn***	
		WORKSHOP CLOSING SESSION	

Invited Contributions

How to define a specification for ceramic breeder pebbles?

Regina Knitter [1], Milan Zmitko [2]
[1] Karlsruhe Institute of Technology, Institute for Applied Materials (IAM), Karlsruhe, Germany
[2] Fusion for Energy (F4E), c/ Josep Pla 2, Barcelona, Spain

In preparation for the ceramic breeder material procurement for the ITER-TBM, a preliminary specification for the European advanced ceramic breeder pebbles was defined, reflecting the current status of the material and process development. A specification has to define material properties or features as criteria that can be unambiguously determined and will later be used to decide, if a production lot has to be rejected. This also implies the definition of a quality control and a validation of the measurement methods used.
The contribution will focus on critical issues when faced with the necessity to define a specification for ceramic breeder pebbles. This includes, for example, the uncertainties of the theoretical density, the scattering of crush load values, or the validity of pebble bed densities. Special attention will be given to the aspect of impurities, as these are largely depending on the used raw materials. It is recommended to separately specify the purity level for the non-nuclear and nuclear phases of ITER. As activation is no issue in the non-nuclear phases, a lower purity grade may here result in significant cost savings. In this case it may also be sufficient to specify the purity level of the raw materials to be used. For the nuclear phases of ITER, when the use of enriched material will be required, the impurities of the Li-6 enriched raw material will strongly influence the impurity level of the produced ceramic breeder material. Based on chemical analysis of commercially available enriched material, best and worst case scenarios for impurities are assessed. Yet, the impurity level may change – for the better or worse – if this material is procured from a different supplier or source.

Keywords: Ceramic Breeder, ITER TBM, Impurities, Lithium-6, specification

Status of lithium ceramic breeder materials development, characterization and R&D activities

Paritosh Chaudhuri[1,2], Harsh Patel[1], Chirag Sedani[1], Maulick Panchal[1], Aroh Shrivastava[1]
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[2]Homi Bhabha National Institute, Anushaktinagar, Mumbai – 400 094, India

Abstract
This article provides an overview of efforts made to conduct various characterizations and simulation studies in the development of lithium ceramic breeder materials. As tritium breeding materials, a number of materials have been created and are being tested for dependability and sustainability. Lithium meta-titanate (Li_2TiO_3) is a popular appropriate candidate material for tritium breeders. At the IPR (Institute for Plasma Research, India), Li_2TiO_3 pebbles have been successfully produced using both the extrusion-spheronization method and the freeze-drying method, and each step of their preparation has been meticulously characterised (power, pellets, and pebbles). A thorough investigation of the thermal conductivity of pebble beds was conducted using steady-state and transient methods, as well as mechanical compression and flow simulations of packed beds.
The effective thermal conductivity of the pebble bed has been predicted using artificial neural network modelling and simulations based on the input factors of temperature, gas environment, and packing fraction. By exposing them over an extended period of time and annealing them at various temperatures under purge flow conditions, investigations of the chemical interactions between Li_2TiO_3 and India Specific Reduced Activation Ferritic Matensitic (IN-RAFM) Steel were conducted. We will go over the specifics of the experiments, their outcomes, simulations, and plan for using an accelerator-based neutron irradiation facility for the irradiation study on Li2TiO3 ceramic breeders.

Keywords: lithium titanate, Pebble bed, effective thermal conductivity, simulation

The HCCP Test Blanket Module: Current Status in Development and Qualification of Ceramic Breeder Material and an Overview of Open Issues

Milan Zmitko[1], Regina Knitter[2], Alessandro G. Spagnuolo[3]
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One of the reference tritium Breeder Blanket concepts developed in the Europe that will be tested in ITER machine under the form of Test Blanket Module (TBM) is the Helium-Cooled Ceramic Pebble (HCCP) TBM concept in which lithiated ceramic pebbles are used as a tritium breeder and beryllium/beryllides as a neutron multiplier material. This concept uses the EUROFER97 reduced activation ferritic-martensitic (RAFM) steel as a structural material and pressurized helium for heat extraction (8 MPa, 300-500°C).

The paper gives a brief general description of the HCCP TBM design and the main design requirements, including the requirements for the ceramic breeder material. The ITER HCCP TBM development and qualification plan with identification of the main milestones will be presented, taking into account recently signed EU-KO TBM Partnership agreement between Fusion for Energy and ITER Korea.

The main part of the paper will be devoted to the presentation of the ceramic breeder material development strategy, qualification plan and overview of the current status of R&D activities. The achieved results on the ceramic breeder (CB) pebbles production (KALOS process), the CB pebbles and pebble beds characterization, and performance under TBM/DEMO relevant conditions, including the performance under neutron irradiation, and thermo-mechanical performance will be briefly overviewed and a new neutron irradiation experiment, foreseen for the functional materials (i.e. for the ceramic breeder and beryllium materials), will be introduced.

A special attention will be focused on open issues which are/will be addressed in future R&D activities (e.g. development of the CB/Be pebbles filling procedure, modelling of the CB dust formation/re-suspension and its impact on purge gas flow, impact of the CB on structural material properties).

A Return of eXperience (RoX) of the TBM programme for future DEMO Breeder Blanket related activities will be briefly discussed as an important part of the fusion development roadmap.

Keywords: ITER, Test Blanket Modules (TBM), ceramic breeder, Return of experience (RoX)

Microstructural change and deuterium permeation of ZrO_2-coated steel exposed to solid tritium breeder pebbles

Takumi Chikada[1], Wataru Matsuura[1], Julia Leys[2], Marcin Rasinski[3], Suguru Nakano[4],
Jae-Hwan Kim[4], Taehyun Hwang[4], Tsuyoshi Hoshino[4], Masaru Nakamichi[4]
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Tritium permeation through structural materials is a critical issue in most fusion reactor blanket concepts. Functional coatings have been developed for nearly half a century to mitigate tritium permeation and showed high permeation reduction using ceramic coatings. The research focus is moving to not only the permeation barrier performance but the corrosion resistance to tritium breeders, coolants, and irradiation etc. The number of studies to elucidate the relationship between corrosion by solid breeders and hydrogen isotope permeation is limited. In this study, interactions of coated steels with solid breeder pebbles and their influence on hydrogen isotope permeation are presented.
Zirconium oxide coatings were fabricated by metal organic decomposition on the reduced activation ferric/martensitic steel F82H substrates. Some of the coated samples were annealed with the solid breeder pebbles for up to 32 days at 550 °C under 1200 mbar helium with 0.1 vol% hydrogen. Three kinds of ceramic breeder pebbles were used in this study: Li_2TiO_3, solid solution of $Li_{2+x}TiO_{3+y}$ with 20 wt% Li_2ZrO_3, and Li_2ZrO_3 with 20–30 mol% Li_2TiO_3. Deuterium permeation measurements for the coated samples were conducted with or after exposure to the pebbles with the driving pressure of 10–100 Pa in the temperature range of 300–550 °C. The samples before and after exposure were characterized by scanning electron microscopy with a focused ion beam system and X-ray diffraction.
The deuterium permeation flux of the coated sample with contacting Li_2TiO_3 pebbles was four orders of magnitude lower than that of uncoated steel, indicating the coating did not degrade during the measurements even with exposure to the breeder pebbles. On the other hand, the coatings were delaminated at the pebble-contacting areas and formed Li-Fe-O ternary oxides after the annealing test with the pebbles of Li_2ZrO_3 with 20–30 mol% Li_2TiO_3, resulting in a much lower permeation reduction. These results suggest that the gas atmosphere during pebble exposure, in particular oxygen and water impurities, strongly affect corrosion behavior. In the presentation, the results of round-robin tests using the other pebbles will be included.
Keywords: tritium, permeation, corrosion, ceramic coating, solid breeder

Design update of the European DEMO
Helium Cooled Pebble Bed breeding blanket

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At the end of the Pre-Concept Design Phase (2014-2020) of the European DEMO programme, a DEMO-wide gate review was conducted. Along with other systems, the design and R&D activities of the European DEMO Helium Cooled Pebble Bed (HCPB) breeding blanket were also reviewed by external panels. It was found that there were several outstanding challenges to be solved. One of the critical challenges is the low reliability of blanket due to too many welds which prone to fail under DEMO operating conditions. To tackle this issue, starting from 2021, a new concept is proposed. In which the pressure of the purge gas is equalized with that of the coolant. By doing so, the welds to prevent in-box loss of coolant are eliminated. However, the blanket box has to bear the 8 MPa pressure even under normal conditions. Consequently, the thickness of the critical structures of the HCPB has to be increased, which will impact the tritium breeding ratio. To have a viable design, the basic nuclear, thermal hydraulic and structural requirements need to be met.
In this invited talk, the design status of the European DEMO HCPB breeding blanket and its tritium extraction and recovery system will be presented, together with the related nuclear, thermal hydraulic and structural analyses.

Keywords: European DEMO, Helium Cooled Pebble Bed, Breeding Blanket, Design

Plans for Breeding Blanket Test Facility in KFE

Mu-Young Ahn[1]*, Min Ho Chang[1], Yoo Lim Cheon[1], Seungyon Cho[1], Moses Chung[2], Hyoseong Gwon[1], NamIl Her[1], SeongHee Hong[1], Hyun Wook Kim[1], Sungjin Kwon[1], Jae-Uk Lee[1], Yonghee Lee[1], Youngmin Lee[1], Sungbo Moon[1], Yi-Hyun Park[1]

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Demonstration Reactor (DEMO) is considered a next-step beyond ITER, where it is expected that electricity is produced by fusion power in a sustainable and (quasi-)continuous way. One of the key components to realize DEMO and fusion power reactors is a breeding blanket that enables fuel self-sufficiency and converting fusion power to thermal power. Although breeding blanket mockups will be tested in ITER through Test Blanket Module (TBM) program, in particular in order to demonstrate the feasibility of the breeding blanket concepts and validate design tools and database in a fusion environment, it has been pointed out that there are significant technical gaps between TBMs and DEMO blankets. Therefore, to tackle the technical gaps, various R&D activities and facilities for those have been proposed and constructed around the world.

Korea Institute of Fusion Energy (KFE) is conducting a pre-conceptual study on the test complex which includes an integrated breeding test facility and a blanket system test facility, with collaborators. The integrated breeding test facility, whose primary purpose is for the blanket component test, is based on a 40 MeV deuteron accelerator-driven system with maximum 10 mA for fusion-like neutron generation and testing, possibly, one-to-one scale breeding unit of DEMO blankets. In particular, it is anticipated that the facility can run in continuous-wave operation to simulate DEMO operating conditions. The blanket system test facility is to demonstrate the reliability and safety of the blanket and its ancillary systems for DEMO-relevant long-term operation in a non-nuclear environment. In this study, overall plans for the test complex and current design status are addressed.

Keywords: breeding blanket, blanket component test, breeding test, DEMO, TBM

The scope of the future IFMIF-DONES facility and how it can be applied in breeder ceramics

Santiago Becerril, in behalf of the IFMIF-DONES España team

Oral Contributions

ACB Pebble Production: Increasing the Capacity of the KALOS Process in Time for ITER

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Advanced ceramic breeder (ACB) pebbles are regarded as the EU reference solid material for tritium breeding. The lithium-rich pebbles, composed of lithium orthosilicate with a strengthening phase of lithium metatitanate, will be used to fill breeder blankets for both the ITER TBM as well as the DEMO HCPB. Presently it is foreseen to start the ACB material procurement for the first ITER-TBM in 2026 and it is essential that it is possible to produce and supply the required amounts of high quality pebbles on time. Approximately 90 kg will be required to fill the blanket for ITER.

The melt-based KALOS (KArlsruhe Lithium OrthoSilicate) process was developed at the Karlsruhe Institute of Technology for the production of ACB pebbles. The process is currently undergoing a significant upgrade with the main goal to increase the production capacity, as well as offering more reliability, a more advanced control of process parameters and to integrate multiple sub-systems into a centralised control system. Examples of new processing equipment include a custom-made furnace with extra heating power, a new larger processing crucible and an extended cooling tower.

The status of the upgrade, as well as results from the first production runs will be presented. This will include the influence of the cooling parameters and studies of the droplet generation at the nozzle using a high-speed camera set-up. Furthermore, plans for the next step of the upgrade, which will see the conversion of the process to a semi-continuous operating mode, will be presented.

Keywords: advanced ceramic breeder, pebble production, KALOS process, lithium orthosilicate, lithium metatitanate

R&D progress of the tritium breeding functional materials and pebble bed technology for the solid breeding blanket at SWIP

Baoping Gong*, Juemin Yan, Hao Cheng, Long Wang, Long Zhang, Xiaoyu Wang, Yongjin Feng
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Solid lithium-based ceramics breeding blanket is considered as one of the most promising tritium breeding blankets of fusion reactor and worldwide efforts have been devoted to its R&D. In China, the Helium Cooled Ceramic Breeder (HCCB) with the pebble bed concept was selected both in the ITER Test Blanket Module and the CFETR blanket. In HCCB blanket, functional materials play an important role in tritium breeding and tritium extraction. Lithium based ceramics are selected as tritium breeding materials, Beryllium and beryllide are the candidate neutron multiplier. Come research works, such as materials and pebble fabrication, characterization and its' pebble bed related experiment, are carried out in Southwestern Institute of Physics (SWIP).

For the functional materials, the Li_4SiO_4 pebbles were developed and fabricated by the new developed facility based on the melting spray method. Both the chemical and physical properties of the Li_4SiO_4 pebbles were characterized experimentally. In order to validate the production scale and performance stability, the engineering qualification of the Li_4SiO_4 pebbles is under way both in SWIP and CAEP. To improve the properties of the breeder materials, a series of new advanced tritium breeders has been developed, such as, cellular structures Li_4SiO_4 ceramic by 3D print, Li_{2+x}TiO3, and so on. In further, the scale fabrication technology of beryllium pebbles was achieved by improving the REP facility. 10 kg pebbles can be produced in each batch. Also, the engineering qualification of the Be pebbles is under way in SWIP to validate the production scale and performance stability. In further, the helium effects in beryllium CN-G01 were investigated by helium ions implantation experiment. The beryllium samples were implanted by helium ions to fluences of 5.0×10^{16} ions/cm^2 to 1.0×10^{18} ions/cm^2 at room temperature. And the irradiation induced hardening of beryllium increased with increasing fluence of 5.0×10^{16} ions/cm^2 to 1.0×10^{17} ions/cm^2. In addition, for the pebble bed technology, the thermal mechanical properties of Li_4SiO_4 pebbles and pebble beds were investigated. The thermal expansion and crush load of Li_4SiO_4 pebbles at high temperature were obtained experimentally. The results obtained in these experiments and future plans will be reviewed and discussed.

Keywords: functional materials, tritium breeder, neutron multiplier, pebble bed, engineering qualification,

System design to fabrication the Core-shell type tritium breeder

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Lithium ceramic materials currently being studied as tritium breeder include Li_4SiO_4, Li_2TiO_3, Li_2O, etc. Among the candidate materials, Li_4SiO_4 and Li_2O have the highest lithium density but are difficult to handle because they are very unstable in the atmospheric environment. On the other hand, Li_2TiO_3 has a lower lithium density than the two materials mentioned above, but has excellent atmospheric stability and thermal stability, enabling stable production of tritium at an operating temperature. In order to increase the efficiency of tritium production, research is needed to produce pebbles with improved lithium density.

In this study, in order to compensate for the complementary strengths and weaknesses of the materials, a system design was attempted to manufacture pebble with a core-shell structure from pebble, a solid tritium breeder. Core-shell pebbles were fabricated through a fluidic system based on the cross-linking reaction between calcium chloride ($CaCl_2$) and sodium alginate (NaAlg). Li_4SiO_4 and Li_2TiO_3 were utilized to fabricate the core-shell pebble, and Li_4SiO_4 made the core pebble, Li_4SiO_4@Li_2TiO_3 pebble. After sintering the prepared Li_4SiO_4@Li_2TiO_3 green pebbles, the morphology, lithium density and mechanical properties of the Li_4SiO_4@Li_2TiO_3 pebbles were analyzed.

As a result, it was confirmed that Li_4SiO_4@Li_2TiO_3 pebble of core-shell structure was fabricated. It was confirmed that the lithium density was improved compared to a single Li_2TiO_3 pebble, and the atmospheric stability was improved compared to Li_4SiO_4.

It is demonstrated that the core-shell pebble fabrication system designed in this study can fabricate pebbles with higher lithium density and excellent atmospheric stability.

Long-term annealing performance of Li-6 enriched biphasic Li_4SiO_4/Li_2TiO_3 pebbles

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For the ITER TBM and the DEMO HCPB, advanced ceramic breeder (ACB) pebbles serve as the solid EU reference tritium breeding material. ACB consists of two coexisting phases: lithium orthosilicate (Li_4SiO_4) and lithium metatitanate (Li_2TiO_3). In a future fusion reactor, ceramic breeders will have to withstand harsh conditions including the neutron radiation, the high temperatures and mechanical stresses. Accordingly, the tritium breeder materials have to be analysed intensively and need to demonstrate a stable performance under reactor-relevant conditions.

For this study, the issue of stability at high temperatures during the long-term operation in the reactor was chosen. Three batches of Li-6 enriched ACB pebbles that differed in their Li_2TiO_3-contents (30 and 35 mol.%) and in their Li-6 enrichments (60 and 90 at.%) were used. The ACB pebbles were heat treated in a He + 0.1 vol.% H_2 atmosphere at DEMO relevant temperatures. Before and after the heat treatment, the ACB materials were characterised with regard to their chemical and phase composition, microstructure, mechanical strength, and porosity.

The thermal long-term stability of different ACB pebbles, relating to their material properties after different annealing durations, will be presented. The results will demonstrate the state of development of EU ACB pebbles for their use in the HCPB blanket. As these ACB pebbles are also foreseen for upcoming neutron irradiations, the long-term annealing experiments will also help to facilitate the future distinction of effects resulting from the neutron irradiation from effects due to high temperatures.

Keywords: advanced ceramic breeder, thermal long-term stability, lithium orthosilicate, lithium metatitanate, material properties

Development of Lithium Titanate Ceramic Pebbles By Freeze Granulation and Freeze Drying Method

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Lithium titanate (Li_2TiO_3) is considered as one of the candidate material for tritium breeding. IPR (Institute for Plasma Research) is developing Li_2TiO_3 powder solid-state reaction method followed by pebble preparation. In this work, Li_2CO_3 (lithium carbonate) and TiO_2 (titanium di-oxide) are used as a raw material for the Li_2TiO_3 preparation. Li_2TiO_3 powder is prepared by high energy ball milling followed by calcination at 1000° C for 5 h duration. The reaction parameters are estimated by the thermo-gravimetric and X-ray diffraction studies. Pebbles are fabricated by freeze granulation method. An experimental set-up is fabricated for freeze-granulation and freeze-drying experiments. Effect of the binder concentrations and nozzles to the pebble diameter are also studied. Pebbles are sintered at different temperature from 950-1150° C for 5 h. Various characterization on the Li_2TiO_3 powder, pellets and pebbles have been carried out for new batches which includes phase purity, surface morphology of the sintered pebbles, bulk density, porosity, pore size distribution. The details of the production procedure, their optimizations, and characterization will be discussed in this paper.

Keywords: lithium titanate, freeze granulation, microstructure, pore-size distribution

Fabrication and Characteristics of Li_2TiO_3 Pebbles Fabricated by using Powder Injection Molding Process

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Tritium breeder materials in solid type are usually used in a pebble form because of the high packing factor, efficiency of tritium extraction, and advantage of thermal conduction with the structural materials. The breeder pebbles are required a good sphericity and uniform microstructure from the viewpoint of a packing factor and a release of bred tritium. Therefore, in this study, the lithium metatitanate (Li_2TiO_3) pebbles have been fabricated by powder injection molding process in order to secure a high sphericity of pebbles. In addition, physical and mechanical properties of the pebbles have been investigated.

The Li_2TiO_3 green pebbles have been fabricated by using powder injection molding process. The diameter and sphericity of green pebbles was about 3.6 mm and 1.01, respectively. And then, the green pebbles were sintered at 1000 °C for 3 h in air atmosphere. The diameter of sintered pebbles about 3.4 mm. The volume shrinkage ratio was about 16 % during sintering process. The shrinkage ratio was used as a parameter to design the mold in order to fabricate the Li_2TiO_3 pebbles with 1 mm in diameter. The uniform microstructure with open pores in inside and outside of the sintered pebbles was observed. It is able to be expected a good release of bred tritium. In addition, the sintered Li_2TiO_3 pebbles had a high crush load due to their good sphericity and uniform microstructure. It is able to reduce production of the radioactive dust during operation of breeding blanket. The detailed fabrication process and characteristics of Li_2TiO_3 pebbles with 1 mm and 3 mm in diameter, such as porosity, crush load, effective thermal conductivity, and so on, will be addressed in this presentation.

Results of neutron irradiation experiments with Li_4SiO_4/Li_2TiO_3 at the WWR-K research reactor

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Reactor experiments are one of the few available methods for studying the release of tritium in lithium ceramics, which are promising materials for breeder blankets in fusion reactors. At the WWR-K reactor, in recent years, neutron irradiation experiments of lithium ceramics of the two-phase composition Li_4SiO_4/Li_2TiO_3 (LOS/LMT) has been carried out with in-situ detection of gas species by the vacuum extraction method [1]. Four irradiation campaigns were performed for ceramics: 1) 75 mol.% LOS + 25 mol.% LMT (pebble size 250-1250 µm), 2) 65 mol.% LOS + 35 mol.% LMT (250-1250 µm), 3) 65 mo.% LOS + 35 mol.% LMT (500-710 µm) and 4) 75 mol.% LOS + 25 mol.% LMT (500-710 µm).
This report will present the goals, description and main results of the irradiation experiments at the WWR-K reactor with bi-phasic lithium ceramics, will explain the general concept of research and will also give an outlook on future plans.

[1] T. Kulsartov, et al., Nuclear Materials and Energy, 30 (2022) 101115, https://doi.org/ 10.1016/j.nme.2022.101115.

Keywords: Neutron irradiation, tritium release, helium, advanced ceramic breeder, Li_4SiO_4 *and* Li_2TiO_3

In-situ neutron irradiation experiment for EU reference ceramic breeder material

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Fundamental knowledge of tritium breeder materials under the impact of neutron irradiation is necessary for the successful operation of ITER and the development of DEMO. Therefore, advanced ceramic breeder (ACB) material that serves as the EU reference material for the ITER TBM and the DEMO HCPB concept are being investigated in in-situ neutron irradiation experiments.
Neutron irradiation experiments will be performed in the WWR-K reactor at the Institute of Nuclear Physics (INP) in Kazakhstan. For the irradiation experiments, ACB pebbles produced using the KALOS process will be used. The ACB pebbles consist of 70 mol% Li_4SiO_4 and 30 mol% Li_2TiO_3 and are highly enriched in lithium-6.
The ACB material will be tested in the form of pebble beds in two temperature ranges (400–600 and 600–900 °C) and under different purge gas compositions (He and He + *x* % H_2). The tritium release behaviour will be investigated in-situ. The influence of changes in temperature and purge gas composition on the tritium release behaviour will be analysed. Furthermore, the tritium residence time will be determined under different conditions.
A detailed plan of the irradiation campaign including the experimental setup, the irradiation conditions and a description of the ACB material will be presented.

Keywords: Neutron irradiation, in-situ tritium release, tritium residence time, advanced ceramic breeder, Li_4SiO_4 and Li_2TiO_3

Progress of design and analyses of CN HCCB TBM for ITER

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ITER is the largest magnetic confinement fusion reactor in the world, which will achieve long-purse plasma operation and obtain up to 500MW of fusion power. According to the ITER project objectives, some DEMO blanket relevant technologies, such as tritium self-sufficiency, extraction of high-grade heat, will be demonstrated by testing of the ITER Test Blanket Modules (TBMs) in dedicated equatorial ports. For each ITER participant, different TBM concept was selected, according to the national fusion development strategy, and China finally determined to develop the Helium Coolant Ceramic Breeder (HCCB) TBM, due to its fantastic characteristics of good compatibility, no MHD pressure drop, high operating temperature and world-wide R&D database.

Since 2015, the conceptual design (CD) of CN HCCB TBM has been approved by ITER organization, which was composed of four sub-modules and one common back plate with size of ~0.462mx~0.48mx~1.67m, made by CN-RAFM steel (CLF-1 or CLAM). Currently in the preliminary design phase, further design update for CN HCCB TBM was carried out, and detailed engineering analyses were carried out to verify the new design, including neutronics analysis, thermo-hydraulic analysis, thermo-mechanical analysis, etc. Preliminary analysis results showed that: 1) the total TPR was slightly reduced to ~54 mg/FPD from conceptual analysis results, due to design change of Be pebble bed packing factor and local structure; 2) the coolant was uniformly distributed among the sub-modules, with max. deviation of 0.4%, 1.0%, 1.1% in the inlet, outlet and bypass manifold, respectively; 3) the maximum temperature of Li_4SiO_4 and Be pebbles for the sub-modules was 1148 K and 897 K respectively, well below their allowable value; 4) the total coolant pressure drop of entire module was ~0.16MPa, which was acceptable for current parameters of helium circulator (Pump head: 0.8MPa); 5) the maximum stresses under typical load conditions were within the RCC-MR limit, including normal operation, hydrostatic test, in-box Loss Of Coolant Accident (LOCA) condition, etc.

In the future, more detail engineering analyses will be performed for CN HCCB TBM, in order to verify its performance under multi-physics environment, and some design verification experiments will be carried out, regarding the flow, thermal, structural characteristics.

Keywords: ITER, CH HCCB TBM, analysis, neutronics, thermo-hydraulic, structural

ORNL Status and Progress in Research on Ceramic Breeder Blanket Materials

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The fusion energy sciences program at the Oak Ridge National Laboratory (ORNL) contributes in a wide range of technical areas to support fusion reactor development. In particular, topics related to the fusion blanket and fuel cycle are among the major thrusts in materials science and technology development at ORNL. Although several specific tritium breeding concepts and their materials needs have been explored, uncertainties remain on the requirements and solutions for the materials needed for the blanket components. Multiple engineering challenges are imposed by the harsh operating environment of neutron exposure, high temperature and stresses, and evolving material property changes due to the irradiation environment. We will present recent updates and progress on our research activities of specific interest for the ceramic breeder blanket concepts. The talk will cover several research topics: 1) neutron irradiation effects in lithium ceramic breeding materials, 2) silicon carbide for electrical insulation, thermal insulation and tritium permeation control, 3) tritium surrogate diffusion behavior under simulated operating conditions, 4) atomistic modeling to improve understanding of solid breeders, and 5) advanced manufacturing of ceramics of high potential for adoption in fusion reactors. Currently active work includes a) atomistic modeling of structural stability and tritium trapping/detrapping in irradiated Li_2TiO_3 and Li_2ZrO_3, b) hydrogen isotope permeation and trapping in candidate membrane and barrier materials, and c) our extensive program on irradiation effects in SiC and SiC composites. Future expansion will include irradiation experiments to examine the stability and properties of lithium ceramics using the established capabilities of the HFIR reactor and our suite of PIE testing and characterization equipment.
Keywords: lithium containing ceramics, irradiation effect, modeling, advanced manufacture

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Development of a novel breeding blanket using the solid-type Pb_xLi_y eutectic alloy for CFETR

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Chinese Fusion Engineering and Test Reactor (CFETR) aims to demonstrate fusion energy production up to 200 MW, and finally reach commercial electricity power level 1GW. Moreover, it will rely on the blanket for tritium production. The blanket is in charge of tritium breeding, neutron shielding and energy conversion. It can be classified into the solid and liquid blanket in view of the breeder materials state. Among them, the solid blanket has the advantages, such as good material compatibility and non-Magnetohydrodynamics (MHD) effects, because it uses the pebble beds as tritium breeder and neutron multiplier. However, the beryllium or alloy (i.e. $Be\&Be_{12}Ti$) is usually adopted for multiplying neutrons, and it will cause the very high cost of the solid blanket due to the scarcity of natural resources for beryllium. In this paper, one novel solid blanket utilizing the Pb_xLi_y eutectic alloy has been proposed to make up the above deficiency. This material can have a higher melting point by adjusting the atomic ratio of Pb/Li, thus it can be used in the solid blanket as tritium breeder and neutron multiplier. Based on this blanket, the neutronic and thermal hydraulic analyses are performed in view of the characterization on the tritium breeding, material temperature as well as the hydraulic design. The results show that this blanket can meet with the requirement of tritium self-sufficiency, and the global Tritium Breeding Ratio (TBR) is larger than 1.0. Besides, the material temperature is not beyond the upper limits. Therefore, this novel solid blanket is promising to greatly reduce the capital cost, and improve the economy of fusion reactor.

Keywords: Fusion blanket, Solid-type Pb_xLi_y, Neutronic, Thermal hydraulic

Corrosion of F82H in ceramic breeder pebble bed and its effect on hydrogen permeation

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Understanding the permeation behavior of bred tritium in a pebble bed breeding blanket is essential for making fuel cycle safe and self-sufficient in a fusion reactor. It is known that double corrosion layers form on reduced activation ferritic-martensitic (RAFM) steel surface by gas releases from ceramic breeder. However, its effect on hydrogen permeation behavior has not been elucidated.

A new permeation rig has been constructed to corrode F82H sample and simultaneously measure hydrogen permeation through it. The experiments were carried out at 623–773 K in sweep gas conditions (Ar gas added with 0.1% H_2 and 1.0% H_2). Both sides of the F82H sample after the permeation test were investigated using X-ray diffraction where the LTZO pebbles were crushed into powder using an agate mortar. The surfaces of the F82H samples in contact with the LTZO pebbles were observed by field emission scanning electron microscope (SEM, Zeiss Ultra 55). Depth profiles of the F82H samples were investigated using GD-OES with a GD Profiler 2 (Horiba Ltd.).

The corrosion layer formed on the F82H sample had a dense microstructure, in which the total thickness of the layer were a few micrometers. It was observed that the surface corrosion layer reduced hydrogen permeation flux at least by one order of magnitude. The permeation reduction factors were 20–50 at the water-coolant temperature of a blanket. The activation energies were obtained to be 0.65–0.78 eV, which were higher than those measured with the bare F82H sample (approximately 0.45 eV). It is expected that the layer has a self-repairing ability as the corrosion occurs spontaneously inside a breeding blanket.

Keywords: Breeding blanket, pebble bed, compatibility, hydrogen permeation

Corrosiveness of solid tritium breeders to RAFM steel considering evaporated lithium compound and irradiation

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In future D-T fusion reactor, self-sufficiency of tritium is one of the critical issues to maintain steady-state operation of the fusion reactor. Lithium ceramics have been proposed as prominent tritium breeder candidates for solid tritium breeding blanket. Tritium breeders and structure materials (RAFM steels) will be contact during operation of the fusion reactor under high temperature for a long time. The RAFM steels will be corroded by tritium breeders. The keys of corrosiveness to RAFM steel by tritium breeders are: (1) High temperature, structure material will be corroded during contact with tritium breeders for long time at high temperature; (2) Irradiation, structure materials will be irradiated by neutrons and energetic particles which will promote corrosion; (3) Evaporated Li compound, in the tritium breeding zone, the highest temperature will be 900°C. At this temperature, large lithium compounds will be evaporated and deposited on structure materials inducing corrosion. Therefore, it will affect the tritium permeability, mechanical and thermal properties of the structural material. In addition, evaporated lithium compounds will deposit in tritium sweeping tubes affecting tritium recovery. Purposes of this work: Investigation on effects of high temperature, evaporated lithium compounds and irradiation on corrosiveness to RAFM steel. The research focuses on the corrosion characteristics of CLF-1 RAFM steel by Li_4SiO_4 and Li_2TiO_3 pebbles under sweeping gas of He + 0.1% H_2. For the corrosiveness considering evaporation deposition of lithium compound, SIMS tests at contact area and non-contact area were supplemented. It can be seen that there is a large amount of lithium in the non-contact zone, which proves that there is a large amount of lithium compound on the non-contact zone, resulting in serious overall corrosion of structural materials. From the cross-section mapping, the corroded depth by Li_4SiO_4 is higher than that by Li_2TiO_3 pebbles. The corroded products are complex including Lithium iron oxide, Lithium chromium oxide, iron oxide, and so on. For the corrosiveness considering irradiation, the corrosion became more and more serious and the corroded layer appeared shedding with the increase of irradiation dose, indicating that irradiation would promote corrosion. A more detailed analysis has been proceeding.

Keywords: corrosion; tritium breeder; nuclear fusion, irradiation; lithium evaporation

Microstructure, corrosion behavior, and mechanical properties of ARAA after compatibility test with Li_2TiO_3 pebbles.

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The Korean test blanket module, a helium-cooled ceramic reflector, aims to achieve a self-sustaining tritium cycle using reduced activation ferritic/martensitic steels and ceramic breeder pebbles. To ensure safe and reliable operation of the fusion reactor, it is crucial to investiage the diffusion between structural materials and tritium breeder since the diffusion of lithium and oxygen can lead to the degradation of mechanical properties. In this study, we investiaged the compatibility between advanced radiation reduced alloy (ARAA) and lithium titanate (Li_2TiO_3) pebbles for up to 35 days in He-0.1vol% H purge gas. After annealing at 550°C, we analyzed the microstructure, corrosion product phases, and mechanical properties of structural materials. The hardness and the elastic modulus before and after the tests were compared using nanoindentation. The ion concentration profiles in pebbles were also measured by secondary ion mass spectrometry and the thickness of the oxide layer was estimated for operation time.

Keywords: ceramic breeder pebble, Li_2TiO_3, ARAA, compatibility, mechanical properties.

Tritium transport model of lithium-based ceramic pebbles in Ecosimpro

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Ecosimpro model of tritium transport in lithium-based ceramic pebbles
The breeding blanket is a fundamental component in a nuclear fusion reactor due to the requirement to ensure fuel self-sufficiency. Different lithium-based ceramics have been studied as possible breeding materials. Various ceramics such as Li_4SiO_4, Li_2TiO_3, or biphasic mixtures of both have been studied as strong candidates for TBMs or future BBs. In past experiments, tritium desorption from ceramics into a purge gas has been observed under different temperature conditions. It has been also detected that the addition of hydrogen and water to the purge gas can encourage the surface processes.

The modelling of processes related to tritium transport is a prediction tool that provides support for the design of breeding blankets. For more than a decade, the tritium transport modelling libraries of EcosimPro have been the reference for the study at the system level of both TBM and BB and their auxiliary systems in Europe.

In this work, the first model of ceramic pebble and its desorption to purge gas in EcosimPro is presented. The model considers the influence of the chemical composition of the purge gas on desorption and on the molecular form that tritium takes. By simulating different representative experiments, the necessary validation exercise is carried out to reduce uncertainty in predictions.

Keywords: breeding blanket, modelling, EcosimPro, tritium transport, ceramic pebbles

Application of Tritium transport model on a demo breeding blanket system

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In implementing the He-cooled breeding blanket system for a fusion reactor, many design elements and experiments must be supported, and the development of a tritium transport model represents a key element for safety and design of the breeding blanket system. Because a tritium release into the environment from the breeding blanket system can be a radioactive risk, its accurate calculation and prediction shall be preceded for a design, safety, licensing and maintenance assessment of breeding blanket system.

To address this point, THETA-FR(Tritium/Hydrogen Enhanced dynamic Transport Analysis Tool for Fusion Reactor) was developed by a collaborative effort between Korea Institute for Fusion Energy(KFE) and the University of California, Los Angeles(UCLA)

THETA-FR is constructed from the integration of Matlab Simulink and COMSOL multiphysics. Because the each components of the breeding blanket system - first wall, breeding unit, pipes and etc. are respectively modelled and interconnected, the dynamic hydrogen isotopes (H/D/T) transport phenomena in the breeding blanket system can be calculated in THETA-FR.

In this paper, the components/system layout, the calculation process of the THETA-FR are introduced, while tritium permeation/release into the environment in a Korea demo breeding blanket system are predicted.

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Keywords: Breeding blanket system, Tritium transport model, COMSOL multiphysics, Matlab Simulink

Study on Hydrogen Isotope Permeation Behavior of High-Entropy Alloys Coatings

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Fusion reactors are fueled by hydrogen isotopes, including deuterium and tritium. As a result of a small atomic radius and high activity, hydrogen iso-tope atoms are easy to diffuse through metal materials, especially for the structural material of a tritium breeder blanket, where lithium atoms combined with fusion neutrons for producing tritium. The common practice is to prepare a tritium permeation barrier on the surface of the structural materials.

High-entropy alloys are novel systems, which contain at least five main elements, and the atomic percentage of each main element is between 5 at% and 35 at%. As a result of special effects, including a lattice distortion effect, cocktail effect, hysteresis diffusion effect and high-entropy effect, high-entropy alloys behave with superior mechanical properties, good corrosion resistance and good irradiation stability. Recently, high-entropy alloy coatings, which behave with an outstanding comprehensive performance, have also been developed. However, the hydrogen permeability of other HEA coatings with different main elements is unknown at present and needs to be further studied.

In this study, three new types of AlCrFeTiNb, AlCrMoNbZr and AlCrFeMoTi HEA coatings were prepared on a CLF-1 substrate by magnetron sputtering. The hydrogen permeability of HEA coatings prepared by magnetron sputtering technology were tested using gas-driven deuterium permeation and electrochemical hydrogen permeation methods. The gas-driven per-meation results show that the deuterium permeation resistance of the AlCrFeTiNb coating is the worst because of the unstable structure at a high temperature. Scanning electron microscope (SEM) and X-Ray Diffraction (XRD) analysis proved a loose surface morphology of the AlCrFeTiNb coating and demonstrated the formation of iron-based oxides after deuterium permeation experiments. A high content of iron in HEA coating is disadvantageous for improving the hydrogen isotope permeability. Differently, electrochemical hydrogen permeation reveals that the AlCrMoNbZr coating could resist hydrogen permeation better in a corrosive environment (0.2 mol/L KOH solution). The AlCrFeMoTi coating was peeled off after an electrochemical hydrogen permeation test due to the poor corrosion resistance. The hydrogen behavior of HEA coatings was discussed in detail. Our study provides a promising thought on hydrogen permeation of HEA coatings.

Keywords: high-entropy alloy coatings; gas-driven permeation; electrochemical hydrogen permeation; corrosion

Modelling transport of fragmentation and dust particles in granular tritium breeder material inside fusion reactors

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The European solid breeder blanket concept Helium Cooled Pebble Bed (HCPB) to be tested in ITER uses advanced lithium ceramic breeder (ACB) material in the form of pebbles. During the HCPB breeder blanket operation, a fragmentation of the pebbles due to thermomechanical loads can occur, as well as the formation of dust. This dust represents a safety issue, as it can block purge gas paths inside the HCPB or it can be transported with the purge gas from the pebble bed into the tritium extraction system (TES) of the HCPB blanket where it can be accumulated in the TES components, especially in filters. Therefore, it is important to understand the ACB pebbles' fragmentation mechanism and dust formation, as well as the subsequent transportation as a result of the mere mechanical interaction between the purge gas and the pebbles. For this purpose, a coupling between the open-source DEM (Discrete Element Method) code LIGGGHTS and the open-source CFD (Computational fluid dynamics) code OpenFOAM is being used in this work. As a first step, we introduce a suitable way to generate particle assemblies with a high packing factor of around 64% by using the pure DEM code LIGGGHTS. Afterwards, a sensitivity study is presented, to find the smallest pebble assembly for which the wall effect doesn't affect the pressure drop gradient anymore. Nine different sizes of assemblies were investigated. As the result, no appreciable differences were found starting from an assembly with a tube-to-particle diameter ratio of about 16. Next, currently ongoing dust transport investigations are discussed, focusing the attention on the transport of a single dust particle first and on a dust population next. From the study of a single dust particle transport inside a pebble assembly, it appears that the dust can travel faster at the boundary and along the edges compared to the bulk, following smoother trajectories. This is due to both the pebbles arrangement and the purge gas flow characteristics. There seems to be also a larger probability of dust blockages in the bulk.

Keywords: HCPB, breeding blanket, CFD-DEM, pressure drops gradient, dust transport

Development of a system-level code and application to the tritium transport for the water cooled ceramic breeder blanket

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Abstract: A dynamic analysis tool for evaluating the tritium permeation, inventory, and losses for the Water Cooled Ceramic Breeder blanket (WCCB) and the whole external fuel circulating system of Chinese Fusion Engineering Test Reactor (CFETR) is presented in this paper. Zero dimensional and one dimensional analysis model are combined in this program, which aims to estimate the tritium related quantities in the system level. The tritium inventory in the material defects is taken into account. The sensitivity analysis about key parameters are carried out and the influence of boundary conditions between different interfaces on tritium transport is considered. The key output quantities of the model are tritium inventory in structural material, purge gas and coolant loops, tritium extracted by the tritium extraction system, tritium purified by the coolant purification system and tritium losses.

Keywords: WCCB blanket; Tritium permeation; Tritium inventory; Boundary conditions

Effects of vibration conditions, spatial confinement and friction on mixing and segregation characteristics of mixed pebble beds for CFETR WCCB blanket.

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In the WCCB blanket, $Be_{12}Ti$ and Li_2TiO_3 particles are mixed uniformly to form a mixed bed of high tritium breeding capability and thermal conductivity. However, the mixed pebble bed, consisting of different types of particles, may transition from the mixed state to the segregation state due to transportation, external vibration and other disturbance. Therefore, it is necessary to investigate the dynamic performance of mixing and segregation characteristics of confined mixed pebble beds under vibration.

This work will present both experimental and numerical results. In the experiments, a transparent acrylic container filled with multi-sized particles is fixed to a vibration device to investigate the segregation characteristics for the confined mixed pebble beds of different particle size ratios under different vibration conditions. To model tritium purge gas in breeder zones of blankets, a helium loop is constructed to provide flow at 0.1~3.0 MPa with a maximum flow rate of 80 Nm^3/h. The evolution of the mixed state of particles can be observed using a high-speed camera. The vibration frequency and amplitude are varied in the range of 0-100Hz and 0-1mm. And the influence of vibration conditions, spatial confinement and particle friction on three dominant segregation mechanisms (percolation, convection, diffusion) can be clarified.

In the simulation of the discrete element method (DEM), a $Be_{12}Ti$ and Li_2TiO_3 mixed pebble bed is vibrated with a frequency of 0-40 Hz and an amplitude of 0-1 mm. The unconfined mixed bed is segregated visibly when Γ (vibration acceleration) exceeds 1 g (gravitational acceleration). when Γ ranges in 1.8-3 g, the segregation index reaches its maximum and the percolation mechanism predominates while the diffusion and convection mechanisms are minimal. Besides, the coefficient of friction was found to have a significant impact on the segregation of the unconfined pebble beds. The unconfined mixed pebble beds transition from the mixed state to the Brazil-nut segregation state by increasing the coefficient of friction from 0 to 1. However, when the top free spatial height of the confined pebble bed is less than 2.5mm, the segregation disappears and the confined pebble bed remains in the mixed state after vibration. This work is useful to find solutions to retain the mixed state of the Li_2TiO_3&$Be_{12}Ti$ mixed pebble bed.

Keywords: Vibration, Segregation, Discrete element method, Mixed pebbed bed

Thermal, uniaxial compression and flow simulations of packed beds of spherical and elliptical shaped pebbles

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Lithium ceramics as pebble beds are chosen for the tritium breeding materials in blanket of fusion reactor. The packing structure of the pebble bed influences the thermo-mechanical behavior of the pebble bed as well as the flow of purge gas through it. Ideally, pebbles are considered as perfect spheres for numerical simulations; but, in practical case pebbles are non-spherical. Thus, the packing of non-spherical ellipsoidal pebbles are studied in this work by Discrete Element Method (DEM) simulations. The non-spherical pebbles were created using multi-sphere approach, a composite pebble of three spheres having individual coordinates and individual boundary forces was created to depict an ellipsoidal shape. The simulations are carried out for mono-sized, binary-sized and poly-dispersed pebbles for both spherical and non-spherical shape. In this work, the uniaxial compression test has been simulated using the DEM and the effect of pebble size ratio and size variation on the stress-strain response, packing fraction (with stress), bed stiffness and average contact force have been studied. The experiments have been performed on alumina (Al_2O_3 - spherical) and lithium metatitanate (Li_2TiO_3 – non-spherical) pebble beds at room temperature under cyclic loads (up to 6 MPa). CFD simulations have been carried out to study the effect of non-sphericity on pressure drop of packed beds and the results are compared with experimental study. Estimation of thermal conductivity of these pebble beds has also been done by steady state thermal simulations and the same were compared with transient hot wire experimental results. The details of the DEM, CFD and thermal simulations will be discussed in this paper.

Keywords: discrete element method, pebble bed, multi-sphere approach, uniaxial compression test, thermal conductivity

A study of purge flow characteristics and effective thermal conductivity of pebble bed: Experiments and Simulation by ANN

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This study aimed to investigate the purge flow characteristics and effective thermal conductivity of pebble beds in fusion blanket applications using a combination of experiments, simulation, and artificial neural networks (ANN). First, experimental tests were conducted to measure the pressure drop and effective thermal conductivity of the pebble bed under different flow rates and temperatures respectively. Next, numerical simulations were performed to simulate the flow characteristics and heat transfer in the pebble bed, and the results were compared with the experimental data. Finally, an ANN model was developed to predict the effective thermal conductivity of the pebble bed based on the input variables of temperature, gas environment, and packing fraction.
The results showed that the effective thermal conductivity of the pebble bed was strongly influenced by the gaseous environment and temperature. The simulations provided valuable insights into the flow patterns and temperature distribution within the pebble bed, which were difficult to measure experimentally. The ANN model was able to accurately predict the effective thermal conductivity and pressure drop of the pebble bed based on the input variables, with a mean absolute error of less than 5%.

Overall, this study demonstrates the importance of understanding the flow characteristics and effective thermal conductivity of pebble beds in fusion blanket applications, and highlights the value of combining experimental, simulation, and ANN approaches for comprehensive analysis.

Keywords: Pebble bed, Pressure drop, Thermal conductivity, ANN, Computational Fluid Dynamics (CFD, DDPM-DEM

Simulation of Mechanical, Thermal, and Flow Characteristics of Pebble Beds for Solid-type Ceramic Breeding Blanket

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Pebble bed configurations are an attractive option for solid-type ceramic breeding blanket systems due to their potential advantages, including improved contact between pebbles and the wall, reduced stress concentration, efficient tritium release, and smooth gas flow. For breeder materials such as Li_2TiO_3 and Li_4SiO_4, understanding their properties in the form of a pebble bed is crucial in predicting the performance of the breeding blanket. The physical properties of pebble beds are critical because they affect not only the breeder material but also the structural material that makes up the breeding blanket. Therefore, researchers have used numerical methods like the discrete element method (DEM) to study the mechanical and thermal properties of pebble beds. Effective thermal conductivity, which is one of the most important characteristics, helps estimate temperature changes during the operation of a fusion power plant. Additionally, the flow characteristics of the purge gas used to extract tritium generated from the breeder material in the pebble bed is an important factor as it affects the design of the tritium extraction system. In this study, we present simulations of mono- and binary-sized pebble beds using the DEM to investigate their mechanical and thermal behavior, as well as purge gas flow characteristics.

Keywords: Discrete Element Method (DEM), Solid-type Ceramic Breeding Blanket, Pebble Bed, Effective Thermal Conductivity

Fabrication and tritium release properties of advanced tritium breeder: $Li_4(Si, Ti)O_4$ ceramic pebble

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Fusion is a potential source of safe, non-carbon emitting and virtually limitless energy. The tririum breeding blanket, which acts as the breeding zone of tritium fuel and serves as the main thermal power conversion system, is one of the most important components of a fusion reactor. The breeding materials are important as one of the key functional materials in the breeding blanket. Increased versatility in the available properties of tritium breeding materials can be of benefit to the design of TBM.

Lithium-containing ceramics are significant tritium breeders for the fusion blanket concept. The $Li_4(Si, Ti)O_4$ was designed as advanced tritium and the ceramic pebbles were fabricated using the freeze-drying method. XRD results showed that Ti was successfully incorporated into the Li4SiO4 structure. Subsequently, the tritium release properties of the $Li_4(Si, Ti)O_4$ sample pebbles were researched by temperature programmed desorption. The sample pebbles exhibited different tritium release characteristics and the tritium release temperatures were around 530 °C after a high dose irradiation. This is because H_2-tritium isotopic exchange reaction, meanwhile, the tritium release temperature gradually decreases with the increase of Ti atom. These results indicate that $Li_4(Si, Ti)O_4$ solid solution was found to be promising tritium breeder materials.

Keywords:Tritium breeder materials, $Li_4(Si,Ti)O_4$ solid solution, Fabrication and tritium release

Influence of various radiation types on radiation-induced processes in lithium orthosilicate-based ceramic breeder materials

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Lithium orthosilicate (Li_4SiO_4)-based ceramics are currently being developed as potential solid-state candidate materials for tritium breeding in future thermonuclear fusion reactors. In the present work, the influence of various radiation types on radiation-induced processes in Li_4SiO_4-based ceramic breeder materials was investigated and described. Investigated samples (i.e., powders, pellets, pebbles, etc.) were exposed to radiation of various types, masses, and energies: (1) photons (X-rays, gamma rays, and *bremsstrahlung*) with energies up to 6 MeV, (2) electrons (beta particles and accelerated electrons) with energies up to 10 MeV and irradiation temperatures up to 1285 K, and (3) accelerated ions with masses up to 28 amu (hydrogen, helium, oxygen, and silicon) and energies up to 10 MeV. The formed and accumulated radiation-induced defects and radiolysis products were analysed using various physico-chemical methods, e.g., *in-situ* luminescence measurement techniques, electron paramagnetic resonance (EPR), thermally stimulated luminescence (TSL), optical absorption (OA), and Fourier-transform infrared (FTIR) spectroscopy. The obtained data was directly compared with the results, which have been acquired for the long-term neutron-irradiated Li_4SiO_4-based ceramic materials from the HICU experiment (**H**igh neutron fluence **I**rradiation of pebble sta**C**ks for f**U**sion) with irradiation temperatures of about 1070-1120 K. On the basis of the obtained results, it can be concluded that radiation type, mass, and energy only slightly influences the structure, electron configuration, and thermal stability of radiation-induced defects and radiolysis products in the Li_4SiO_4-based ceramic breeder materials. However, significant impact of the fabrication-related effects (i.e., chemical and phase composition, fabrication approach, pre-treatment, etc.) and irradiation conditions (i.e., absorbed dose, dose rate, irradiation temperature, irradiation atmosphere, etc.) were detected on the radiation-induced processes during exposure to photons and electrons.

Keywords: Lithium orthosilicate, Tritium breeding, Radiation-induced processes

Secondary Ion Mass Spectrometry (SIMS) as an important tool to track compositional variations from ion-implanted and ion-damaged Advance Ceramic Breeder (ACB) compositions

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Accelerated ion beams are considered a convenient and powerful tool to approach the effect of neutrons on fusion materials. During the last years and framed within the FP8 and FP9 EUROfusion activities, both the radiation tolerance and the diffusion and release behaviour of H-isotopes from advanced ceramic breeders (ACB) have been studied at LNF-CIEMAT by means of high-energy ion beams and post-irradiation characterization techniques. Both implantation of light species and structural damage can be performed on ACB compositions with the aid of accelerated ion beams, with the additional advantage of the straight characterization of non-active irradiated samples. Tailoring the energy and fluence of the accelerated ion beam, an induced damage is caused in the solid structure due to the nuclear and electronic interaction of the impinging ions with the lattice atoms as the beam losses energy.
Therefore, IAM-KIT supplied ACB compositions of the LOS + 30-35mole% LMT oxide mixture were firstly submitted to heavy ion irradiation followed by implantation of H-isotopes to simulate the presence of bred tritium on the structurally damaged ceramic. At the CMAM facility (Madrid, Spain), irradiation campaigns by accelerating Fe (IV), I (IV) and H (II) ion beams on ACB pellets were then compositional and microstructurally analyzed at LNF-CIEMAT (Madrid, Spain) to determine the damage induced and the likely effects of the produced defects on the diffusion behaviour of implanted H-isotope ions.

The secondary ion mass spectrometry (SIMS) is a valuable analytical technique to directly determine the elemental composition of solids. Applied to ACB ion-irradiated pellets, the SIMS depth profiles of positive and negative ions assure both the damage induced to the original crystalline structure and the implanted light ions' distribution. As predicted by SRIM simulations, an effective damage is detected, the recoil matrix ions concentrating before the ion beam projected range. Furthermore, changes recorded with SIMS in the depth distribution of the implanted H(II) ions after lab temperature treatments indicate the gas release before 700°C. Therefore, the HCPB-type BB operational temperature is suitable for the tritium bred to be efficiently removed from the defect ACB structures with likely trapping centers.

Keywords: Advance ceramic breeders, ion-irradiation induced damage, secondary ion mass spectrometry, deuterium SIMS depth profile

TEM STUDIES IN SUPPORT OF THE HIGH RADIATION RESISTANCE OF ION-IRRADIATED ADVANCED CERAMIC BREEDERS

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One of the critical points for developing nuclear fusion technology is the design and development of tritium breeder blankets. Given the requirements for the future fusion reactor to generate the fuel necessary for continuous operation, an optimal solution must be found for a breeder blanket containing the Li necessary for the neutrons from the fusion reaction to produce T through nuclear reactions.

One of the possible technologies is the HCPB, which will use pebbles of a two-phase ceramic compound based on Li_4SiO_4 and Li_2TiO_3, the European reference materials for ITER. Although the scientific community is still researching to find the optimal fabrication method, it is essential to generate knowledge about the effect that radiation will have on pebbles. By developing well-controlled experiments simulating the neutron environment, we will obtain results to understand the relationship between the microstructure and the manufacturing process, with possible degradation of mechanical properties and defects produced by irradiation.

For this purpose, several irradiations using high-energy, heavy ions (O, Si, Fe and I) have been carried out at RT on Li_4SiO_4 - Li_2TiO_3 pebbles. FIB lamellae were extracted from the damaged region of each sample to study the microstructure. Other lamellae were extracted far away from the irradiated region to be considered undamaged bulk microstructure. S/TEM technique was used to characterize all samples in-depth and evaluate the generation and evolution of microstructural defects produced by ion-irradiation. Chemical and structural stability of the compound was determined as a critical feature in order to evaluate the radiation resistance.

Keywords: breeding ceramic, ion-irradiation, TEM microstructure, microestructural defects

Is H-isotope effectively trapped in structural defects of ion-beam damaged ceramic breeders?

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Breeder ceramics will be influenced by particles and ionizing radiation when located after the first wall in a fusion reactor. In fact, the functionality of this lithium-based ceramic oxide is the production of the tritium (T) fuel from the transmutation reactions with plasma neutrons. Therefore, the T transport through the breeder down to the pebble surface is still an issue for the researchers since the presence of radiation-induced defects in the crystalline network may trap the light ion changing the diffusion dynamics. Serious detrimental effects, such as the decrease of the tritium breeding ratio (TBR) and the consequent increase of the tritium inventory in the BB, will depend on the degree of trapping.
A systematic study on the defects produced by ion beams on breeder ceramics and their influence on H-isotopes diffusion is then within LNF-CIEMAT's priorities in the framework of the EUROfusion programme. By simulating neutron transmutation of lithium, the presence of tritium in the advanced European breeder composition has been experimentally committed at the LNF-CIEMAT. Pellet samples, prepared from IAM-KIT-supplied mixture of lithium ortho-silicate mechanically reinforced with lithium meta-titanate, were in-depth implanted with high energy H (or D) ions and its efficient outgassing recorded by techniques such as thermally stimulated desorption (TDS) and differential thermal analysis (DTA/TG). However, defect production is dynamic in an operating breeder where lithium is continuously consumed. Therefore, the H-isotope recovery should be tested in solids with the presence of defects, comparable to those induced by neutrons. Then, H-isotope implantation experiments have been recently carried out on breeder ceramics previously damaged with accelerated ions, a widespread practice to simulate the neutron action and nuclear damage formation. Several campaigns were addressed using heavy ion beams (Si, O, Fe, I), preparing a breeder ceramic with structural damage prior to H-isotope implantation and monitor its outgassing. This work summarizes the conclusions of the last 5 years experimental campaigns, obtained after analyzing the solid crystalline structure and the records of the H isotopes thermal stimulation desorption. In collaboration with the University of Latvia, an attempt has been made to identify radiation-induced defects and their role in the retention of H-isotopes by applying the electron spin resonance (ESR) technique. A high tolerance of the ceramic to ion-irradiation has been demonstrated, concluding the reduced production of new defects, although the characterization techniques demonstrate the effective retention of the implanted species.

Keywords: HCPB breeding ceramics, Ion-irradiation, defects, damage, H-isotope trapping

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