22-27 September 2019
Budapest, Hungary

ISFNT14

14th International Symposium on Fusion Nuclear Technology

Book of Abstracts

ISBN 978-615-5270-58-1

© ISFNT-14

This book is working material for the 14th International Symposium on Fusion Nuclear Technology.

All rights reserved, expect for the authors' right to publish their materials wherever they decide.

The professional and grammatical quality of the abstracts is the authors' responsibility.

TABLE OF CONTENTS

KEYNOTE LECTURES

Status of the ITER Project: Challenges and Opportunities

KN-1

Bernard Bigot *

ITER Organization, St Paul-lez-Durance Cedex, France

ITER reached in December 2018 completion of about 60% of the work required to achieve First Plasma. Significant progress has been made on ITER infrastructure since the 2017 ISFNT-13; the most visible is the completion of many key buildings, such as the tokamak assembly building, the cryogenic plant, and the magnet power supply building. The tokamak building will be ready for equipment in 2020 and the bioshield is already to full height. Key systems have begun commissioning in 2018, including the steady-state electric network and the component cooling water. The cryogenic plant and the magnet power-supply buildings are complete, and these systems begin commissioning in 2019.

Equally impressive is progress toward manufacturing components of the ITER tokamak. The base and lower cylinder of the cryostat have been assembled on the ITER site. The first of the six modules of the central solenoid has been wound, and three of the six poloidal field coils are presently being wound. The first winding pack of the toroidal field magnets is complete, as is the first casing, which has been verified to meet the high tolerances required (<0.5 mm). The first complete set of parts comprising a vacuum vessel sector has been fabricated and demonstrated to meet strict tolerances (<1 mm). A divertor cassette body prototype is manufactured and will be delivered to the ITER site.

Major components are arriving onsite with increasing frequency. More than 80 Highly Exceptional Loads (HEL) have been delivered as of the end of 2018. Installations will continue in the coming year, with full-scale assembly beginning in 2020. The sequence of ITER operation from First Plasma (FP) to the achievement of the Q = 10 project goal has been consolidated in a Staged Approach. The ITER Research Plan has been revised in 2017 to be consistent with the systems available in each phase. Recently, an adjustment of the baseline configuration of the machine has been made with two equatorial ports allocated to the Test Blanket Systems and the development of a better performing Disruption Mitigation System.

Keywords: *ITER, Magnetic Fusion, Tokamak, Burning Plasma, Tritium Breeding*

Corresponding author: *Bernard.Bigot@iter.org*

The European Fusion Roadmap

KN-2

Antonius J.H. Donné *

EUROfusion, Garching, Germany

The European Roadmap to the realisation of fusion energy breaks the quest for fusion energy into eight missions. For each mission, it reviews the current status of research, identifies open issues, and proposes a research and development programme. It points out the needs to intensify industrial involvement and to seek all opportunities for collaboration outside Europe.

A long-term perspective on fusion is mandatory since Europe has a leading position in this field and major expectations have grown in other ITER parties on fusion as a sustainable and secure energy source. The roadmap covers three periods: The short term which is roughly until 2030, the medium term until 2040 and the long term.

ITER is the key facility of the roadmap as it is expected to achieve most of the important milestones on the path to fusion power. Thus, the vast majority of resources proposed in the short term are dedicated to ITER and its accompanying experiments. The medium term is focused on taking ITER into operation and bringing it to full power, as well as on preparing the construction of a demonstration power plant DEMO, which will for the first time supply fusion electricity to the grid. Building and operating DEMO is the subject of the last roadmap phase: the long term. It might be clear that the Fusion Roadmap is tightly connected to the ITER schedule. A number of key milestones are the first operation of ITER (presently foreseen in 2025), the start of the DT operation foreseen in 2035 and reaching the full performance at which the thermal fusion power is 10 times the power put in to the plasma.

DEMO will provide first electricity to the grid. The Engineering Design Activity will start a few years after the first ITER plasma, while the start of the construction phase will be a few years after ITER reaches full performance. In this way ITER can give viable input to the design and development of DEMO. Because the neutron fluence in DEMO will be much higher than in ITER (atoms in the plasma facing components of DEMO will undergo 50-100 displacements during the full operation life time, compared to only 1 displacement in ITER), it is important to develop and validate materials that can handle these very high neutron loads. For the testing of the materials a dedicated 14 MeV neutron source is needed. This DEMO Oriented Neutron Source (DONES) is therefore an important facility to support the fusion roadmap.

The presentation will focus on the strategy behind the fusion roadmap and will describe the major challenges that need to be tackled on the road towards fusion electricity. Encouraging recent results will be given to demonstrate the outcome of the focused approach in European fusion research.

Corresponding author: *tony.donne@euro-fusion.org*

Lessons Learned from 40 Years of Fusion Research and Future Directions for Fusion Nuclear Science and Technology R&D

KN-3

Mohamed A. Abdou *

University of California- Los Angeles, Los Angeles, CA, USA

Fusion Technology R&D has been pursued worldwide for > 40 years. There has been much progress, but also much disappointment in resolving a number of serious issues that are fundamental to the feasibility and attractiveness of fusion energy systems. Many of these issues are in the area of Fusion Nuclear Science and Technology (FNST) where there are critical go/no-go problems for which HOW and WHERE to perform the R&D is a challenge, yet there is not a credible strategy being adopted, communicated, nor pursued. Therefore, it is prudent to reflect on successes and failures of the past 40 years to understand where we succeeded and why, where we failed and why, and drive key lessons learned. This will illuminate our quest for developing an effective strategy for successful development of FNST, and fusion energy systems, in the next 40 years at an affordable cost and reasonable time.

**Corresponding author:* *abdou@fusion.ucla.edu*

Chinese ITER Project Overview - Status & Progress

KN-4

Delong Luo *, Yalan Zhu *, Kaihui He, Wen Zhao, Tongzhen Fang

China International Nuclear Fusion Program Execution Center (ITER China Domestic Agency), Ministry of Science and Technology, Beijing, China

As one of the mega-science projects that may fundamentally resolve energy problems for mankind in the future, the ITER Project is a crucial step forward and a joint endeavor collectively made by China, Europe, India, Japan, Korea, Russia and the United States after years of designing and negotiations. The Chinese government, domestic fusion community and industrial enterprises are actively engaged in the implementation of ITER and in the promotion of fusion energy activities in China. The purpose of this paper/presentation is thus to introduce the progress of ITER Project implementation in China, the general domestic fusion research and development and the next step for fusion energy.

As the founding member of ITER Organization (IO), China is dedicated to the construction of the reactor and organizational management. Ever since China joined ITER, 14 procurement arrangements and 4 amendments were signed with IO, and impressive results have been accordingly yielded and a large amount of components were delivered to IO and other Domestic Agencies both on time and with quality. Over the years, there witnessed considerable development in fusion industry in China. For instance, the superconductor procurement packages enables China to contribute ITER superconducting strand and also other area such as MRI. For the first wall procurement package, beryllium-tiled "fingers" from China have performed successfully under high heat flux testing at a dedicated facility in Russia. The test results confirm that the joining technique chosen for beryllium to copper bonding has been shown to meet all ITER requirements, thus making China the first ITER Member to complete such challenging task. Also, a full-scale prototype of a blanket shield block manufactured in China successfully passed acceptance tests, including the challenging hot helium leak testing in February, marking an important qualification milestone in the ITER blanket program. China is providing an extensive array of supports and clamps for ITER's superconducting magnet systems - all in all, more than 1,600 tons of equipment have been and will be delivered to site. The successful qualification of the Correction Coils case welding with 20 kW laser welding machine will not only ensure the welds with better quality, and properly protect the winding packs from being overheated, but also it will help improve the production efficiency significantly. Such results were highly appraised by IO and international experts in this field.

Over the decade, the ITER Project, together with its Members' domestic fusion facilities, provides China with a precious opportunity to improve and industrialize fusion as a potential industry. An up-to-date standard system and information channel are in the blueprint, and more work are underway. China's domestic enterprises, factories, and workshops have and are still benefiting from their involvement in ITER and are becoming more competitive in the international market.

Progress has also been achieved in the field of tokamak physics and engineering technology, particularly in those core technologies such as plasma confinement and transport, MHD instability, high energetic ions physics, plasma heating and current driving, new fueling technique and divertor physics. Domestic programs for fusion R&D have been deployed to boost the fusion reactor design, the upgrade of EAST and HL-2A, fusion material, DEMO reactor and TBM as well. As with other ITER Members, China is also making its own ambitious roadmap for fusion energy. Conceptual design of China Fusion Engineering Test Reactor (CFETR) were completed in 2015, and integration engineering design of CFETR and related researches are on the way. An advanced test platform for fusion reactor key components was approved to be established in Hefei and the International Fusion Energy Center for joint research was also set up in Hefei headed by renowned fusion experts.

**Corresponding author:* *luodl@iterchina.cn, zhuyl@iterchina.cn*

PLENARY LECTURES

Overview of Fusion Nuclear Technologies in China

PL-1

Yican Wu *, FDS Team

Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Hefei, Anhui, China

Plasma technology and fusion nuclear technology (FNT) are two key aspects of fusion energy systems. During the last two decades, the fusion facilities including several tokomaks have been set up in China mainly to study the physics of plasmas. But great challenge for fusion energy application still lies ahead. Due to the harsh environment (e.g. high neutrons fluxes, magnetic fields, etc.) and the requirements of steady-state operation and tritium self-sufficiency, fusion energy application will be determined largely by FNT.

In this paper, we will review the activity plans, status and main results on FNT in China both for ITER and DEMO projects. The wide variety of R&D activities on FNT currently underway in China will be summarized by four categories: neutronics, materials and blanket, tritium cycle technology, and safety, environment & socio-economics and the highlights in each category will be emphasized. For example, the D-T fusion neutron source named HINEG-I has been constructed in China for basic research on neutronic physics and radiation protection, whose neutrons yield has achieved the highest level among the same kind neutron generator currently in operation. The Super Multi-functional Calculation Program for Nuclear and Radiation Simulation (SuperMC) developed by China has been widely applied in more than 60 nations. China Low Activation Martensitic steel (CLAM) steel has exhibited superior performance compared to other RAFM steel and its code qualification for RCC-MRx is also undergoing. These technologies from China have made important contributions to the field of FNT.

In addition, the paper will present the development strategy for the next step in particular regarding the implications on the design and technology advancement and will also summarize the FNT related international collaborative programs advocated by China.

Keywords: *fusion nuclear technology, HINEG, CLAM, SuperMC*

Corresponding author: *yican.wu@fds.org.cn*

Overview of Japan-US Project PHENIX: Technological Assessment of He-Cooled Divertor with Tungsten for DEMO Reactors

PL-2

Yuji Hatano [a]*, Daniel Clark [b], Yoshio Ueda [c], Yutai Katoh [d], Takehiko Yokomine [e], Minami Yoda [f], Adrian S. Sabau [d], Tatsuya Hinoki [e], Akira Hasegawa [g], Lauren M. Garrison [d], J. Wilna Geringer [d], Yasuhisa Oya [h], Masashi Shimada [i], Dean Buchenauer [j], Takeo Muroga [k]

[a] *University of Toyama, Japan*
[b] *United States Department of Energy, USA*
[c] *Osaka University, Japan*
[d] *Oak Ridge National Laboratory, USA*
[e] *Kyoto University, Japan*
[f] *Georgia Institute of Technology, USA*
[g] *Tohoku Univeristy, Japan*
[h] *Shizuoka University, Japan*
[i] *Idaho National Laboratory, USA*
[j] *Sandia National Laboratory, Livermore, USA*
[k] *National Institute for Fusion Science, Japan*

The main goal of Japan-US PHENIX project is to evaluate the technical feasibility of helium (He) gas-cooled, tungsten (W)-armored divertor components for fusion demonstration reactor (DEMO). To achieve this goal, the project examined cooling performances of high pressure and high temperature multi-nozzle impinging jets, and neutron irradiation effects on various properties of W and newly developed W-based alloys (W-Re alloy, K-doped W, K-doped W-Re alloy, etc.). The heat transfer studies using a He loop at the Georgia Institute of Technology at maximum heat flux of 8 MW/m^2 resulted in a correlation for the thermal-hydraulics of modular “finger-type” He-cooled divertors; numerical parametric design studies suggest that a simpler configuration, the “flat design”, may have similar thermal-hydraulic performance. Neutron irradiation was performed in High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) at ~500, 800 and 1100 °C with thermal neutron shielding up to the fast neutron fluence of 3.8×10^{25} m^{-2} (E > 0.1 MeV). Extensive post-irradiation examinations were performed at ORNL to understand irradiation effects on microstructure, thermal diffusivity, heat load resistance, and mechanical properties. K-doped W-Re alloy showed no serious ductility degradation after neutron irradiation, while significant embrittlement was observed for W. Deuterium (D) retention in irradiated W was examined using Tritium Plasma Experiment (TPE) at Idaho National Laboratory. The amount of D retained in W decreased with increasing irradiation temperature due to recovery of defects but still significant after irradiation at 1100 °C. Interestingly, He-seeding in D plasma resulted in orders-of-magnitude

reduction in D retention even for neutron-irradiated samples. The technical feasibility of He gas-cooled divertor components will be discussed in the presentation based on these accomplishments.

Keywords: *Divertor, Tungsten, Helium Cooling, Neutron Irradiation, Heat Transfer, Fuel Retention*

**Corresponding author:* *hatano@ctg.u-toyama.ac.jp*

Overview of Recent ITER TBM Program Activities

PL-3

Luciano M. Giancarli [a]*, Xavier Bravo [a], Seungyon Cho [b], Marco Ferrari [c], Takumi Hayashi [d], Byoung-Yoon Kim [a], Artur Leal-Pereira [a], Jean-Pierre Martins [a], Mario Merola [a], Romain Pascal [a], Iva Schneiderova [a], Qian Sheng [e], Amit Sircar [f], Yuri Strebkov [g], Jaap van der Laan [a], Alice Ying [h]

[a] *ITER Organization, St. Paul Lez Durance, France*
[b] *NFRI, Daejeon, Korea*
[c] *Fusion for Energy, Cadarache Antenna, Saint Paul lez Durance, France*
[d] *QST, Rokkasho, Aomori, Japan*
[e] *ITER China Domestic Agency, Beijing, China*
[f] *Institute for Plasma Research, Bhat, Gandhinagar, India*
[g] *Nikiet, Moscow, Russia*
[h] *University of California, Los Angeles (UCLA), USA*

In 2018, the ITER Test Blanket Module (TBM) Program has significantly evolved from both organizational and technical points of view. Organizationally, a TBM Project Team has been established. From the technical side, the Test Blanket Systems (TBSs) are now located in two Equatorial Ports, and consequently, allowing only four TBSs to simultaneously operate while preserving the possibility of testing six TBSs over the ITER lifetime. This reconfiguration enables reallocation of the space in the Tokamak Complex to install TBS components that were suffering due to initial space constraints and integration issues.

The selection of the four TBSs as the initial configuration for ITER testing is not defined yet; one possible option could involve two water-cooled TBSs (e.g., one using lithiated-ceramic pebbles as breeder and the other using the liquid lithium-lead alloy) and two helium-cooled TBSs both using lithiated-ceramic pebbles as a breeder. The corresponding TBS designs could be similar to the various approved TBS conceptual designs. These initial TBSs will start operating during the last ITER non-nuclear phase. Other kinds of TBSs could possibly operate in some later nuclear phases of ITER.

The design of the infrastructures needed for hosting the four TBSs (e.g., port plugs, port cell common components, common maintenance tools and equipment) has significantly progressed. The design of the TBS Connection Pipes (CPs) System has reached its final phase since the CPs are captive components and need to be installed before the First Plasma.

After recalling the main features of the various TBSs, this paper describes the main changes introduced into the TBM Program and their consequences on the associated testing strategy, the associated safety and integration aspects, and the interfaces. It also addresses the main progresses achieved in the design of the associated infrastructures and of the TBS CPs System.

Keywords: *TBM, breeding blanket, tritium, TBS*

**Corresponding author:* *Luciano.Giancarli@iter.org*

Preliminary Technical Performance Assessment of the European DEMO Breeding Blanket through the TBM Project

PL-4

Italo Ricapito [a]*, Y. Carin [b], F. Cismondi [c], G. Federici [c], Y. Poitevin [d], M. Zmitko [a]

[a] *Fusion for Energy, Barcelona, Spain*
[b] *Fusion for Energy, Garching, Germany*
[c] *EUROfusion, Garching, Germany*
[d] *Fusion for Energy, St. Paul Lez Durance, France*

Europe initiated in 2014 a comprehensive design study of a DEMOnstration Fusion Reactor (DEMO) with the final objective to generate at around the middle of the century several hundred MWs of net electricity and operating with a closed tritium fuel-cycle achieving the tritium self-sufficiency.

In the European DEMO the functions of extracting thermal power and breeding tritium will be accomplished by the Breeding Blanket (BB). The strategy for the BB selection and operation in DEMO has been recently revised in the light of the strong impact of some interfacing systems (Plasma, Primary Heat Transfer System (PHTS), Remote Maintenance) on the technical BB features and layout. In such a context, a clear definition of the key design and performance requirements of DEMO BB is essential to drive a consistent design and integration process. Moreover, this allows the Test Blanket Module (TBM) Project to define and execute the best strategy for the preliminary demonstration of compliance with those requirements through the technical performance assessment. In the TBM Project the technical performance assessment is substantiated through all Project life cycle, from the conceptual design up to TBS (Test Blanket System) installation, operation in ITER and dismantling.

In this paper some high level DEMO BB requirements have been assessed against the present status and future development plan of the European TBM Project. They are:

- *Tritium Breeding Ratio:* the BB System shall ensure a TBR of 1.05;
- *Breeding Blanket Structural Material:* the DEMO BB shall use reduced activation steel, in particular EUROFER from a family of Reduced Activation Ferritic-Martensitic (RAFM) steels, in order to minimize the waste radioactivity burden;
- *Limit to tritium release:* chronic tritium release from DEMO plant to the environment shall be limited to less than 1 g/year. The BB shall allow the DEMO plant to comply with this limit;
- *Power extraction*: the DEMO BB Cooling System shall be capable of extracting the generated thermal power and transferring it to PHTS at optimal thermal-hydraulic conditions to maximize the plant efficiency while respecting the limits imposed by the structural material.

After an introductory description of the status of development of the European BB DEMO and TBM Project, the paper describes the strategy to preliminarily demonstrate the compliance to the above mentioned requirements, providing the technical performance assessment. Through the topics treated in the paper it is highlighted the importance to have a continuous and strict connection between the DEMO BB development and the execution of the TBM Project along all their phases.

Keywords: *Test Blanket Modules, DEMO breeding blanket, tritium breeding*

****Corresponding author:*** *italo.ricapito@f4e.europa.eu*

The Material Plasma Exposure eXperiment: Mission and Conceptual Design

PL-5

Juergen Rapp [a]*, Arnold Lumsdaine [a], Clyde Beers [a,b], Theodore Biewer [a], Timothy Bigelow [a], Ted Boyd [a], Juan Caneses [a], John Caughman [a], Robert Duckworth [a], Richard Goulding [a], William Hicks [a], Cornwall Lau [a], Pawel Piotrowicz [a,c], David West [a] and the MPEX team [a]

[a] *Oak Ridge National Laboratory, Oak Ridge, TN, USA*
[b] *University of Tennessee, Knoxville, TN, USA*
[c] *University of Illinois, Urbana Champaign, IL, USA*

Mastering Plasma Material Interactions (PMI) is key for obtaining a high performance, high duty-cycle and safe operating fusion reactor. Numerous gaps exist in PMI, which have to be addressed before a reactor can be built. In particular the lack of data at high ion fluence, fusion reactor divertor relevant plasma conditions and neutron displacement damage requires new experimental devices to be able to develop plasma facing materials and components. This has been recognized in the community and the U.S. fusion program is addressing this need with a new linear plasma device the Material Plasma Exposure eXperiment (MPEX).MPEX will be a superconducting linear plasma device with magnetic fields of up to 2.5 T. The plasma source is a high-power helicon source (200kW, 13.56 MHz). The electrons will be heated via Electron Bernstein Waves with microwaves using multiple 70 GHz gyrotrons (up to 600 kW in total). Ions will be heated with the ICR in the so-called "magnetic beach heating" scheme in the frequency range of 6-9 MHz (up to 400 kW in total). A short review of the plasma source development and heating science research on Proto-MPEX will be given. The results from Proto-MPEX feed directly into the design of MPEX. The conceptual design of MPEX has been completed; an overview of the conceptual design and the design requirements is given.

Keywords: *Plasma Material Interaction, Materials, Divertor, Linear Devices, DEMO*

****Corresponding author:*** *rappj@ornl.gov*

DEMO physics challenges beyond ITER

PL-6

Mattia Siccinio [a,b]*, Emiliano Fable [b], Gianfranco Federici [a], Filip Janky [b], Hanni Lux [c], Francesco Maviglia [a,d], James Morris [c], Hartmut Zohm [b]

[a] *EUROfusion Consortium, Garching bei München, Germany*
[b] *Max-Planck-Institut für Plasmaphysik, Garching bei München, Germany*
[c] *Culham Centre for Fusion Energy, Abingdon, UK*
[d] *Associazione EURATOM-ENEA, Frascati, Rome, Italy**

The most natural choice for future, electricity producing fusion reactors like EU-DEMO is to minimize the differences in terms of plasma scenario from the ITER baseline, as such scenario will rely on the largest amount of experimental evidence regarding plasmas dominated by alpha heating. Nevertheless, there are some aspects in which ITER and EU-DEMO have to differ, as the simple exercise of up-scaling from ITER to a larger device is constrained both by physical nonlinearities and by technological limits. In this work, the most significant differences between ITER and the current EU-DEMO baseline in terms of plasma scenario are discussed. Firstly, EU-DEMO is assumed to operate with a very large amount of radiation both from the scrape-off layer and, especially, from the core. This radiation is obtained by means of seeded impurities, whose presence significantly affects many aspects of the scenario itself. Secondly, because of the need to breed tritium the EU-DEMO wall is much more fragile than the ITER one. This implies that a sudden stop of the plasma discharge, for example due to the presence of a divertor reattachment event, cannot rely on massive matter injection finally triggering a disruption, but on other strategies that need to be developed. Thirdly, the ITER solution for the control of the so-called saw teeth (ST) has been shown to be too expensive in terms of auxiliary power requirements, thus other solutions have to be explored. Finally,

the problem of actively mitigating, or suppressing, the Edge Localised Modes (ELMs) has recently increased the interest in naturally ELM-free regimes (like QH-mode and I-mode) for EU-DEMO, thus deviating from the standard ELMy H-mode scenario adopted for ITER.

Keywords: EU-DEMO, ITER, plasma scenario, ELM-free regimes

**Corresponding author: mattia.siccinio@euro-fusion.org*

Recent Developments of Plasma Exhaust and Divertor Design for Tokamak DEMO Reactors

PL-7

Nobuyuki Asakura *

National Institutes for Quantum and Radiological Science and Technology (QST), Aomori, Japan

The power exhaust scenario and appropriate divertor design are crucial for all DEMO projects. Recently, the design concepts have been proposed in most DEMO designs such as Japan (JA), Europe (EU), United State, China and Korea. The divertor design concepts have been progress in the two DEMO concepts, i.e. pulse EU DEMO and steady-state JA DEMO, by using relatively conventional physics and engineering design base, which will be expected in ITER design and operation [1]. Common views have been established that the radiative cooling is a primary approach for the power exhaust, and that the total radiation power fraction in the main plasma and divertor (P_{rad}/P_{heat}, P_{heat} is the total heating power of 430-460MW) is requested to ~80% in order to reduce the peak heat load (q_{target}) lower than 10MWm^{-2} level in the detached divertor. On the other hand, power exhaust concepts are different. The former expects large radiation power in the main plasma ($f_{rad}^{main} = P_{rad}^{main}/P_{heat}$ ~0.7) using high-Z impurity seeding, in order to apply the ITER divertor design for P_{sep}/R_p = 17 MWm^{-1}. Experiment studies of the high f_{rad}^{main} promoted recently in existing devices. The latter has developed the divertor design appropriate for high P_{sep}/R_p (~30 MWm^{-1}) with increasing the divertor size, while f_{rad}^{main} is a slightly larger than ITER-level (0.40-0.45) to operate higher confinement and β_N plasma. The progress of the power exhaust concepts and issues for DEMO proposals are reviewed. Divertor simulation results of the plasma detachment and the modelling issues in the DEMO conditions are also summarized.

Concepts of the heat removal components based on the ITER technology, i.e. W-monoblock, Cu-alloy and water-cooling, and the cassette design appropriate for the remote maintenance have been integrated into the divertor design. Common baseline and R&D issues of the heat sink design, neutron irradiation and material restrictions are reviewed under the DEMO operation and maintenance conditions, as well as influences of transient heat loads on the Plasma Facing Components (PFCs). Developments of improved/reinforced PFCs and heat sink are also shown.

Physics and engineering issues of so call "advanced" magnetic configurations, divertor and heat removal proposals will be shown, compared to the conventional approaches. Different approaches of the power exhaust scenario and divertor design will provide important case studies for the future decision of the DEMO divertor design, and will help to design the fusion power plant (3GW level).

[1] N. Asakura, et al., Fusion Eng. Design, 136 (2018) 1214.

Keywords: DEMO, Power exhaust, Divertor design, Radiative cooling, Heat removal component

**Corresponding author: asakura.nobuyuki@qst.go.jp*

Privately-Funded Fusion Research

SPARC and ARC: A Compact, High-Field Pathway to Fusion Energy

IC-1

Dennis Whyte *

Massachusetts Institute of Technology (MIT) Cambridge, Massachusetts, USA

The advent of commercially available Rare Earth Barium Copper Oxide (REBCO) superconductors enables a new strategic pathway to fusion energy based on high-field, compact tokamaks. A high-field REBCO D-T tokamak, SPARC, can obtain net energy gain at very small size (R~1.65 m), retiring key scientific (burning plasma) and technical risks (superconductor magnets) at a small enough scale that it can be funded solely by the private sector. The new magnets then enable ARC, a pilot-plant net electricity tokamak which is similar in size to JET. We will describe the technology innovations enabled by REBCO including demountable coils, advanced diverters, an immersion liquid blanket and the use of FLiBe as a coolant/breeder. Ongoing research activities toward SPARC and ARC will be discussed.

**Corresponding author: whyte@psfc.mit.edu*

Unlocking Fusion Energy Through Advanced Beam-Driven FRCs

IC-2

Michl W. Binderbauer

TAE Technologies, Rancho Santa Margarita, CA, USA

In this talk, we will share the latest achievements in plasma stability and confinement on TAE's fifth generation experimental platform C-2W a.k.a. Norman for the ultimate goal of developing and distributing clean, safe, and economic fusion power. We will also explain how the company's frontier science and technological advances impact innovation across industries including electric mobility, life sciences and grid level power management.

***Corresponding author:** michl@tae.com*

Oral Presentations

Test Blanket Modules (ITER) and Breeding Blanket (DEMO): History of Major Fabrication Technologies Development of HCLL and HCPB and Status

01-1.1

Laurent Forest [a]*, Lorenzo Virgilio Boccaccini [b], Laurence Cogneau [c], Antonella Li Puma [a], Heiko Neuberger [b], Jörg Rey [b], Noel Thomas [d], Jérôme Tosi [a], Milan Zmitko [e]

[a] *DEN-Service d'Études Mécaniques et Thermiques (SEMT), CEA, Université Paris-Saclay, Gif sur Yvette, France*
[b] *KIT, Karlsruhe Institute of Technology, Karlsruhe, Germany*
[c] *DRT-Laboratoire d'Innovation pour les Technologies des Energies Nouvelles (LITEN), CEA, Grenoble, France*
[d] *ATMOSTAT, Villejuif, France*
[e] *Fusion for Energy (F4E), Barcelona, Spain*

Two breeding blanket concepts, the Helium-Cooled Lithium-Lead (HCLL) and the Helium-Cooled Pebble-Bed (HCPB), developed in Europe in the frame of DEMO and relevant TBM studies, present many similarities in terms of design and manufacturing. As an example, all structure subcomponents are internally cooled by helium circulating in meandering squared section channels. Several technologies have been investigated in the frame of EU fusion research programme for the manufacturing of blanket and TBM sub-components (Side Caps (SCs), Stiffening Plates (SPs), First Wall (FW)) manufacturing (machining, welding and control) and their assembly, main of which are discussed in this paper. The manufacturability of DEMO blanket modules Back Supporting Structure (BSS), a big structure located behind blanket modules and supporting them, is also discussed. The applicability of technologies takes into account specificities of EUROFER steel, foreseen as structural material.

For each assessed technology, main results, e.g. in terms of mechanical properties and microstructure of weld joints, are presented and main advantages and drawbacks are summed up, in order to identify most promising technology/ies and to propose manufacturing scenarios. It should be noted that these developments are performed according to standards and professional codes (RCC-MRx). Further development strategies are also briefly discussed in the paper.

Keywords: *Test Blanket Module, Breeding Blanket, EUROFER, Manufacturing*

***Corresponding author:** laurent.forest@cea.fr*

Development Status of Helium Cooled Ceramic Breeder Tritium Breeding Blanket in China

01-1.2

Xiaoyu Wang [a]*, D.L. Luo [b], M.Z. Lei [c], H.L. Chen [d], S.M. Liu [c], F.C. Zhao [a], X.H. Wu [a], X.F. Ye [a], Y. Yao [b], Y.J. Feng [a], Q.X. Cao [a], B. Zhou [a], F. Wang [a], H.X. Zhang [a], H.B. Liao [a], B.P. Gong [a], X.R. Duan [a], M. Xu [a], Q. Sheng [e], K.H. He [e] and Chinese HCCB TBB Team

[a] *Southwestern Institute of Physics, Chengdu, China*
[b] *China Academy of Engineering Physics, Mianyang, China*
[c] *Institute of Plasma Physics, Chinese Academy of Science, Hefei, China*
[d] *University of Science and Technology of China, Hefei, China*
[e] *China International Nuclear Fusion Energy Program Execution Center, Beijing, China*

As one of the key components in fusion reactor, the tritium breeding blanket (TBB) is indispensable to realize the functions of fusion power generation and tritium self-sufficiency. But the development of tritium breeding blanket still faces a lot of technical challenges before engineering utilization.

In order to meet these technical challenges, the development strategy of the helium cooled ceramic breeder (HCCB) tritium breeding blanket has been established in China. Firstly China has signed the Helium Cooled Ceramic Breeder Test Blanket Module Agreement (HCCB TBMA) with ITER organization to fabricate and test the HCCB TBM and its ancillary system (totally HCCB TBS) in ITER facility, which provides the chance to validate the basic TBB technologies, such as heat removal, tritium extraction, coolant purification, system integration, safety and so on, also provide the experience and data for the development of CFETR TBB. This is the first step toward the future HCSB TBB for CFETR and DEMO. Secondly, China has set up the domestic research and development plan for the preliminary engineering design and R&D for the China Fusion Engineering Test Reactor (CFETR) that is the next device in the roadmap for the realization of fusion energy in China. As the key component of CFETR, the development of HCCB TBB is also supported by this plan for the design, R&D and construction of test facility, which will provide more contribution for the TBB technology development at reactor scale, such as materials, fabrication, tritium handling, maintenance, system integration, safety and so on. This will bridge the gaps between ITER TBM and the TBB in DEMO reactor in the next step.

The conceptual design of HCCB TBS has been approved by ITER Organization in 2015. After that, the design optimization and more R&D activities are under implementation for the preliminary design. Currently the design of TBM-set was updated basing on the R&D of material and fabrication technology to ensure the manufacturability and structure reliability, also the structure integrity was verified by simulation and experiment. The design of ancillary subsystems was optimized considering the flow process optimization, system performance and structure integrity under ITER operation condition. As the structural material, two kinds of reduced activation ferritic/martensitic (RAFM) Steel (CLF-1 steel and CLAM steel) have been developed and reached the 5 tons ingot scale. The fabrication processes of the lithium silicate pebble, as tritium breeder, and the beryllium pebble, as neutron multiplier, have been developed. The semi-prototype of TBM has been fabricated to verify the manufacturing process. Several test platforms for the breeding blanket technology development have been constructed and started experiments to test components, processes and get the operation data.

The pre-engineering design and R&D of TBB for CFTER were started from 2017 under the cooperation of several domestic institutes and universities, and its most design features and material selection are similar to HCCB TBM. The preliminary profile of CFETR TBB has been obtained after iteration with plasma physics. Recently the design of TBB focuses on the detail design optimization considering the tritium breeding ratio (TBR) and shielding performance. The integration design with vacuum vessel and the compatibility assessment with remote handling are going on. The R&D for fabrication technology firstly focus on the connection of tungsten with CLF-1 steel and fabrication process of First Wall (FW). Also the support for the development of advanced material and its irradiation has been considered by government. The several test facilities, such as pebble bed test facility and helium cooling experimental loop (HeCEL-3), are under design and will start construction in next two year.

The development strategy of HCCB TBB is one of the most important part of China fusion development toward DEMO. The design and R&D of HCCB TBS is being implemented based on the ITER schedule and it is providing the indispensable experience for the design of CFETR TBB from both successful and unsuccessful aspects.

***Corresponding author:** *wangxy@swip.ac.cn*

Nuclear Responses of WCCB TBM with Different Container Designs

01-1.3

Hyoseong Gwon *, Wenhai Guan, Kentaro Hattori, Hiromasa Iida, Hisashi Tanigawa, Takanori Hirose, Yoshinori Kawamura

National Institutes for Quantum and Radiological Science and Technology, Aomori, Japan

Test blanket module (TBM) will be installed into an equatorial port in ITER. In TBM program blanket major functions, which are tritium production, heat extraction for electric power production and neutron shielding, will be demonstrated under fusion reactor environments in ITER. A box-shaped container had been applied to water cooled ceramic breeder (WCCB) TBM design. The container design of WCCB TBM has been changed recently from the box-shaped to a cylindrical which has outstanding pressure tightness for water ingress into the container (In-Box LOCA). For the different designs, we investigated nuclear responses of TBMs including tritium production rate (TPR), nuclear heating, and neutron activation. The nuclear properties were analyzed by using Monte Carlo code (MCNP5.1.60) and the nuclear data library for fusion neutronics applications (FENDL-2.1).

TPR in the box-shaped TBM was two times smaller than that in the cylindrical TBM. Container wall thicknesses were decreased by introducing the cylindrical shape. The decrease in the container wall thicknesses led to increasing TPR in the cylindrical TBM. An increase in TPR caused an increase in nuclear heating of breeder material (Li_2TiO_3). The nuclear heating of Li_2TiO_3 in the cylindrical TBM was twice larger than that in the box-shaped TBM. The total nuclear heating of the cylindrical TBM was smaller than that of the box-shaped TBM because the nuclear heating of structural material (F82H steel) was dominant to the total nuclear heating. The weight and the corresponding nuclear heating of structural material (F82H) in the cylindrical TBM is smaller than the box-shaped TBM. Differences in neutron activation were also discussed based on analyses of neutron spectrum.

Keywords: *Monte Carlo method, blanket design, tritium production rate, nuclear heating, neutron activation*

***Corresponding author:** *gwon.hyoseong@qst.go.jp*

Conceptual Design and Analysis of the HCCR Breeding Blanket for K-DEMO with Newly Adopted Multiple Enrichment Breeder

01-1.4

Sunghwan Yun [a]*, Seong Dae Park [a], Dong Won Lee [a], Cheol Woo Lee [a], Hyung Gon Jin [a], Chang Wook Shin [a], Suk-Kwon Kim [a], Jae Sung Yoon [a], Mu-Young Ahn [b], Seungyon Cho [b]

[a] *Korea Atomic Energy Research Institute, Daejeon, Korea*
[b] *National Fusion Research Institute, Daejeon, Korea*

Helium-cooled ceramic reflector (HCCR) blanket has been studied as a candidate breeding blanket for the Korean DEMO reactor. A detailed three-dimensional neutronics model based on the K-DEMO model and a simplified thermal-hydraulic model were established for the HCCR-DEMO blanket conceptual design analysis. Multiple enrichment concept was newly employed to satisfy both of the temperature limitation of functional materials and the TBR requirement for DEMO fusion power reactor. The insertion of graphite reflector resulted more than 50% reduction of the required total beryllium multiplier volume, while its influence to the tritium breeding ratio (TBR) was evaluated less than 5%. The final overall TBR of the HCCR-DEMO blanket was estimated as more than 1.15.

Keywords: *Helium Cooled Ceramic Reflector (HCCR), Breeding blanket, DEMO, multiple enrichment*

Corresponding author: *syun@kaeri.re.kr*

Impact of Plasma Thermal Transients on the Design of the EU DEMO

01-2.1

Francesco Maviglia [a,b]*, Christian Bachmann [a], Gianfranco Federici [a], Thomas Franke [a,c], Mattia Siccinio [a,c], Christian Vorpahl [a], Raffaele Albanese [d], Roberto Ambrosino [d], Emiliano Fable [c], Mehdi Firdaous [e], Jonathan Gerardin [e], Vincenzo Paolo Loschiavo [d], Massimiliano Mattei [d], Francesco Palermo [c], Maria Lorena Richiusa [f], Fabio Villone [d], Zsolt Vizvary [f]

[a] *EUROfusion – Programme Management Unit, Garching, Germany*
[b] *Associazione EURATOM-ENEA sulla Fusione, Frascati, Rome, Italy*
[c] *Max-Planck-Institut flr Plasmaphysik, Garching, Germany*
[d] *Consorzio CREATE, Univ. Napoli Federico II - DIETI, Napoli, Italy*
[e] *CEA, St. Paul-Lez-Durance, France*
[f] *CCFE, Culham Science Centre, Abingdon, UK*

The present European DEMO design steady state heat load capability of the Breeding Blanket (BB) First Wall (FW) is limited to ≈1 MW/m^2, for both the considered helium cooled and water cooled concepts, due to the specific requirements on high neutron irradiation resistant materials, high coolant temperature for efficient energy conversion, and tritium breeding. While this limit is achievable in nominal conditions in the present DEMO BB concept designs, the greatest challenges arise from the occurrence of plasma transients for both normal (e.g. plasma ramp-up), and off-normal events, (e.g. vertical displacement events, disruptions or sudden loss of confinement exceeding the control system capability) leading to a plasma-wall contact. A strategy to mitigate the risk of damaging the main in-vessel components is being developed, considering the inclusion of discrete heat flux limiters able to provide protection of the FW, and designed to maintain the integrity of the limiter cooling system itself during transients. the present investigations include 2D/3D electromagnetic modelling and plasma simulations on a list of critical transient events, based on synthetic and experimental data from several machines. The heat flux calculation with 3D field-line tracing codes is used to determine the design and the required number of limiters. Also, the plasma equilibrium is optimized to design the preferred plasma-wall contact locations to be close to maintenance ports, (e.g. outer equatorial plane, or upper vertical port), when possible, so to replace more easily the limiters in case of failure. The limiters are ultimately considered as sacrificial components and are designed to survive a number of transient events, depending on the power density and deposition time, and are designed to be replaceable independently of the BB.

Keywords: *DEMO, first wall load, electromagnetic simulations, plasma scenario optimization*

Corresponding author: *francesco.maviglia@euro-fusion.org*

Recent Progress in the Design of the K-DEMO Divertor

01-2.2

Sungjin Kwon [a]*, Kihak Im [a], Suk-ho Hong [a], Hyungho Lee [a], Thomas Rognlien [b], William Meyer [b], Keeman Kim [a]

[a] *National Fusion Research Institute, Daejeon, Korea*
[b] *Lawrence Livermore National Laboratory, Livermore, CA, USA*

The preliminary conceptual design of the Korean fusion demonstration reactor (K-DEMO) with a major radius of 6.8 m has been studied since 2012 [1]. The overall configuration of the K-DEMO divetor system based on the ITER-like water-cooled tungsten technology is a double-null type symmetric divertor subdivided into 32 toroidal modules for the vertical maintenance. A detached divertor scenario with impurity seeding was considered as the primary approach for the power exhaust to reduce the peak heat flux lower than the engineering limit of 10 MW/m^2. The power exhaust performance at the scrap off layer was estimated by using UEDGE-2D code a two-dimensional fluid transport code for collisional edge plasma and neutral species. Particle and heat flux on inboard and outboard divertor target were calculated for the

detached cased depending on parameters such as the impurity seeding rate, pumping rate, and the pedestal density.

Based on the physical parameters obtained by the model calculation, engineering analyses were carried out. Thermal stability was estimated by the thermo-hydraulic analysis when the case of the peak heat flux was set to 10 MW/m^2 on the outboard divertor target. The optimal tungsten monoblock designs were derived for two candidates of heat sink materials: the reduced activation ferritic martensitic (RAFM) steel and CuCrZr alloy. The design issues caused by the low thermal conductivity of RAFM steel and the activation of CuCrZr alloy were discussed. Meanwhile, electromagnetic (EM) analysis was carried out to estimate the EM loads caused by the abnormal behaviors of plasma such as major disruption and vertical displacement event, since EM loads are one of the most important external loads for designing a DEMO divertor. The results of structural analysis under EM loads and thermal stress will be presented.

[1] K. Kim et al., "Design concept of K-DEMO for near-term implementation," Nuclear Fusion 55, 053027 (2015).

Keywords: *K-DEMO, divertor, edge plasma analysis, electromagnetic analysis, thermal analysis*

****Corresponding author:*** *sjkwon@nfri.re.kr*

US Enabling Technology for Fusion

01-2.3

Richard E. Nygrena*

Sandia National Laboratories, Albuquerque, New Mexico, USA

The US is performing an extensive community-wide planning study under the auspices of the American Physical Society Division of Plasma Physics (APS-DPP). The goal is a consensus view of the path forward for US fusion R&D and includes all fusion programs in the US Department of Energy's Office of Fusion Energy Sciences. As part of the core team in this Community Planning Process (CPP), the author is leading an activity to characterize the technologies where significant advances, e.g., in high-field high-temperature superconducting magnets and in advanced manufacturing, have the potential to enable an R&D path for simpler and less expensive fusion power. The paper will review and summarize these technologies.

Keywords: *plasma facing components, enabling technology,*

****Corresponding author:*** *renygre@sandia.gov*

The Design of the Divertor Remote Handling System for ITER

01-2.4

Carine Van Hille [a]*, Carlo Damiani [a], Ottó Bede [b], *et al.*

[a] Fusion for Energy, Torres Diagonal Litoral B3, Josep Pla 2, 08019 Barcelona, Spain

[b] ITER Organization, Route de Vinon sur Verdon, 13115 Saint Paul Lez Durance, France

Remote maintenance is among the novel challenges posed by ITER, due to the combination of high level of gamma radiation, space constraints, size and weight of the components to be handled, and required millimetric accuracy.

This is particularly true in the case of the divertor, segmented in 54 cassettes, whose partial or global replacement could occur due to damages localized to a limited number of cassettes or wide spread across the whole divertor, respectively.

This paper will present the new, detailed design of the ITER divertor remote handling system (DRHS): the divertor maintenance scheme, the RH devices (Cassette Multifunctional Mover, Cassette Toroidal Mover and related end effectors and manipulator arms, pipe isolator and tooling, control system, key technologies on board, etc.) and the main design issues that will be addressed in the next stage of the design. This will be complemented by the results of the support R&D activities, including trials on full scale cassette mock-ups and other ongoing technological developments. Finally, it will give an overview on the next phase of design, and on subsequent manufacturing and testing phases, including delivery to the ITER site.

Keywords: *ITER, Divertor, Remote Maintenance*

****Corresponding author:*** *Carine.VanHille@f4e.europa.eu*

Heating and Current Drive Systems for DEMO

01-3.1

Minh Quang Tran *, on behalf of the whole WPHCD team

Ecole Polytechnique Federale de Lausanne (EPFL), Swiss Plasma Center (SPC), Lausanne, Switzerland

In the European Fusion Road Map [1], DEMO is the single step between ITER and a Fusion Power Plan (FPP). DEMO main high level requirements encompass two points which strongly influence the design of the Heating and Current Drive (HCD) systems: the self sufficiency of Tritium and the supply of electricity to the grid. The first requirement implies the minimization of the opening in the breeding blankets. The second one will lead to a careful consideration on RAMI (Reliability, Availability, Maintanability, Inspectability), which, for HCD systems, needs to be developed. The overarching consideration is naturally the fulfillments of the physics requirements during all the plasma phases, from the breakdown through the flat- top phase until the controlled ramp down of the plasma current with the necessary control of the magneto-hydrodynamics instability.

Since the beginning of EUROfusion, the Work Package Heating and Current Drive (WPHCD) is developing three main HCD systems, using electron cyclotron (EC), ion cyclotron (IC) waves and neutral beam injection (NBI). The design and development of these systems are fully compliant with the design approach of DEMO: the technical design must fully meet the requirements of DEMO, such as the neutronic shielding of all DEMO components, the impact on the Tritium breeding ratio, maintenance concepts and the integration into the DEMO machine baseline. While less important than for a FPP, the efficiency of the systems (from plug to plasma) is also an element for consideration in the programme to avoid having too high recirculating or grid power. In the present phase, DEMO is still in the pre-conceptual phase and WPHCD expertise is being brought to the physics design of DEMO concepts.

The presentation will describe in details the results of R&D performed on the 3 HCD systems. For all of them, we shall present the concepts, which have been selected in order to fulfill both the physics requirements and their integration into DEMO (the necessary penetration - and the corresponding neutronic influence - into the machine and their impact on the various buildings). The talk will outline the antenna concepts for EC and IC systems and the NBI system. For EC, the R&D also encompasses the gyrotron, the present requirements of which do not match the ones for DEMO. These results will serve as basis for the pre-conceptual design review of DEMO, foreseen in 2020.

[1] T. Donné, Journal of Instrum. 2017, Vol. 12, https://doi.org/10.1088/1748-0221/12/10/C10008.

Keywords: *Heating systems, Electron cyclotron wave system, Ion cyclotron wave system, Neutral Beam Injection*

***Corresponding author:** *minhquang.tran@epfl.ch*

Achievement of the ITER NBI Ion Source Parameters for Hydrogen at the Test Facility ELISE and Present Status for Deuterium

01-3.2

Ursel Fantz *, D. Wünderlich, R. Riedl, B. Heinemann, F. Bonomo and the NNBI-Team

Max-Planck-Institut für Plasmaphysik, Garching, Germany

The two neutral beam injection (NBI) systems for heating and current drive at ITER are based on generation and acceleration of negative ions and are designed to inject up to 16.5 MW per beamline of either 870 keV H or 1 MeV D atoms for up to one hour. Due to limited neutralization efficiency and losses in the beamline the required accelerated ion currents are 46 A for H− and 40 A for D− generated in an RF-driven ion source at 0.3 Pa and accelerated in a 7-stage accelerator system.

The ELISE test facility is an integral part of the roadmap towards the achievement of the ITER requirements. ELISE went into operation in 2013 with the goal to demonstrate with an ion source of half the ITER-size (1×1 m2 source area with 640 apertures for extraction) the ITER requirements with respect to the extracted current densities: 329 A/m2 for H^- and 286 A/m2 for D^- at tolerable amount of the co-extracted electrons (less than one) and with a beam uniformity better than 90%.

After approaching in short pulses (10 s) the required current densities for H- and for D-, focus was laid on the extension of the pulse lengths. Extremely challenging is the achievement of a temporal stability of the co-extracted electron currents. Recently the target parameters for hydrogen were demonstrated for the required length of 1000 s representing a milestone for the success of ITER NBI. Prerequisite for this performance were the introduction of external permanent magnets and internal potential rods as well as a dedicated caesium conditioning technique, which also allowed achieving good reproducibility of this high performance without the need of lengthy re-conditioning phases between the pulses. As for deuterium the amount of co-extracted electrons is higher with stronger temporal dynamics than in hydrogen, they limit the source performance and further suppression and control is mandatory to achieve the target parameters for deuterium beams.

Keywords: *Neutral Beam Injection, ITER NBI, ELISE, Negative Ion Source*

***Corresponding author:** *ursel.fantz@ipp.mpg.de*

Testing and Analysis of Steady-State Helicon Plasma Source for the Material Plasma Exposure eXperiment (MPEX)

01-3.3

Arnold Lumsdaine [a*], Joseph Tipton [b], Saikat Chakraborty Thakur [c], Juan Caneses [a], Richard Goulding [a], Dean McGinnis [a], Michael Simmonds [c], George Tynan [c], Juergen Rapp [c], John Burnett [d]

[a] *Oak Ridge National Laboratory, Oak Ridge, TN, USA*
[b] *Lipscomb University, Nashville, TN, USA*
[c] *University of California, San Diego, San Diego, CA, USA*
[d] *MPF Products, Inc., Gray Court, SC, USA*

Preparing for next step fusion facilities will require developing materials that can withstand the high ion and neutron fluences that will be present in the divertor region. These fluences are inaccessible in current toroidal devices. The Material Plasma Exposure eXperiment (MPEX) is a steady-state linear plasma device that is currently undergoing conceptual design that proposes to reach ion fluences as high as 10^{31} m-

[2]. It will also be able to receive neutron irradiated samples to examine the multivariate effects of plasma material interactions. A target exchange chamber will be employed so that the MPEX target can be removed and placed in a separate diagnostic station without leaving vacuum. In order to operate in steady-state, the MPEX plasma will be confined using superconducting magnets, with active cooling for all plasma interacting and plasma facing components. The plasma source will be a high power (200 kW) helicon antenna, which will be placed outside of the vacuum chamber. The window for this antenna must be water cooled, and have a very low dielectric coefficient to limit the dielectric losses. The water-to-vacuum seal should not be an elastomer seal to limit impurities at the plasma source. It is proposed to use a ceramic-to-metal joint. A prototype water-cooled helicon antenna window and assembly have been manufactured and tested in long pulse conditions up to 10 kW in the Controlled Shear De-correlation eXperiment (CSDX) at the University of California, San Diego. Thermal results have been correlated with computational fluid dynamics (CFD) simulation. The heat fluxes are extrapolated to 200 kW to confirm that the design is compatible with 200 kW power.

Keywords: *plasma materials interaction, plasma source, linear plasma facilities*

Corresponding author: *lumsdainea@ornl.gov*

Modeling for MHD Issues Related to Liquid Metal Fusion Divertor and Blanket

01-3.4

Ming-Jiu Ni [a,b]*, Long Chen [a], Jun-Hua Pan [a], Jie Zhang [b]

[a] *University of Chinese Academy of Sciences, China*
[b] *Xi'an Jiaotong University, China*

In this presentation, a consistent and conservative scheme will be briefly reviewed for simulation of liquid metal MHD flows with good accuracy at a high Hartmann number. Then, a solver is well developed for MHD flows in a prototypical liquid metal fusion blanket with special focuses on MHD thermal convections and MHD turbulences. Also, we are interested in modeling of free surface MHD flows, for which the lithium film flow will be carefully simulated for possible application in a liquid metal fusion divertor. Series of simulations for MHD flows in a liquid metal fusion blanket and in a fusion divertor will be presented.

Corresponding author: *mjni@ucas.ac.cn*

The Frontier Investigations of the Liquid Breeders Using Oroshhi-2 Heat and Mass Transfer Loop

02-1.1

Juro Yagi [a]*, Teruya Tanaka [b], Akio Sagara [b], Takeo Muroga [b]

[a] *Institute of Advanced Energy, Kyoto University, Kyoto, Japan*
[b] *National Institute for Fusion Science, Gifu, Japan*

The Fusion Engineering Research Project (FERP) of NIFS is widely conducting studies on fundamental technologies for high performance, long operating life and safe liquid blanket systems under collaboration with universities. In 2014, the FLiNaK/Pb-Li twin-loop system Oroshhi-2 (Operational Recovery Of Separated Hydrogen and Heat Inquiry-2) with a 3 T superconducting magnet has been constructed in NIFS as a "real function" integrated test stand for all the liquid blanket technologies except for neutron irradiation [1]. At present, Oroshhi-2 is under operation as a center of collaborative liquid blanket studies in Japan. For examples, MHD drag experiment of double bended Pb-Li pipe up to 3T [2] and heat transfer experiment in a liquid gallium alloy film flow in the magnet have been successfully completed with the group of Prof. Kunugi (Kyoto Univ.). Using the FLiNaK loop, a heat transfer measurement / enhancement experiment up to 3T and material corrosion test in 1T permanent magnet channel are just under preparation with the group of Prof. Ebara (Tohoku Univ.) and Dr. Yagi (Kyoto Univ.), respectively. In the Pb-Li loop, a hydrogen isotope recovery experimental setup based on the vacuum sieve tray concept [3] is under construction with the group of Prof. Konishi (Kyoto Univ.), which has been confirmed the performance by an offline test. SiC/SiC_f composite compatibility experiment is also under preparation with the group of Dr. Bringuier (General Atomic) using a composite cylindrical tube operating around 873 K.

These recent works and plans for the experiments in Oroshhi-2 are introduced in the conference.

[1] A. Sagara et al, Fusion Sci. Tech., 68 (2015) 303.

[2] S. Nakamura et al., Fusion Eng. Des., 136 (2018) 17.

[3] F. Okino et al., Fusion Eng. Des. 109-111 (2016) 1748.

Keywords: *Heat and mass transfer loop, Molten salt, Lead-lithium eutectic alloy, Real function experiment*

Corresponding author: *j-yagi@iae.kyoto-u.ac.jp*

Overview of Russian Activity on ITER Blanket Procurement Arrangement

O2-1.2

Andrey Yu. Leshukov [a]*, Yu.S. Strebkov [a], S.Yu. Kirillov [a], S.V. Makarov [a], M.N. Sviridenko [a], I.V. Danilov [a], A.V. Razmerov [a], E.V. Parshutin [a], S.E. Khomyakov [a], V.Yu. Kolganov [a], I.I. Poddubniy [a], S.N. Tomilov [a], I.V. Mazul [b], A.A. Gervash [b], V.M. Safronov [c], A.B. Putrik [c]

[a] *Joint-Stock Company "N.A. Dollezhall Research and Development Institute of Power Engineering", (JSC"NIKIET"), Moscow, Russian Federation*

[b] *JSC"NIIEFA" (D.V. Efremov Institute), St. Peterburg, Russian Federation*

[c] *Institution "Project Center ITER", Moscow, Russian Federation*

According to the ITER blanket-relevant Procurement Arrangements (PA) which were signed by Russian Federation and ITER Organization in 2014 the JSC "NIKIET" is an Official Supplier of Enhanced Heat Flux (EHF) First Wall (FW) Panels(1.6.P1ARF.01 "Blanket First Wall") and blanket module connectors: flexible supports, electrical insulating key pads, shield block/vacuum vessel electrical straps(SB/VV ESs) and W bimetallic pedestals for electrical strap assemblies (1.6.P3.RF.01 "Blanket Module Connections").

Enhanced Heat Flux (EHF) First Wall (FW) Panels(179 items) are intended to withstand the heat flux from plasma up to 4.7 MW/m2, and there are two institutions in Russian Federation responsible for the manufacturing, testing and delivering of these panels on the ITER site: JSC "NIIEFA" (Efremov Institute) and JSC "NIKIET".JSC "NIIEFA" (Efremov Institute) will manufacture the plasma-facing components (PFC) of EHF FW Panels and perform the final assembling of the panels while JSC "NIKIET" will manufacture the FW beam structures, load-bearing structures of PFC, mechanical attachment system on the Shield Block (SB) and FW/SB electrical strap assemblies.

As for the PA 1.6.P3.RF.01 "Blanket Module Connections"the JSC "NIKIET" is the alone official Supplier and will manufacture and procure the following blanket attachment components: flexible support assemblies (2109 items), electrical insulating key pads (2336 items), shield block/vacuum vessel electrical straps (1052 items) and W bimetallic pedestals for SB/VV ESs (1052 items).

Both the Procurement Arrangements are now on the implementation of Phase III "Processes Qualification" and the progress on the qualification of critical technological processes (welding, deposition of electrical insulating and anti-seizing/anti-friction coatings, bimetallic joining techniques, etc.) is considered in details in this overview.

This article also briefly describes the joint activity of JSC "NIKIET" and Efremov Institute in the framework of 1.6.P1ARF.01 "Blanket First Wall" Procurement Arrangement and the activity on 1.6.P3.RF.01 "Blanket Module Connections". The following achievements on both Russian blanket-relevant PAs realization during the period of 2018–2019 are presented: qualification results, issued pre-production documentation, the results of design and analysis activities and experimental validation.

Keywords: *blanket, in-vessel components, first wall panels, connectors, experimental validation, technological processes*

Lithium-6 Enrichment Technology Using Innovative Electrodialysis with Lithium Ionic Conductor

O2-1.3

Tsuyoshi Hoshino *, Kenji Morita, Makiko Sekiya

National Institutes for Quantum and Radiological Science and Technology (QST), Aomori, Japan

The tritium as a fuel for fusion reactors is produced by the neutron capture reaction of lithium-6 (^{6}Li). However, natural Li contains only about 7.8 % ^{6}Li, and the enrichment of ^{6}Li up to 90% is required for the fusion reactor. In previous study, a new lithium isotope separation technique using the lithium ionic conductor as a separation membrane was established. While lithium ions can move through the lithium ionic conductor by electrodialysis, the higher mobility of ^{6}Li ions due to its lighter mass than that of ^{7}Li ions enables ^{6}Li to be enriched on the cathode side. Principle demonstration was completed. Then a long-term evaluation test of the lithium-6 enrichment was performed as a next step.

The new method involves the use of $Li_{0.29}La_{0.57}TiO_3$ (LLTO) as an Li isotope separation membrane (LISM) whereby only Li ions permeate from the positive electrode side to the negative electrode side during electrodialysis. The area and thickness of the LISM are 25 cm^2 (5.0 cm × 5.0 cm) and 0.5 mm, respectively. The positive side of the dialysis cell was filled with 0.1M LiOH solution. With the electrode area being set to 16 cm^2, the relationship between the ^{6}Li separation coefficient and the electrodialysis time was investigated by ICP-AES and ICP-MS.

After 132 days electrodialysis, we obtained a maximum of 1.06 for the ^{6}Li isotope separation coefficient. This result showed that the ^{6}Li isotope separation coefficient of this method is the same as that of the amalgamation process using mercury (1.06). Thus, this method has the potential to be a superior ^{6}Li enrichment method to produce 90% enriched tritium breeder for fusion reactors.

Keywords: *Lithium Isotope Separation, ^{6}Li, Lithium Ionic Conductor, Lithium Isotope Separation Membrane, Electrodialysis*

***Corresponding author:** *hoshino.tsuyoshi@qst.go.jp*

Perspectives on the FESAC Transformative Enabling Capabilities: Priorities, Plans and Status

O2-1.4

Arnold Lumsdaine [a]*, Rajesh Maingi [b]

[a] *Oak Ridge National Laboratory, Oak Ridge, TN, USA*
[b] *Princeton Plasma Physics Laboratory, Princeton, NJ, USA*

In early 2017, the Fusion Energy Sciences Advisory Committee (FESAC), and advisory committee of the United States Department of Energy, was charged with identifying transformative enabling capabilities (TECs) that could that could promote efficient advance toward fusion energy, building on burning plasma science and technology." A subcommittee with broad expertise was formed and sought feedback from scientific experts, including experts from outside of the fusion community. Three workshops were conducted, and a report was approved by FESAC in 2018 that identified four of the "most promising" TECs: Advanced algorithms, high critical temperature superconductors, advanced materials and manufacturing, and novel technologies for tritium fuel cycle control. In addition, one "promising" TEC was identified: fast flowing liquid metal plasma facing components. This presentation will give details on the most promising TECs, and give an overview on considerations of these TEC in in the US since the publication of the report.

Keywords: *plasma materials interaction, plasma source, linear plasma facilities*

****Corresponding author:*** *lumsdainea@ornl.gov*

Standardization of Eurofer Material, a First Step Toward Industrialization

O2-1.5

Thierry Lebarbé *, Cécile Petesch, Jorge Muñoz Garcia

DEN-Service d'Études Mécaniques et Thermiques (SEMT), CEA, Université Paris-Saclay, Gif-sur-Yvette, France

The diversity and the innovative characteristic included in the fusion technology are in opposition with a codification status, as code and standards are based on the recognized industrial feedback. By definition, no code exists for a real innovative system.

The extension of the scope of a code or a standard to innovative systems such as fusion reactors leads to revisit the background of the code to define the requirements to introduce a new process or a new material.

The developed methodology has been applied for the introduction of the X10CrWVTa9-1 steel (Eurofer), which is today in the Probationary Phase Rules of RCC-MRx. It was the first time that a "new" material was introduced into the code, new in the sense of non-existing in any current standardization. This process, still in progress, highlights the need to have a minimum of information on the expectation of the code regarding the material data.

This paper describes the different steps of the introduction of the Eurofer in the RCC-MRx code as well as the tools developed to facilitate the process.

Keywords: *standardization, Eurofer, Fusion, RCC-MRx*

****Corresponding author:*** *Thierry.lebarbe@cea.fr*

Current Status and Technical Issues of Welding and Joining Technologies of Reduced Activation Ferritic/Martensitic Steel F82H

O2-1.6

Hiroyasu Tanigawa [a]*, Takanori Hirose [a], Masami Ando [a], Dai Hamaguchi [a], Hiroaki Mori [b], Hisashi Serizawa [c], Yoshiaki Morisada [c], Hidetoshi Fujii [c], Takashi Nozawa [a], Yutai Katoh [d]

[a] *National Institutes for Quantum and Radiological Science and Technology, Rokkasho, Aomori, Japan*
[b] *Department of Management of Industry and Technology, Graduate School of Engineering, Osaka University, Suita, Osaka, Japan*
[c] *Joining and Welding Research Institute, Osaka University, Ibaraki, Osaka, Japan*
[d] *Oak Ridge National Laboratory, Oak Ridge, TN, USA*

Welding and joining technologies are the essential technologies in the development of reduced activation ferritic/martensitic (RAFM) steels as the structural material of fusion in-vessel components. F82H (Fe-8Cr-2W, V, Ta) is the RAFM steel which has been developed and studied in Japan was designed to have excellent heat resistance, toughness, and weldability in the first place. This work summarizes R&D activities on welding and joining technologies of F82H including its irradiation response.

The applicability of electron-beam (EB), tungsten-inert-gas (TIG), and fiber laser welding technologies have been demonstrated on various thickness of F82H. Neutron irradiation on TIG weld metal (WM) up to 5dpa at 300 °C showed a slightly larger hardening and loss of ductility compare to that of base metal (BM), but elastoplastic fracture toughness of TIG WM at 300 °C significantly lost after irradiation compared to that of BM. Hot and cold cracking sensitivities of F82H were examined, and it was indicated that F82H cracking sensitivities were the same or better than those found in commercial stainless steels. Friction stir welding successfully joins F82H thin plates without inducing phase transformation and showed a less hardening compared to that of BM after the Fe ion irradiation up to 20 dpa at 300 °C.

Keywords: *F82H, weld, joint, irradiation effect*

**Corresponding author:* *tanigawa.hiroyasu@qst.go.jp*

System Studies of Spherical Tokamaks

02-1.7

Stuart I. Muldrew [a*], Hanni Lux [a], Tim C. Hender [a], Bhavin Patel [a,b], Peter J. Knight [a], Garry M. Voss [a], Howard R. Wilson [a,b]

[a] *Culham Centre for Fusion Energy, UK Atomic Energy Authority, Culham Science Centre, Abingdon, Oxfordshire, UK*
[b] *Department of Physics, York Plasma Institute, University of York, Heslington, York, UK*

Spherical Tokamaks offer a number of potential advantages for a future fusion power plant. They have a high ratio of thermal to magnetic field pressure (beta) and strong flows, either of which could result in reduced turbulence. Fewer Toroidal Field (TF) coils and a different geometry offers the potential for new methods of remote maintenance and lower magnet costs. The UK Atomic Energy Authority has long had an interest in spherical tokamaks having hosted START, MAST and now MAST-U.

Systems codes can be employed to scope out parameter space quickly by using a set of simplified models to rapidly determine feasible tokamak designs. Spherical tokamaks have a number of differences compared with their conventional aspect ratio counterparts, and we present the spherical tokamak specific models implemented in the systems code PROCESS. There is an alternative relation between the plasma current and the ratio of the toroidal magnetic field to the safety factor, to account for an increased ratio of I_p/aB that can be accommodated at low aspect ratio. We also include the contribution of the diamagnetic current to the overall plasma current, which is higher than in a conventional aspect ratio device due to the higher beta. Various options are available to alter the build of the device; these include the ability to remove the central solenoid and avoid inboard breeding blankets, to join the TF coils to a single centerpost, to reposition the shaping poloidal field coils within the TF coil, and to increase the divertor space. We apply low and high temperature superconducting magnets to spherical tokamaks within PROCESS and explore the balance-of-plant to illustrate the requirements for net electricity production. Using these models, we present a benchmarking of PROCESS against published spherical tokamak designs, and then highlight interesting areas of parameter space that a future spherical tokamak for energy production could be positioned to operate in.

Keywords: *System Studies, Spherical Tokamaks, Electricity Production, Fusion Power Plants*

**Corresponding author:* *Stuart.Muldrew@ukaea.uk*

Basic Concept and Strategy of Japan's Fusion Demonstration Plant: JA DEMO

02-1.8

Ryoji Hiwatari *, N. Asakura, H. Utoh, S. Tokunaga, N. Nakajima, Y. Homma, Y. Miyoshi, Y. Someya, Y. Sakamoto, K. Tobita and Joint Special Design Team for Fusion DEMO

National Institutes for Quantum and Radiological Science and Technology, Aomori, Japan

Japan's demonstration plant (JA DEMO) missions are defined as follows: (1) steady and stable power generation beyond several hundred megawatts, (2) availability prospect for commercialization, and (3) overall tritium breeding to fulfill self-sufficiency of fuels. To realize those missions, the basic concept of JA DEMO has been developed based on the feasible technology as applied in the ITER design. First, the major radius R=8.5m of JA DEMO enables to accommodate the CS coil large enough for both pulse and steady-state discharge, to bridge the gap between ITER and DEMO. The fusion power P_f=1.5GW of JA DEMO is determined by the heat-handling capability of the divertor investigated by 2D divertor transport code. Because of neutron irradiation by longer operation than ITER, not only Wmono-block/Cu-alloy-pipe (for the strike-point region) but also W-mono-block/F82H-pipe (for the baffle and dome region) are applied for divertor target. This divertor enables longer operation period than about 1 year, and it contributes to increase the plant availability. The vertical maintenance method is applied for blanket replacement, and the replacement maneuver is originated using firm grip method by both up and down supports of the end-effector. The blanket module of honeycomb structure has pressure tightness against the pressurized water condition to avoid the in-vessel loss of coolant accident. The advanced functional materials (Li_2TO_3 and $Be_{12}Ti$) developed in the BA activity are applied to avoid hydrogen generation. This honeycomb blanket module achieves tritium breeding ratio TBR=1.07, and its critical R&D issue is manufacture method by hot isostatic pressing (HIP). As for the toroidal field coil (TFC), the design stress is improved from 667MPa of the ITER condition to 800MPa based on the existing cryogenic steel for high pressure gaseous hydrogen, while TFC follows the radial plate winding method. An operation plan of JA DEMO is also prepared to show the strategy to realize the JA DEMO missions. The operation plan is recently applied to discuss commissioning method, and it reveals requirement of the initial tritium

loading. Furthermore, to improve plant availability, the operation plan proposes the high core radiation discharge and the second divertor concept without Cu-alloy-pipe. Those basic concept and operation plan of JA DEMO are the starting point to discuss the shift to the DEMO construction phase in Japan.

Keywords: *plasma design, divertor, blanket, maintenance, operation plan*

**Corresponding author:* *hiwatari.ryoji@qst.go.jp*

Neutronics Conceptual Research on a Hybrid Blanket of China Fusion Engineering Test Reactor

02-2.1

Xueming Shi *, Guiming Qing, Xianjue Peng

Institute of Applied Physics and Computational Mathematics, Beijing, China

The engineering concept research of China Fusion Engineering Test Reactor (CFETR) is underway. In order to achieve tritium self sufficiency and lower tritium startup inventory, a fusion fission hybrid blanket scheme is proposed in this paper as a backup for the traditional pure fusion blanket concepts. In the hybrid blanket design, natural uranium is used as fuel, light water as coolant and Li_4SiO_4 as tritium breeder. MCORGS, a burnup code coupled by MCNP and ORIGENS, is used in the research. One dimensional design and optimization is firstly made to obtain maximum Tritium Breeding Ratio (TBR) and a moderate energy Multiplication (M). A 3D neutronics model of CFETR based on detailed CAD design is then used in the blanket conceptual research. At the beginning of the core, TBR_{3D} is 1.26 which is bigger than the max achievable TBR for pure fusion blanket (around 1.15), M is 3.18 which means the tritium startup inventory will be 3 times lower than the pure fusion blanket in case of a fixed blanket thermal power. After 12 years burnup with 200MW fusion power, TBR and M will be 1.28 and 4.05 respectively. The hybrid system is in a ultra deep subcritical state, the effective multiplication constant varies from 0.161 to 0.227.

Keywords: *CFETR, Tritium self-sufficiency, Fusion fission hybrid, Blanket*

**Corresponding author:* *sxm_shi@qq.com*

High-Heat Flux Tests of Fusion Materials with Stationary Plasma in the PLM Device

02-2.2

Viacheslav P. Budaev [a,b]*, S.D. Fedorovich [a], Y.V. Martynenko [b], A.V. Karpov [b], D.N. Gerasimov [a], A.Yu. Marchenkov [a], A.P. Sliva [a], E.V. Sviridov [a], M.V. Lukashevsky [a], A.V. Lazukin [a], I.E. Lyublinsky [c], A.V. Vertkov [c], K.S. Rogozin [a], Z.A. Zakletsky [a], D.S. Gvozdevskaya [a]

[a] *National Research University "MPEI", Moscow , Russia*
[b] *National Research Center "Kurchatov Institute", Moscow, Russia*
[c] *JSC "Red Star", Moscow, Russia*

Tests of in-vessel components in modern tokamaks have shown the erosion effects under the powerful plasma load. High heat flux tests of fusion materials are required for the ITER, DEMO and a fusion neutron source. Full-scale tests of the divertor and the first wall components like tungsten, lithium, tin components are extremely important to model relevant loads on plasma-facing materials of a fusion reactor. For such purposes, the PLM plasma device was constructed. The PLM plasma device is a linear system of a 8-pole multicusp magnetic field with parameters similar to the SOL plasma in a tokamak: stationary plasma density is up to $5\times10^{18}m^{-3}$, the electron temperature is up to 10 eV with a fraction of hot electrons of ~50 eV, the ion plasma flux onto the test sample is up to $5\times10^{21}m^{-2}s^{-1}$, plasma heat load on test target samples is of 1—5 MW/m^2. The ITER grade tungsten and divertor tungsten mock-up were irradiated with stationary plasma in the PLM. The combined tests of ITERgrade tungsten samples with e-beam load of 10—50 MW/m^2 and stationary plasma load of 1—2 MW/m^2 led to erosion, cracking, and nanostructured "fuzz" structure growth on the material surface. Capillary porous system of liquid tin was tested with stationary plasma in the PLM during ~200 minutes demonstrating sustainability to the high heat plasma load. Lithium materials deposited in the T-10 tokamak during experiments with lithium capillary-porous system were irradiated with stationary plasma in the PLM to test the evolution of the deposits under long-term plasma load. The work was supported by the RSF grant No. 17-19-01469, the ASNI of the PLM was supported by the RF Megagrant No. 14.Z50.31.0042.

Keywords: *high heat flux test, tungsten, tin, lithium, fusion materials, plasma-facing materials, ITER, fusion reactor*

**Corresponding author:* *Budaev@mail.ru*

Status of Preconceptual Design and Technology R&D for the EU DEMO Divertor

O2-2.3

Jeong-Ha You [a*], G. Mazzone [b], E. Visca [b], T. Barrett [c], P. Frosi [b], M. Fursdon [c], B-E. Ghidersa [d], H. Greuner [a], Ch. Harrington [c], P. Di Maio [e], D. Marzullo [f], A. v. Müller [a], M. Richou [g], R. Villari [b], C. Vorpahl [h] and WPDIV team

[a] *Max Planck Institute for Plasma Physics, Garching, Germany*
[b] *ENEA, Department of Fusion, Frascati, Italy*
[c] *CCFE, Culham Science Centre, Abingdon, UK*
[d] *KIT, Eggenstein-Leopoldshafen, Germany*
[e] *Department of Energy, Università degli Studi di Palermo, Palermo, Italy*
[f] *CREATE, Università degli Studi di Napoli Federico II, Napoli, Italy*
[g] *CEA, IRFM, Saint Paul Lez Durance, France*
[h] *EUROfusion, PMU PPPT, Garching, Germany*

As a prototypical demonstration reactor, a DEMO is supposed to provide verification of stable power exhaust capability to withstand high heat flux (HHF) from the edge plasma in all operational situations including transient plasma instabilities that may cause extreme thermal excursions. The divertor is the key component deployed for this crucial requirement. It will be exposed to harsh and complex loading environment characterized by severe neutron irradiation and electromagnetic impact.

In the framework of the EUROfusion Consortium, the work package "Divertor" has been performing pre-conceptual design and technology R&D dedicated for the divertor of the EU DEMO. The ultimate objective of these large-scale integrated multidisciplinary R&D efforts is to deliver at least one holistic design concept eligible for the subsequent conceptual design phase together with verified technology options associated with the design. All major loading features of DEMO divertor as mentioned above were quantitatively assessed by multi-physics analyses. The detailed CAD design of a divertor cassette and subsystems was optimized in an iterative way taking into account the results of the multi-physics analyses, particularly in terms of neutronic shielding and thermo-hydraulic performance of cooling.

Advanced design concepts and technologies were developed for the targets. In HHF qualification test campaigns, small-scale mock-ups of four water-cooled target concepts successfully passed fatigue tests at 20MW/m² (coolant: 130°C) up to at least 500 load cycles, overload tests at 25MW/m² (coolant: 20°C) up to 100 load cycles and screening tests (5 cycles) up to 32MW/m² (coolant: 20°C).

In this paper, the final results of the WPDIV activities in the first project phase since 2014 are presented.

Keywords: *DEMO, divertor, high-heat-flux, plasma-facing components, pre-conceptual design*

Corresponding author: *you@ipp.mpg.de*

European DEMO First Wall Shaping and Limiters Design and Analysis Status

O2-2.4

Zsolt Vizvary [a*], W. Arter [a], C. Bachmann [c], T.R. Barrett [a], B. Chuilon [a], P. Cooper [a], M. Firdaouss [b], T. Franke [c,d], J. Gerardin [b], R. Gowland [a], M. Kovari [a], F. Maviglia [c], M.L. Richiusa [a], E.V. Rosa Adame [a], C. Vorpahl [c], A. Wilde [a], Y. Xue [a]

[a] *UKAEA, Culham Science Centre, Abingdon, UK*
[b] *IRFM CEA Cadarache, Saint Paul lez Durance, France*
[c] *EUROfusion – Programme Management Unit, Garching, Germany*
[d] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*

The anticipated heat flux limit of the European DEMO first wall is ~1MW/m^2. During transient and off normal events, the heat load deposited on the wall would be much larger than that the steady state heat load and exceed the first wall limit, therefore the breeding blanket first wall needs to be protected in such events. This involves dedicated discrete limiters in certain regions of the machine that would take the brunt of the heat load as well as adequate shaping of the first wall. The current concept envisages limiters at a few (3-4) equatorial ports to cope with the ramp-up of the plasma; upper limiters (in ~8 upper ports) are considered for upward vertical displacement events. Two design options have been considered for these limiters: a modular design where the limiter plasma facing components are attached to individual plates that are assembled together so that transient electromagnetic loads can be reduced, and in case of damage the plates can be replaced/repaired individually; and a divertor-like design where the plasma facing components are attached to a single FE analysis results show that the integrity of the cooling pipes can be maintained during the Eurofer cassette. Other limiters considered include inner wall limiters in case of plasma contraction and lower limiters may be needed for downward vertical displacement events. The thermal hydraulic anticipated transient events. The limiters are considered to be sacrificial and designed to be replaceable independently from the breeding blanket system. The design has to allow that installation, removal or replacement of the limiters can be performed remotely. Strategy to tackle outstanding issues and required R&D is also discussed.

Keywords: *DEMO, First Wall shaping, Limiter, PFC, thermo-hydraulic analysis*

Corresponding author: *Zsolt.Vizvary@ukaea.uk*

The EUROfusion Materials Property Handbook for DEMO In-Vessel Components – Status and the Challenge to Improve Confidence Level for Engineering Data

O2-3.1

Michael Gorley [a]*, Eberhard Diegele [b,c], Ermile Gaganidze [c], Ferenc Gillemot [d], Gerald Pintsuk [e]

[a] United Kingdom Atomic Energy Authority, Culham Centre for Fusion Energy, Culham Science Centre, Abingdon, UK
[b] EUROfusion PMU, Garching, Germany
[c] Karlsruhe Institute of Technology, Institute for Applied Materials, Eggenstein-Leopoldshafen, Germany
[d] MTA Centre for Energy Research, Budapest, Hungary
[e] Forschungszentrum Jülich GmbH, Institute for Energy and Climate Research, Partner of the Trilateral Euregio Cluster (TEC), 52425, Jülich, Germany

The development of a specific materials database and handbook, for engineering design of in-vessel components of EU-DEMO, is an essential requirement for assessing the structural integrity by design. For key in-vessel materials, including EURFOER97, CuCrZr, Tungsten and dielectric and optical materials, this development has been ongoing now within the Engineering Data and Design Integration sub-project of the EUROfusion Materials Work Package for several years. However, the status still is that presently none of the mentioned materials exist in established nuclear codes and currently have in-sufficient data to ensure reliable engineering design and safety.

In this paper an overview of the current status of EU-DEMO database and handbook for these key invessel materials is provided. This comprises highlighting the practical steps taken to obtain the raw data, screening procedures and data storage, to ensure quality and provenance. We also discuss how this procedure has been utilized to produce materials handbook chapter on EUROFER97 as well as the critical challenges in data accumulation for CuCrZr and Tungsten, planned mitigations and the implications this has on structure design. Finally, key principles and philosophy behind our strategy to develop the materials database and handbook for the in-vessel materials are outlined, including concepts to accommodate sparse irradiated materials data, links to design criteria and importance to EU-DEMO engineering design.

Keywords: *Structural Integrity, Database and Handbook, Engineering Design, In-vessel Components*

Corresponding author: *mike.gorley@ukaea.uk*

Design of Fracture Resistant Tungsten Composites for High Heat Flux Applications

O2-3.2

Richard J. Kurtz *, Ba Nghiep Nguyen, Charles H. Henager, Jr.

Pacific Northwest National Laboratory, Richland, Washington, USA

Tungsten (W) is the leading solid material for fusion plasma facing component (PFC) applications because of its many desirable properties. Future fusion power system PFCs must tolerate an extremely hostile environment that includes high heat loads, neutron damage, and surface modifications driven by energetic particle impingement. However, W and most W-alloys generally exhibit low fracture toughness and a high ductile-to-brittle transition temperature that would render them brittle during operation. Therefore, W-alloys toughened by engineered reinforcement architectures, such as ductilephase toughening (DPT), are being studied. In DPT, a ductile phase is included in a brittle matrix to increase the overall work of fracture for the composite material.

Here we report experimental studies on W-composites that were examined from both a composite development perspective and from a model material standpoint. Bend specimens with various microstructures were tested at room and elevated temperatures in purified argon at over a wide range of displacement rates. As expected, the general principles of DPT were observed in the deformation and fracture behavior of these materials.

A finite element (FE) microstructural dual-phase model where the constituent phases were finely discretized based on digitized images and described by a continuum damage model was developed. In this approach homogenized meshed regions adjacent to a dual-phase meshed region of an actual bend specimen were created. The model is capable of modelling deformation, cracking, and crack bridging for DPT W-composites. We simulated the stress-strain responses and fracture morphologies of W-NiFe specimens subjected to three-point bending and compared predictions to corresponding experimental results to validate the model. The model was then used as a tool for tailoring composite architectures by examining a series of lamellar-like lattice microstructures to investigate the effects of microstructure on composite stress-strain response, damage, and fracture patterns. Model predictions show that a regular "brick" lattice microstructure exhibits significantly higher strength compared to random lamellar-like microstructures. The developed approach appears to be robust and an efficient methodology for tailoring the mechanical properties of DPT W composites.

Keywords: *tungsten, plasma facing components, ductile-phase toughening, finite element modeling*

**Corresponding author:* *rj.kurtz@pnnl.gov*

Validation of the Linear IFMIF Prototype Accelerator (LIPAc) in Rokkasho

O2-3.3

Keitaro Kondo [a*], Tomoya Akagi [a], Nicolas Bazin [b], Beatriz Brañas [c], Philippe Cara [d], Yann Carin [e], Nicolas Chauvin [b], Hervé Dzitko [e], Takashi Ebisawa [a], Enrico Fagotti [f], Dominique Gex [e], Roland Heidinger [e], Yosuke Hirata [a], Antti Jokinen [e], Atsushi Kasugai [a], Sunao Maebara [a], Alvaro Marchena [c], Alvaro Marqueta [e], Jacques Marroncle [b], Purificación Méndez [c], Ivan Moya [e], Guy Phillips [e], Andrea Pisent [f], Ivan Podadera [c], Giuseppe Pruneri [f], Keishi Sakamoto [a], Francesco Scantamburlo [f], Yoshito Shimosaki [a], Takahiro Shinya [a], Masayoshi Sugimoto [a], Moises Weber [c]

[a] *QST, Rokkasho-mura, Aomori, Japan*
[b] *CEA Saclay, Gif-sur-Yvette, France*
[c] *CIEMAT, Madrid, Spain*
[d] *IFMIF/EVEDA Project Team, Rokkasho-mura, Aomori, Japan*
[e] *Fusion for Energy, BA/IFMIF, Garching, Germany*
[f] *INFN-LNL, Legnaro, Italy*

For the development of the IFMIF (International Fusion Materials Irradiation Facility) aiming at material tests for fusion power plants, the construction of the Linear IFMIF Prototype Accelerator (LIPAc) has been conducted at Rokkasho, Japan under the Broader Approach Agreement. The aim of this accelerator is to demonstrate the validity of the low energy section of an IFMIF deuteron accelerator up to 9 MeV with a beam current of 125 mA in continuous wave (CW). This project has two major challenges that it is devoted to the construction of the world's most powerful deuteron accelerator and is a collaboration between Japan and Europe through in-kind contribution. The commissioning is carried out in three stages. In the 1st stage, the ion source and the LEBT (Low Energy Beam Transport line) was commissioned and showed the capability to produce the 100 keV deuteron beam of 140 mA, CW. At present, the 2nd commissioning stage of the RFQ (Radio Frequency Quadrupole) and the MEBT (Medium Energy Beam Transport line) is underway in order to demonstrate the capability of deuteron acceleration up to 5 MeV at low duty cycle (0.1 %). In the final stage, a Superconducting RF (SRF) linac will be installed to accelerate the beam up to 9 MeV. The assembly of the superconducting RF cavities and superconducting solenoid magnets of the SRF linac has been started in a clean room in Rokkasho, and the installation of accelerator components used in the final stage, i.e. the HEBT (High Energy Beam Transport line) and the beam dump accepting 1 MW, CW beam, has been completed. In the present paper, the progress of the validation of LIPAc carried out in Rokkasho as well as the preparation for the final goal of the commissioning will be described.

Keywords: *LIPAc, IFMIF, neutron source, accelerator, RFQ*

Corresponding author: *kondo.keitaro@qst.go.jp*

CLAM-ODS Steel as Structural Material beyond ITER

O2-3.4

Qunying Huang *, Xiaodong Mao, Liangliang Song, Shaojun Liu, Yican Wu and FDS Team

Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Anhui, China

The Science and Technology Ministry of China has launched a national key program in 2018 to develop innovative structural materials for CFETR (China Fusion Engineering Test Reactor) and fusion reactors. Oxide dispersion strengthened (ODS) steel and innovative manufacturing technologies are the main focus of this program. The program is leading by Institute of Nuclear Energy Safety Technology, and over 10 institutions including institutes, universities and industry are involved. An overview on the aim, scope and recent progress on the program is presented.

The development of ODS steel is based on CLAM (China Low Activation Martensitic) steel, which has been developed to satisfy the material requirements of ITER-TBM. In 2016, CLAM steel reached a mass production of 6-ton scale ingots, indicating the fabrication processes were mature. Its properties, including all around mechanical properties, corrosion in liquid PbLi alloy and neutron irradiation up to 21 dpa dose etc., and joining technologies for 1/3 scaled TBM mockup fabrication basically met the requirements of ITER-TBM.

To further improve the application temperature and irradiation resistance of the structural material for applications beyond ITER, the ODS version of CLAM is designed and being developed. Si-containing oxidation resistant CLAM-ODS steel containing high number density (>10^{23} m^{-3}) of nano-sized oxide particles (<10 nm) is developed, which exhibits an extra high strength of up to 800 MPa at temperature of 823K. The program covers research topics such as oxide/matrix interface design based on diffusion-multiple technique, mass production by electromagnetic assisted casting, complex component fabrication based on wire and arc additive manufacturing, as well as high dose heavy ion irradiation tests and related analyses. One of the major concerns of the CLAM-ODS program is the available manufacturing technology, which requires that excellent overall properties, the mass production technology and component fabrication are fulfilled at the same time. A new casting technology was developed, which consists of electromagnetic-ultrasonic driven smelting and strip casting. The

production scale is expected to reach ton-scale, and properties comparable to ODS steel fabricated by mechanical alloying and consolidation. It is expected that neutron irradiation tests of the newly developed CLAM-ODS steel be started no later than 2021.

Keywords: *CLAM, ODS, ITER, fabrication, structural material*

**Corresponding author:* *qunying.huang@inest.cas.cn*

The CFETR Tritium Plant: Requirements and Design Progress

O2-3.5

Guangming Ran [a]*, Jinguang Cai [b], Heyi Wang [a], Zhi Zhang [b], Chengjian Xiao [a], Xiaolin Wang [c]

[a] *Institute of Nuclear Physics and Chemistry, China Academy of Engineering Physics, Mianyang, Sichuan, China*
[b] *Institute of Materials, China Academy of Engineering Physics, Jiangyou, Sichuan, China*
[c] *China Academy of Engineering Physics, Mianyang, Sichuan, China*

Tritium plant technology is crucial for the successful development of China Fusion Engineering Test Reactor (CFETR) which aims to bridge the gap between a fusion experimental reactor (namely ITER) and a demonstration reactor. CFETR Engineering Design Activities (EDA) started at the end of 2017 and will last until the end of 2020. During this phase, a detailed conceptual design of tritium plant for CFETR is foreseen to be complete. At the end of 2014, a pre-conceptual design of CFETR tritium plant was already complete based on an early CFETR concept (200 MW). However, the main parameters of CFETR have changed a lot from the early version to the current one (200-1500 MW). The most significant change should be that the current CFETR design will cover both the engineering test phase and the DEMO validation phase. Accordingly, the pre-conceptual design of CFETR tritium plant is considered to be outdated. In this report, the interfaces and requirements of CFETR tritium plant will be redefined and the most recent design of CFETR tritium plant systems will be presented.

Keywords: *CFETR, tritium plant, interfaces and requirements, design progress*

**Corresponding author:* *guangmingran@caep.cn*

Optimization of the EU-DEMO Fuel Cycle Using Dynamic Modelling

O2-3.6

Yannick Hörstensmeyer *, Christian Day

Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany

The fuel cycle represents a key infrastructure element of the foreseen EU-DEMO fusion reactor. The large quantity of unburned fuel downstream the torus is to be processed and separated from any impurities in several consecutive steps before reinjection. The whole system is constrained by numerous boundary conditions - such as the addition of breeding blankets – which were not imposed on preceding machines. This calls for a simulation tool to investigate the setup and optimization of every fuel cycle subsystem as well as the functionality of the system as a whole to advance to the conceptual design phase starting in the following year. Consequently a fuel cycle simulator has been developed using the commercial software ASPEN Custom Modeler.

Key features of the program are dynamic modelling and custom sub-models to account for the intrinsic dwell periods of the reactor and the unique requirements of the various technologies used. In this context the underlying multi-loop architecture is explained as well as simulation approaches and model features of selected subsystems within the tritium plant.

In order to analyse the impact of key parameters of the fuel cycle, a sensitivity study was carried out. The results of the study with regard to the improvement potential of individual systems and the arrangement and control of the overall system are presented. A major finding is the performance dependency on the total throughput of DEMO and the impact of any hydrogen isotope imbalances.

Keywords: *EU-DEMO, Fuel Cycle, Modelling*

**Corresponding author:* *yannick.hoerstensmeyer@kit.edu*

Progress of China's TBM Tritium Technology

O2-3.7

Luo Deli *, Yao Yong, Huang Guoqiang, Huang Zhiyong, Song Jiangfeng, Chen Chang-an

China Academy of Engineering Physics, Mianyang, Sichuan, China

It is a main engineering goal of the ITER to test design concepts of tritium breeding blankets relevant to larger reactors like DEMO. China has decided to develop and test the helium-cooled ceramic breeder (HCCB) test blanket module (TBM) concept during ITER operation. Tritium related technology is one of the most important research fields in China ITER TBM program. Thus, the ancillary systems, the Tritium

Extraction System (TES), the Tritium Measurement System (TMS) and the helium Coolant Purification System (CPS), will be built to get experimental dada of the HCCB TBM during ITER D–T operation for evaluation of the tritium self-sufficiency as well as energy extraction technology for future fusion reactor design.

This paper presents the brief introduction of the R&D on the ancillary systems for operation of the TBM because most of the TBM experiment data will be got by operation of these systems. This paper introduces the progress of the R&D on these systems. A hydrogen isotope separation subsystem is integrated into the TES for accurate measurement of tritium as well as less tritium contamination hydrogen release into the environment compare with the original design of the TES. A tested assembly of a TES and CPS based on the results has been designed based on the results and the experiment results on the components show that the present design is feasible.

Keywords: *TBM, CPS, TES, tritium*

****Corresponding author:*** *luodeli2005@hotmail.com*

Progress of the Conceptual Design of the European DEMO Breeding Blanket, Tritium Extraction and Coolant Purification Systems

03-1.1

Fabio Cismondi [a]*, L.V. Boccaccini [b], P. Chiovaro [c], S. Ciattaglia [a], I. Cristescu [b], C. Day [b], A. Del Nevo [d], P.A. Di Maio [c], G. Federici [a], F. Hernandez [b], C. Moreno[e], I. Moscato [c], D. Rapisarda [e], A. Santucci [d], G.A. Spagnuolo [b], M. Utili [d]

[a] *EUROfusion Consortium, Programme Management Unit, Garching, Germany*

[b] *Karlsruhe Institute of Technology, Karlsruhe (KIT), Germany*

[c] *Dipartimento di Ingegneria, Università di Palermo, Palermo, Italy*

[d] *Fusion and Technology for Nuclear Safety and Security Department, ENEA, Italy*

[e] *CIEMAT, Fusion Technology Division, Madrid, Spain*

In the frame of the EUROfusion consortium activities the Helium Cooled Pebble Bed (HCPB) and the Water Cooled Lithium Lead (WCLL) concepts are being developed as possible candidates to become driver Breeding Blanket (BB) for the EU DEMO, which aims at the tritium self-sufficiency and net electricity production. The two BB design options encompass water or helium as coolants and solid ceramic with beryllium/beryllides or PbLi as tritium breeder and neutron multipliers. The BB segments have evolved towards a more stable conceptual design taking into account multiple feasibility aspects and requirements imposed by interfacing systems. The reference and back-up technologies for the Tritium Extraction and Removal (TER) from the helium purge gas and the PbLi are developed addressing key feasibility aspects and implications on the tokamak layout. The impact of water coolant activation is assessed by studying the spatial distribution of ^{16}N and ^{17}N isotopes dose rates, in particular in proximity of isolation valves. As the BB internals offer and ideal environment (high temperatures, thin structural material) to promote the tritium permeation, studies are devoted to the assessment of the permeation rate and inventory in the coolant. Those are the key parameters for the feasibility assessment and technology selection for the Coolant Purification Systems (CPS) as well as for assessment of the tritium permeation at the Steam Generator and the associated Safety analyses of the DEMO plant.

Keywords: *Breeding Blanket, Coolant Purification, Tritium Extraction and Removal*

****Corresponding author:*** *fabio.cismondi@euro-fusion.org*

Tritium Extraction from PbLi: Technologies and Progresses

03-1.2

David Rapisarda [a]*, Ion Cristescu [b], Belit Garcinuño [a], Marco Utili [c]

[a] *CIEMAT, Madrid, Spain*

[b] *KIT, Eggenstein-Leopoldshafen, Germany*

[c] *ENEA Brasimone, FSN-ING, Camugnano, Italy*

Tritium self-sufficiency has become one of the main issues in the consecution of fusion towards a feasible energy source. For this reason, important efforts are being conducted in Europe with the aim of developing reliable tritium extraction technologies. The lead-lithium eutectic alloy is employed as breeder, in particular, in one of the so-called ‘driver’ blankets, the Water Cooled Lithium Lead (WCLL), but also in more advanced concepts, such as the Dual Coolant Lithium Lead (DCLL) breeding blanket. Different facilities are being constructed or upgraded in Europe with the aim of demonstrating the validity of the extraction techniques which are at present being considered, e.g. Gas Liquid Contactors (GLC), Permeation Against Vacuum (PAV) or Vacuum Sieve Tray (VST). The operational conditions of these facilities have been agreed in order to cover full specifications of the breeding blankets providing a test-matrix that improves the analysis and comparison of results. In this sense, experiments with lead-lithium at high and low velocities, at different temperatures and in the presence/absence of tritium or other hydrogen isotopes are envisioned.

The present work provides a general overview of these techniques focusing on their most relevant progresses and known issues. The development and current status of the different facilities for conducting experiments at different laboratory scales (CLIPPER at CIEMAT, TRIEX-II at ENEA or VST at KIT) is also presented.

Keywords: *tritium, lead-lithium, extraction technologies, DEMO*

***Corresponding author:** *david.rapisarda@ciemat.es*

Conceptual Design for Higher Capability of the Tritium Production by the Honeycomb Structure Blanket of JA DEMO

03-1.3

Youji Someya *, Kenji Tobita, Ryoji Hiwatari, Yoshiteru Sakamoto and Joint Special Design Team for Fusion D EMO

National Institutes for Quantum and radiological Science and Technology, Rokkasho Aomori, Japan

The conceptual design of the breeding blanket with a honeycomb structure has been created with pressure tightness against in-box Loss-of-coolant accident based on a water-cooled solid breeder. In the previous design, the breeding area of the module was divided into 0.1-m-squared cells with rib structure. As a honeycomb structure is higher in pressure tightness than a square prism structure, the area for filling the mixed pebbles breeder of Li_2TiO_3 pebbles and $Be_{12}Ti$ ones can be enlarged. Then, the overall TBR is improved to increase the packing ratio of the tritium breeding material.

In the created blanket, the capabilities of the pressure tightness, tritium breeding and heat removal are studied using interaction analyses of the neutronics, stresses and fluid dynamics analysis. As a result, a rib with the thickness of 0.015 m is needed to withstand the design pressure of 17.2 MPa by a stress analysis. The packing factor of the mixed pebbles breeder increase to 77 % from 68 % by changing the rib structure from a square prism structure to a honeycomb structure. From the 3D neutronics analysis results, the target of the overall TBR (>1.05) is achievable. The cooling system for the created blanket is designed by fluid dynamics analysis based on the PWR water conditions which are the coolant temperature of 290 - 325 °C and the operation pressure of 15.5 MPa, respectively. In addition, the tritium extraction system in the created blanket is proposed together with the purge gas system which does not clog the holes. The saturated time of the tritium extraction is also estimated to grasp the tritium inventory in the breeding area.

Keywords: *Honeycomb structure, Mixed pebbles breeder, in-box LOCA, tritium extraction system, JA DEMO*

***Corresponding author:** *someya.yoji@qst.go.jp*

Nuclear Design of Divertor Tokamak Test (DTT) Facility

03-1.4

Rosaria Villari [a]*, Maurizio Angelone [a], Barbara Caiffi [a], Andrea Colangeli [a], Flavio Crisanti [a], Davide Flammini [a], Nicola Fonnesu [a], Raul Luis [b], Giovanni Mariano [c], Daniele Marocco [a], Fabio Moro [a], Gian Mario Polli [a], Sandro Sandri [d]

[a] *ENEA, FSN Department, Frascati, Rome, Italy*
[b] *Instituto de Plasmas e Fusão Nuclear, Instituto Superior Técnico, Universidade de Lisboa, Lisboa, Portugal*
[c] *DIAEE, Sapienza University of Rome, Rome, Italy*
[d] *ENEA, Institute of Radiation Protection, Frascati, Rome, Italy*

The Divertor Tokamak Test facility (DTT) is an Italian project aimed at testing the physics and technology of alternative power and particle exhaust solutions for the DEMO reactor. DTT will be a medium size, high field, superconductive tokamak device. It will operate in deuterium with expected 2.5 MeV neutron yield rate more than 10^{17} n/s in high performance phase and a total neutron production of 3-4 $x10^{22}$ n during its lifetime. Therefore, the machine components will be exposed to an intense neutron and gamma irradiation and high neutron-induced activation, with a significant impact on the design of the tokamak components, auxiliary heating and diagnostics systems, as well as on licensing, radiation protection, maintenance, decommissioning and waste management.

This paper presents the three-dimensional neutronics, activation and shutdown dose rate analyses performed with MCNP5 Monte Carlo code, FISPACT-II inventory code and Advanced D1S dynamic tool for the design and licensing of DTT. Advanced compact shielding solutions and mitigation strategies have been studied to guarantee sufficient protection of the superconducting coils and to reduce the streaming and the neutron-induced radioactivity. The present nuclear design study provides main outcomes for the loads assessment, shielding and materials requirements and on maintenance strategy and storage of activated components.

Keywords: *Divertor Tokamak Test facility, Neutronics, MCNP, Activation, Shielding*

***Corresponding author:** *rosaria.villari@enea.it*

Key Design Integration Issues Addressed in the EU DEMO Pre-Concept Design Phase

O3-1.5

Christian Bachmann *, S. Ciattaglia, F. Cismondi, C. Gliss, G. Federici, T. Härtl, G. Keech, R. Kembleton, F. Maviglia, M. Siccinio

PPPT Department, EUROfusion Consortium, Garching, Germany

The EU fusion roadmap defines the development of a DEMO, which achieves a high plasma operation time and demonstrates Tritium self-sufficiency and net electricity output. The pre-conceptual design phase will be concluded with a gate review in 2020. Eight key design integration issues have been identified as critical, either because the corresponding solution found in ITER is not suitable in DEMO or because the issues are DEMO-specific and not present in ITER. All of these will affect in their resolution the design and possibly the technology of several tokamak and plant systems or even the DEMO architecture: (i) Feasibility of wall protection limiters during plasma transients, (ii) integrated design of breeding blanket and ancillary systems, (iii) power exhaust taking advantage of advanced divertor configurations, (iv) tokamak architecture based on vertical blanket segments, (v) direct or indirect power conversion concept, (vi) configuration of plant systems in the tokamak building, (vii) feasibility of hydrogen separation in the torus vacuum pump and direct recirculation, and (viii) plasma scenario.

For each of these issues potential solutions have been identified. In some cases alternative solutions are recognized and are addressed for risk mitigation. The key design integration issues will be introduced and discussed in this article including a summary of the identified solution concepts and results achieved so far.

***Corresponding author:** *christian.bachmann@euro-fusion.org*

Progress of Water Cooled Ceramic Breeder Test Blanket Module System

O3-2.1

Yoshinori Kawamura [a*], Hyoseong Gwon [a], Wenhai Guan [a], Hisashi Tanigawa [a], Takanori Hirose [a], Atsushi Wakasa [a], Tamon Ouchi [b], Seiji Yoshino [b], Kentaro Hattori [b], Misato Nakata [b], Noriaki Chiba [c], Takuya Kushida [c], Seiji Mori [d], Hiromasa Iida [d], Takumi Yamamoto [d], Toshihiko Yamanishi [e], Hiroyasu Uto [a], Yoji Someya [a], Kentaro Ochiai [a], Hideo Sakasegawa [a], Jaehwan Kim [a], Hirofumi Nakamura [a], Hiroyasu Tanigawa [a], Shigeru Ohira [a], Takumi Hayashi [a]

[a] *National Institutes for Quantum and Radiological Science and Technology, Rokkasho, Japan*

[b] *Yamato System Engineer Co., Ltd., Hitachi, Japan*

[c] *Nihon Axis Co., Ltd., Hitachinaka, Japan*

[d] *Nippon Advanced Technology Co., Ltd., Tokai, Japan*

[e] *National Institutes for Quantum and Radiological Science and Technology, Tokai, Japan*

A Water-Cooled Ceramic Breeder (WCCB) Test Blanket Module (TBM) System is being developed as one of the most important steps toward DEMO blanket in Japan. As for the ITER-TBM program, the conceptual design of WCCB-TBM system has been approved by the ITER organization (IO). And two years have already passed after the start of the preliminary design activity. WCCB TBM Team had a concern about TBM box structure withstanding over 15 MPa of the coolant pressure, and decided to change the design as the result of study on structural integrity. Recently, the design change of the TBM structure has been approved domestically. So, WCCB TBM Team has applied the design change to the IO, and agreed with the IO to receive the conceptual design review which is limited the scope to the TBM structure.

This paper provides an overview of the recent achievements of the development of the WCCB Test Blanket Module System in Japan.

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

Keywords: *WCCB-TBM, structural integrity, F82H, breeder, tritium*

***Corresponding author:** *kawamura.yoshinori@qst.go.jp*

Fusion Technologies Development at ENEA Brasimone Research Centre: Status and Perspectives

03-2.2

D. Martelli, A. Del Nevo, I. Di Piazza, M. Utili,
Mariana Tarantino *

Experimental Engineering Division, Fusion and Technology for Nuclear Safety and Security Department, ENEA Brasimone Research Centre, Camugnano, Italy

The Experimental Engineering Division (FSN-ING) by the ENEA Brasimone Research Centre (R.C.) includes decades of scientific expertise in experimental and numerical research activities in supporting Fusion Technology development. In the European and international framework, ENEA coordinates the Italian fusion program, supported by linked third parties as universities, research institutes and industries. In this context Brasimone R.C. is involved in activities related to Breeding Blanket (BB) and Divertor technologies development. The experimental engineering division is involved in the design of an integrated concept of the Water-Cooled Lithium Lead (WCLL) BB. In parallel, activities dealing with Helium Cooled Lithium Lead (HCLL), Dual Coolant Lithium Lead (DCLL) and Helium Cooled Pebble Bed (HCPB) have been performed so far. This paper describes the scientific works presently ongoing at Brasimone R.C. enveloped in PbLi, Lithium, Helium and pressurized water technologies, thermomechanical characterization of structural materials, analysis of materials corrosion rate and development and qualification of anti-permeation/corrosion barrier. This paper describes the experimental activities conducted for the investigation of safety-relevant scenarios as In-Box LOCA (water-PbLi interaction in the WCLL or shock waves propagation generated by He injection in PbLi in the HCLL/DCLL-BBs), for the development of coolant purification systems of HCLL/HCPB-BB, qualification of instrumentations and investigation of erosion-corrosion resistance of materials exposed to lithium flowing conditions focusing on the development of protective coating and coating technics as anti-permeation and anticorrosion barriers. In addition, experimental facilities are dedicated to the thermal-hydraulic characterization of full-scale prototype of the ITER Divertor cassette, to the investigation of tritium extraction systems and to define procedure for the refurbishment operation for ITER divertor cassettes. Finally, new activities have been planned to support the Divertor Tokamak Test (DTT) Divertor characterization, the large-scale water-PbLi interaction and a PbLi components validation in relevant scale for WCLL-BB.

Keywords: *Breeding Blankets, Research Infrastructures, Divertor, Brasimone*

***Corresponding author**: mariano.tarantino@enea.it*

Required and Achievable TBR for the European DEMO

03-2.3

Ulrich Fischer [a]*, C. Bachmann [b], F. Cismondi [b], M. Coleman [c], C. Day [a], Y. Hörstensmeyer [a], F. Moro [d], P. Pereslavtsev [a]

[a] *Karlsruhe Institute of Technology, Karlsruhe, Germany*
[b] *EUROfusion - Programme Management Unit, Garching, Germany*
[c] *Culham Centre for Fusion Energy, Culham Science Centre, Abingdon, UK*
[d] *ENEA, Department of Fusion and Nuclear Safety Technology, Frascati, Rome, Italy*

The Power Plant Physics and Technology (PPPT) programme of EUROfusion aims at the development of a DEMOnstration fusion power plant (DEMO) as central element of the European roadmap to fusion energy. DEMO is anticipated to deliver a substantial amount of electricity to the grid and operate with a closed tritium fuel cycle. Tritium self-sufficiency is a strict pre-condition for the operation of such a DEMO which is designed to produce a fusion power in the order of 2 GW. Thus a net Tritium Breeding Ratio (TBR) ≥ 1.0 is required, i. e. it must be assured that per D-T fusion reaction one triton, generated in the breeding blankets surrounding the plasma chamber, is finally available for injection into the plasma. This needs to be proven by appropriate neutronic calculations which provide estimates for the global TBR and are validated, as far as possible, against experiments. In effect, a global TBR with some additional margin in excess of unity must be demonstrated to account for uncertainties in the neutronic simulations for DEMO and the loss budget expected for the processing of the tritium in the fuel cycle. Another margin has to be added when considering DEMO to provide the start-up inventory for a follow-up power plant.

This work presents an up-to-date assessment of the TBR requirement for DEMO as deduced from neutronics considerations including uncertainties and limitations in the underlying design calculations, and resulting from advanced tritium fuel cycle concepts with a significantly reduced tritium loss budget and the demand to build-up a tritium start-up inventory. The TBR requirement is evaluated against the actual TBR performance which is achievable for DEMO based on latest results for the HCPB (Helium Cooled Pebble Bed) and WCLL (Water Cooled Lithium Lead) driver blankets.

***Corresponding author:** ulrich.fischer@kit.edu*

Recent Advances in Tritium Modeling and Impacts on Tritium Management for Outer Fuel Cycle

O3-2.4

Alice Ying [a]*, Marco Riva [a], Mu-Young Ahn [b], Carlos Moreno [c], Ion Cristescu [d]

[a] *University of California, Los Angeles (UCLA), USA*
[b] *NFRI, Daejeon, Korea*
[c] *CIEMAT, Madrid, Spain*
[d] *Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany*

Recent tritium transport, permeation, and extraction modeling in both detailed component as well as system levels development for ceramic tritium breeding blankets has shined some light into tritium management issues associated with this line of blanket, particularly with He as the coolant. A detailed component model, taking into account of multi-physics, design and operational features, allows for a more accurate estimation of the tritium source term present in the primary cooling system that impacts the tritium permeation into the building/environment- a safety and licensing concern for a fusion nuclear reactor. This is particularly crucial in tritium streams with the existence of other hydrogen isotopes due to an operational practice where about 0.1% of H_2 is added into the helium purge gas stream to promote tritium release. This significantly reduces the amount of tritium permeation from the breeding zone to the He coolant stream; the accumulation of the tritium inside the He coolant stream could produce reverse effects. This fairly high H_2 partial pressure in the helium purge gas further complicates tritium extraction methodology, compromises extraction efficiency, and results in a higher initial start-up tritium inventory.

In light of the significance of the results from these recent modeling efforts, this paper provides a concentrated analysis to help clarify situations involved in tritium management in the He-cooled ceramic breeding blankets such as the effects of co-permeation. The analysis will also include a review of the past irradiation experiments that concluded the practice of the addition of 0.1% H_2, and a review of tritium extraction techniques proposed with the goals to provide an R&D guidance to this line of breeding blankets. This paper also represents a research progress conducted under a working group embedded in the IEA Nuclear Technology of Fusion Reactors (NTFR) solid breeder blanket subtask.

Keywords: *Tritium transport, outer-fuel-cycle tritium flow, tritium processing, modeling*

**Corresponding author:* *ying@fusion.ucla.edu*

The Issue of Tritium in DEMO Coolant and Mitigation Strategies

O3-2.5

Alessia Santucci [a]*, Marco Incelli [a], Leonardo Noschese [b], Carlos Moreno [c], Marco Utili [d], Silvano Tosti [a], Christian Day [e]

[a] *ENEA-Frascati, Frascati, Italy*
[b] *University of Rome "Tor Vergata", Rome, Italy*
[c] *CIEMAT, Madrid, Spain*
[d] *ENEA-Brasimone, Camugnano (BO), Italy*
[e] *Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany*

Key requirements for the DEMO fusion reactor are the demonstration of the production of net electricity and the tritium self-sufficiency. To accomplish the first point, an optimum heat transfer has to be guaranteed in both the breeder blanket (BB) and the steam generators (SGs) regions, this means that large surface areas with thin walls are operated at high temperature. At the same time, the fulfillment of the self-sufficiency implies a tritium generation rate of about 320 g/day coupled with efficient tritium extraction systems and very limited tritium losses. Under these conditions, the problem of tritium permeation from blanket into coolant and from coolant into environment becomes of great relevance.

The intent of this work is to provide a comprehensive overview of the tritium migration issue with particular focus on the status of the mitigation strategies in both helium and water coolants. Such mitigation strategies are essentially represented by: i) the size and efficiency of the coolant purification system (CPS) and ii) the feasibility and efficiency of anti-permeation barriers. Therefore the pre-conceptual design of the CPS is presented and its impact on the tritium inventory in coolant and on the tritium permeation from coolant into SGs is discussed.

Keywords: *tritium permeation, coolant purification system, anti-permeation barriers*

**Corresponding author:* *alessia.santucci@enea.it*

Introduction of Fusion Reactor as Transmutation System for Minor Actinides and Fission Product

O3-3.1

Hidetoshi Hashizume *, Noritaka Yusa, Hiroki Shishido

Department of Quantum Science and Energy Engineering, Graduate School of Engineering, Miyagi, Japan

Following the ITER project to aim at stable fusion reaction, activities to design demo reactors are now being carried out enthusiastically throughout the world. An attractive future fusion reactor design from the viewpoint of power station is, however, not shown due to the limitation

of fusion power caused by integrity of divertor system under extremely high heat flux. That is, cost of electricity (COE) is much higher than that of other existing system to result in not strong support for further fusion R&D.

In this study, therefore, we propose a new usage of fusion reactors to close fission fuel cycle by transmuting minor actinides (MA) and fission product generated after reprocessing nuclear fission fuels.

The numerical analysis based on a simple model with a small change in the fusion reactor design indicate that support factors (transmuted mass in fusion reactor/generated mass from 1GWe of PWR) for both the minor actinides and fission product in case of 0.5 GW fusion power reactor are around 10, which means that in Japan, we needs about 4 times larger fusion power than that of the ITER to avoid geological disposal for the high level radioactive waste.

By adding this attractive functionality to the fusion reactor, not only fusion reactor becomes extremely attractive to enhance the R&D for fusion reactor but also fusion reactor design becomes free from the COE competition in the near future to allow us to develop the fusion power reactors to produce electric power.

In the full paper, the support factors will be discussed based on precise model.

Keywords: *transmutation, divertor*

**Corresponding author: hidetoshi.hashizume@qse.tohoku.ac.jp*

Recent Advances in Nuclear Instrumentation and THEIR Application to Fusion Blankets

O3-3.2

Pattrick Calderoni *

Idaho National Laboratory, Idaho Falls, USA

The US Department of Energy (DOE) engages in research and development activities for the development of nuclear instrumentation as part of its Nuclear Energy (NE) program portfolio. In recent years the activities have been consolidated under the framework of the In-Pile Instrumentation (I2) program, part of the Crosscutting Technology Development (CTD) element of Nuclear Energy Enabling Technologies (NEET). The program mission is to enable an enhanced real-time understanding of nuclear fuels and materials performance during irradiation by providing accurate inpile measurements of test conditions, material properties, and the characterization of structural and chemical states. Sensors and measurement techniques under consideration are divided in three categories based on their development status and intended purpose: baseline capabilities (high TRL); innovative sensors (low TRL) and integrated measurement systems (single effect test). Baseline capabilities comprise self-powered detectors to measure neutron or gamma flux; thermocouples and passive monitors to measure local temperature and LVDTs to measure deformation and pressure. Innovative sensors are better described in terms of technology rather than specific sensor configuration, since in many cases design optimization is not yet completed. They include electrical sensors, ultrasonic sensors, fiber optic sensors and sensors fabricated by advanced manufacturing techniques. Integrated measurement systems are designed for the characterization of a specific parameter of interest, such as thermal conductivity (photothermal radiometry, three-omega sensor, and needle probe) or mechanical properties (creep testing, diameter gauges; crack growth and void formation). The paper will discuss the development status of each technology, the results or near-term plans of qualification testing in neutron irradiation facilities and their relevance to fusion blanket technology.

Keywords: *nuclear energy, fusion technology, nuclear instrumentation, fiber optics, acoustic sensor*

**Corresponding author: pattrick.calderoni@inl.gov*

Advanced Structural Steels for the Fusion Blanket: Alloy Development, Irradiation Effects and Characterization at ORNL

O3-3.3

Arunodaya Bhattacharya [a]*, **Xiang Chen** [a], Lizhen Tan [a], Ying Yang [a], Yukinori Yamamoto [a], Sebastien Dryepondt [a], Caleb Massey [a], David Hoelzer [a], Niyanth Sridharan [a], Kevin Field [c], Tim Graening [d], Chad M. Parish [a], Samara M. Levine [c], Andrew R. Lupini [a], Steven J. Zinkle [a,b], Yutai Katoh [a]

[a] *Oak Ridge National Laboratory, Oak Ridge, TN, USA*
[b] *University of Tennessee, Knoxville, TN, USA*
[c] *University of Michigan, Ann Arbor, MI, USA*
[b] *Karlsruhe Institute of Technology, Karlsruhe, Baden-Württemberg, Germany*

Reduced activation ferritic-martensitic (RAFM) steels are currently the most promising structural material candidates for the DEMO fusion blanket. However, the safe operating temperature window for structural components fabricated using RAFM steels will be very narrow, between ~350-550 °C. This undesirable narrow temperature window arises from two well-known issues that RAFM steels suffer from: irradiation-induced embrittlement causing a loss of fracture toughness which gives the lower temperature limit, while the upper temperature limit is a consequence of poor creep strength at elevated temperatures, which is expected to be exacerbated by the irradiation. ORNL is leading an effort towards designing new radiation tolerant alloys using conventional/advanced manufacturing routes, neutron irradiation experiments using the High

Flux Isotope Reactor (HFIR), small specimen testing techniques and advanced characterization. This talk will present the latest developments in the next generation RAFM steels such as Castable Nanostructured Alloys (CNAs), additively manufactured FM steels and Oxide Dispersion Strengthened (ODS) steels that show promise of better mechanical properties and microstructure stability under irradiation when compared with the current generation of fusion relevant steels like Eurofer97 or F82H. The applicability of advanced electron microscopy based characterization that can significantly improve the understanding of RAFM/ODS steel microstructures in unirradiated and irradiated form, thereby, aiding alloy design activities will also be presented.

Keywords: *RAFM steels, ODS steels, neutron irradiations, alloy designing, advanced characterization*

Corresponding author: *bhattacharya@ornl.gov*

The DEMO Plant Electrical System: Issues and Perspective

03-3.4

Elena Gaio [a]*, A. Ferro [a], A. Maistrello [a], M. Dan, F. Lunardon [a], L. Barucca [b], S. Ciattaglia [b], G. Federici [b]

[a] *Consorzio RFX (CNR, ENEA, INFN, Università di Padova, Acciaierie Venete S.p.A.), Padova, Italy*

[b] *EUROfusion Consortium, PPPT Department, Garching, Germany*

DEMO represents a fundamental step between ITER and a commercial reactor. Its main aim is to demonstrate the capability of producing electricity and operate with a closed fuel-cycle. In parallel to the studies to explore the technology feasibility of key parts, there is also the need of others addressed at improving the plant RAMI and contributing to the overall assessment of the technical and economic viability of commercial fusion.

With specific reference to the electrical power plant, the design solutions mainly adopted in the last decades need to be reconsidered now at least for two main reasons; the first is that the requirements for DEMO are more demanding, because the power peaks envisaged are significantly higher than in ITER and the handling of the generated and recirculated power poses issues not experienced in previous experimental devices, neither they will be in ITER.

The other reason is that the continuous progress in the field of power semiconductors and converter topologies together with the decreasing of their cost could make more attractive, in perspective, different design solutions with respect to traditional ones.

In DEMO, the Plant Electrical System (PES) is one of the largest systems; its scope covers the electrical power generation and the Power Supply (PS) systems necessary for supplying all the plant loads. The studies conducted so far have been addressed to well understand, on the one hand the specific design challenges with respect to an equivalent system for a fission power plant and on the other hand the additional issues with respect to the ITER electrical system.

In order to satisfy the power needs for the DEMO operation, minimizing the request of contribution from the electrical grid, and in particular high peaks of active power and huge amount of reactive power, R&D has been launched to explore suitable electrical/magnetic energy storage systems and advanced power converter topologies to maximize the energy exchange within the plant.

This talk will give an overview of the DEMO PES main requirements and system architecture, of the progress of the studies conducted so far, highlighting the main issues to be faced, the paths taken to address them and the preliminary results achieved.

Corresponding author: *elena.gaio@igi.cnr.it*

Ongoing Activities and Future Directions for the U. S. Fusion Safety Program

03-3.5

Brad Merrill *, Lee Cadwallader, Masashi Shimada, Paul Humrickhouse, Chase Taylor, Robert Pawelko

Idaho National Laboratory, Idaho Falls, USA

The Idaho National Laboratory (INL) Fusion Safety Program (FSP) is the Office of Fusion Energy Sciences' (OFES) lead laboratory for Magnetic Fusion Energy (MFE) Safety. Our mission is to assist the US and international fusion communities in developing the inherent safety and environmental potential of fusion power. This mission requires fusion safety data we obtain by conducting experiments in the Safety and Tritium Applied Research (STAR) laboratory, a Department of Energy (DOE) OFES facility, possessing the unique capabilities needed to experimentally investigate plasma or gas pressure driven tritium retention and permeation in activated materials and assess the chemical reactivity of toxic materials. Presently, STAR upgrades are underway that will improve building safety, expand the facility's tritium experimental space, and build new tritium science experimental test stands. The new tritium science test stands include a dedicated linear plasma experiment for super-permeable membrane development and a tritium extraction test stand built to examine the physics of extracting tritium from fusion coolants (liquid metals and helium). In addition to experimental data, the FSP develops fusion safety accident analysis computer codes, performs risk assessments, collects component failure rate data, and develops waste management strategies. Recent progress in developing the MELCOR-TMAP code has resulted in a self-consistent safety analysis tool capable of simultaneously analyzing the various coolants contained within a fusion reactor. Future plans for this code include porting fusion modifications into a recent version of MELCOR. This article describes these activities and future directions of our program.

Note: This work was prepared for the U. S. Department of Energy, Office of Fusion Energy Sciences, under the DOE Idaho Field Office contract number DE-AC07-05ID14517.

Keywords: *Fusion Safety, Tritium, Material Characterization, Computer Code Development*

***Corresponding author:** *Brad.Merrill@inl.gov*

Integrative Neutronics Simulation of SuperMC for Reactors Design

04-1.1

Jing Song *, Liqin Hu, Pengcheng Long, Lijuan Hao, Bin Wu, Shengpeng Yu, Guangyao Sun, Lei Wang, Quan Gan, Peng He, Jun Zou, Guomin Sun, Yican Wu and FDS Team

Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Hefei, Anhui, China

Reactors are extremely complicated due to the multi-physics coupling phenomenon including various neutronics processes, thermo-hydraulics and etc. Integrative neutronics simulation with high fidelity will significantly speed up the design and analysis processes and enable accurate predictions of the performance features. The integrative neutronics simulation of SuperMC will be introduced.

The integrative neutronics simulation including transport, burnup, activation and dose simulations with taking into account the thermo-hydraulics feedback, can be performed using the internally coupled approach. Bucket-sorting energy grid searching, automatic criticality search and data decomposition parallel calculation were developed for burnup calculation. The advanced D1S method based on response of collision with nuclide was developed, which can consider all decay photons emitted during the complete transmutation process. Parametric-surface-represented entity clipping method was developed to achieve the data mapping between constructive solid geometry for neutronics calculation and unstructured mesh for CFD thermo-hydraulics calculation.

The shielding performance analysis of bio-shield plugs in B1 of ITER with updated models and the activation data handbook for ITER were performed with the neutronics simulation functions of SuperMC. Furthermore, space reactors were designed especially to mainly optimize the weight of shadow shielding and simulate the multi-physics fields for prediction.

Keywords: *SuperMC, integrative neutronics simulation, reactor design*

***Corresponding author:** *jing.song@fds.org.cn*

Neutronics Experiments for Advanced Nuclear Systems at HINEG Facility

04-1.2

Yican Wu *, Yongfeng Wang, Zhigang Wang, Qi Yang, Wen Wang, Canjun Liang, Size Chen, Jun Gao, Taosheng Li, Lijuan Hao, Jing Song, Pengcheng Long, Chao Liu, Fang Wang, Liqin Hu, Qunying Huang and FDS Team

Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Hefei, Anhui, China

For the research and development (R&D) of nuclear technology and safety for advanced nuclear systems, a high intensity D-T fusion neutron generator is keenly needed, which is an important testing platform for basic research on neutronics, including the neutronics method and software validation, neutronics performance test of components or materials, core physics study of advanced reactors, nuclear data measurement and validation, etc.

High Intensity D-T Fusion Neutron Generator (HINEG) developed by FDS team has a D-T fusion neutron yield of 6.4×10^{12} n/s. It has also been coupled with the Lead-based Zero Power Critical/Subcritical Reactor CLEAR-0 in order to perform the physical and engineering validation of advanced nuclear energy systems. Recently, a series of experiments have been carried out on HINEG facility, such as neutronics performance test of fusion reactor blanket, irradiation tests of concrete for fusion reactor system, validation of the neutronics software SuperMC, neutron biological effects of C. elegans, neutron radiography nondestructive testing, and so on. Taking the experiment of neutronics performance test as an example, in order to validate the neutronics design of the dual function lithium-lead (DFLL) test blanket module (TBM), an experiment was carried out with scaled DFLLTBM mock-up. The comparison of experimental data and corresponding calculated results by SuperMC with HENDL3.0 library was presented, and a good agreement within 10% was reached. In addition, the upgrade of HINEG for higher neutron yield is on-going. The current of the D^+ beam on target has been increased from 35 mA to more than 55 mA, which will contribute to the D-T neutron yield of 1.0×10^{13} n/s. This presentation will introduce the latest experiment campaigns as well as the upgrade campaigns of HINEG.

Keywords: *fusion neutron generator, neutronics experiment, nuclear technology*

***Corresponding author:** *yican.wu@fds.org.cn*

Compatibility of Advanced Tritium Breeders and Neutron Multipliers

O4-1.3

Jae-Hwan Kim *, Masaru Nakamichi

Fusion Energy Research & Development Directorate, Department of Blanket Systems Research, National Institutes for Quantum and Radiological Science and Technology, Aomori, Japan

Advanced tritium breeders and neutron multipliers for a demonstration (DEMO) fusion reactor have been developed to improve stability at high temperatures.

With respect to the advanced tritium breeder, Li-enriched Li_2TiO_3 and Li_2TiO_3 solid soluted with Li_2ZrO_3 have been developed by a emulsion process. As for the advanced neutron multipliers, beryllium intermetallic compounds (beryllides) $Be_{12}V$ and $Be_{13}Zr$ with Si dopant have been developed by a cominational process with a plasma sintering and a rotating electrode process. Further characterizations are still undergoing to verify the superiority of these advanced functional materials.

In parallel with researches and developments of materialsl, in Japan, new concept of blanket systems for the DEMO reactor design has been paid attention with great intrest to improve a tritium breeding ratio by pebble packing with binary size of pebbles under mixture packing of breeders and multipliers. Since the maximum operation temperature is anticipated at 1173 K, both materials must be stable at this temperature and in specific, compatibility of each material must be clarified to meet the stability requirment for this new concept.

In this study, compatibility of advanced tritium breeders and neutron multipliers is investigated, focusing on analytical approaches of a x-ray dffraction (XRD) and electron probe micro-analyzer (EPMA) with mircostructure obesrevations of a scanning electron microscopy (SEM) and a transmission electron micrscopy (TEM) depending on the holding times up to 1000 h at 1173 K.

Keywords: *Advanced neutron multiplier, Advanced tritium breeder, DEMO, Compatibility*

***Corresponding author:** kim.jaehwan@qst.go.jp*

R&D Strategy Including DT Neutron Irradiation Tests of In-vessel Components by the Year-Long Operation in the Helical Fusion Reactor FFHR-b1

O4-1.4

Junichi Miyazawa [a,b]*, Takuya Goto [a,b], Teruya Tanaka [a,b], Hitoshi Tamura [a], Nagato Yanagi [a,b], Akio Sagara [a,b] and the FFHR Design Group

[a] *National Institute for Fusion Science, Gifu, Japan*
[b] *The Graduate University for Advanced Studies, Gifu, Japan*

Toward realization of the helical fusion reactor FFHR-d1 of ~3 GW fusion output, a step-by-step approach including the FFHR-b1 as the HEVNS (HElical Volumetric Neutron Source) has been proposed. In these FFHR series, new technologies of High Temperature Superconducting (HTS) coils, the fusible metal pebble divertor, and the cartridge-type molten salt RAFM blanket are adopted. Important R&D issues to be solved have been identified including these new technologies. The FFHR-b1 is assigned as the device to investigate and solve nuclear-related issues of, for example, the neutron irradiation up to 10^{23} n/m^2 on the HTS coils in the real reactor-relevant situation, blanket component tests on tritium breeding and electricity generation, detritiation and maintenance of divertor and blanket units in a high-radiation environment, and decommissioning. The device size of the FFHR-b1 is as small as the LHD, while the magnetic field is doubled. The construction cost of the FFHR-b1 is roughly estimated to be ~200B JPY. Neutrons are generated by the beam-plasma fusion reaction. At this moment, it is supposed that neutral D beams of ~20 MW tangentially injected to the helical T plasma will produce ~7 MW of fusion output, more or less, which corresponds to ~10^{16} n/(m^2 s) (~0.7 dpa/FPY) of neutron flux on the blanket first wall of ~200 m^2. Before the DT phase, several years of the DD phase is planned including a one-year steady-state operation with NBI supported by ECRH demonstrating the availability of > 70 %. The tritium production and electricity generation by the blanket system already start at this DD phase with ~10^{14} n/(m^2 s) of neutron flux to prepare the initial loading tritium for the DT phase. Decommissioning will be also performed as one of the important issues after ~15 years of operation.

Keywords: *helical reactor, volumetric neutron source, HTS magnets, divertor, blanket, maintenance*

***Corresponding author:** miyazawa@LHD.nifs.ac.jp*

AI Nuclear Design and Safety Simulation System Coupling with SuperMC

O4-1.5

Pengcheng Long *, Liqin Hu, Jing Song, Tao He, Fang Wang, Lijuan Hao, Bin Wu, Shengeng Yu, Guangyao Sun, Quan Gan, Qi Yang, Lei Wang, Jin Wang, Leiming Shang, Yican Wu and FDS Team

Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Anhui, China

Advanced numerical simulation can restore the complex physical processes as much as possible, and predict system behavior and safety performance, and it facilitates accurate design and assessment. But the optimization of system design is still a challenging work. It is great significant to analyze and optimize the system design and safety level of fusion systems through artificial intelligence based simulations as the artificial intelligence (abbreviated AI) is growing up fast.

To serve the nuclear design and safety evaluation of nuclear energy systems, the Virtual Nuclear Power Plant in Digital Society Environment (Virtual4DS) has been developed by the Institute of Nuclear Safety Technology, Chinese Academic of Sciences · FDS Team. It is a comprehensive simulation platform featured with artificial intelligence and cloud computing. It coupled with the multi-functional neutronics calculation program SuperMC, Neutronics-Thermohydaulics coupled simulation program NTC, tritium analysis program TAS, fusion database management system FusionDB, etc. Users could access the platform through the network and perform analysis flexibly. It shares various data, and supports the integrated neutronics and multi-physics analysis.

In Virtual4DS, various AI algorithms, including differential evolution, ant colony algorithm, and artificial neural networks, etc., has been used in the modeling and simulation to raises complex issues on neutronics and multi-physics coupling. The AI simulation capabilities of Virtual4DS has been verified by a series of international benchmarks, and has been used in design and analysis of key systems of fusion reactor including blankets, radiation shielding systems, etc. In this manuscript, the system architecture, verifications and featured AI simulation applications will be presented.

Keywords: *nuclear design, safety evaluation, Artificial Intelligence, SuperMC*

Corresponding author: *pengcheng.long@fds.org.cn*

Development of GDT Based Fusion Neutron Source CVNS

O4-1.6

Yican Wu [a*], Jie Yu [a], Zhibin Chen [a], Dehong Chen [a], Qiusun Zeng [a], Alexander Ivanov [b], Peter Bagryansky [b], Igor Kotelnikov [b], Vadim Prikhodko [b], Dmitry Yakovlev [b] and FDS Team [a]

[a] *Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Anhui, China*

[b] *Budker Institute of Nuclear Physics, Siberian Branch, Russia Academy of Sciences, Russia*

Gas dynamic trap (GDT) is a kind of mirror fusion system which can be a compact volumetric fusion neutron source based on current magnetic confinement fusion technologies, such as superconducting magnet, plasma auxiliary heating, neutral beam injection, plasma diagnostics and controlling, etc. It is in principle easy to reach steady-state operation and high neutron flux with relative low fusion power (~3MW) and low tritium consumption (<200g per year), so that it can provide a real fusion nuclear environment with relatively large test volume for fusion material test and components test but tritium breeding will be not necessary.

Based on these, an international mega-science project on compact volumetric neutron source (CVNS) based on GDT was jointly proposed by Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences and Budker Institute of Nuclear Physics (BINP), Russia Academy of Sciences. FVNS will be used for fusion nuclear technology and fusion-fission hybrid reactor development, and also for other early fusion applications such as neutron radiography, medical isotope production, neutron therapy, etc. It is planned to be designed in 5 years, constructed in 5 years and operated in 20 years in both DD and DT operation modes.

In this paper, the latest R&D of FVNS will be introduced as well as the challenges. In order to promote the FVNS project, an international preparatory committee on FVNS has been established, and the wider international collaboration is also being built under the framework of the international preparatory committee.

Keywords: *gas dynamic trap, fusion volumetric neutron source, international mega-science project*

Corresponding author: *yican.wu@fds.org.cn*

Approach on Improving Reliability of DEMO Technical Solutions

04-1.7

Tonio Pinna *, Danilo Nicola Dongiovanni

ENEA, FNS-FUSTEC Department, Frascati, Italy

One of the key elements in the engineering development of the Demonstration Power Plant (DEMO) reactor is to demonstrate the capability in achieving satisfactory reliability and availability (RA) targets. Since these targets are strictly related also to maintenance and inspection activities, the integrated approach in RA verification and optimization needs to be based on all the following issues: Reliability, Availability, Maintainability and Inspectability (RAMI).

After an initial effort to establish guidelines to perform RAMI assessments and define RAMI requirements, recent years activities have mainly addressed the validation of proposed design solutions in terms of reliability and availability. RAMI assessments play an important role during all design cycle phases by focusing on different aspects depending on the development stage. In fact, system functions, related requirements and deriving constraints are the main objective of RAMI assessment during pre-conceptual and conceptual design phases, when getting insight on the rationale of the plant. The identification of potential failure mechanisms and of design updates proposals to remove such mechanisms or mitigate related consequences, become the main goal of analysis as design details increase during subsequent plant development phases. Moreover, being DEMO a nuclear facility, the RAMI approach for management of the technical risks shall integrate and be in the first instance driven by the safety aspects. Inadequate reliability or failed failure indications of components deemed safety critical items might directly jeopardize the public and worker safety.

Some of the RA analyses performed till now for different DEMO systems and breeding blanket configurations are here presented. Results in terms of expected annual frequency of failures of components and of specific failure events will be reported together with indications about criticality and suggestions for designers to improve the design solutions.

Keywords: *Reliability, Availability, FMEA, Failure Rate, DEMO, RAMI*

**Corresponding author:* *tonio.pinna@enea.it*

ARC Reactor: Radioactive Safety Assessment and Preliminary Environmental Impact Study

04-2.1

Massimo Zucchetti [a]*, Raffaella Testoni [a], Stefano Segantin [a], Zachary Hartwig [b], Dennis Whyte [b]

[a] *DENERG, Politecnico di Torino, Italy*

[b] *Plasma Science and Fusion Center, MIT, Cambridge (MA), USA*

Affordable Robust Compact reactor is a conceptual design for a Tokamak conceived by Massachusetts Institute of Technology (MIT) researchers. It represents a new generation of fusion reactors. The ARC design is under development and update.

Since ARC will be a D-T tokamak, neutron generation and material activation will be main issues for safety studies and assessment of environmental impact and siting questions. For ARC, the safety assessment goal is to demonstrate that, however an experimental nuclear reactor connected to the electric grid, it could be easily sited in several sites in the US, without particular problems and the need of any emergency plan implying population evacuation or sheltering. Another safety feature that will be verified is the need of a containment building in which the reactor should be put.

Starting from activation studies already developed for the ARC's vacuum vessel structure and the liquid blanket as well, a further and deeper analysis, that includes the first wall and neutron multiplier layer activation, has been carried out. Afterwards, taking advantage of the GENII/FRAMES population dose code, the study arrives to the assessment of doses to MEIs (Most Exposed Individuals) from accidental activated material release in atmosphere, including possible tritium releases: those source terms are estimated by a preliminary deterministic study on selected accidental sequences. Results of the work are presented in the paper and the quite excellent safety characteristic of ARC are confirmed.

Keywords: *ARC, Tokamak, Safety, Activation, GENII*

**Corresponding author:* *massimo.zucchetti@polito.it*

Safety Requirements and Design Strategy for Advanced Fusion Neutron Source A-FNS

04-2.2

Makoto M. Nakamura *, S. Sato, K. Ochiai, M. Oyaidzu, M. Ohta, S. Kwon, C. Park, A. Kasugai

Rokkasho Fusion Institute, National Institutes for Quantum and Radiological Science and Technology, Rokkasho, Japan

Advanced Fusion Neutron Source (A-FNS) is an accelerator-driven (IFMIF-type) fusion neutron source to offer fast neutrons to study irradiation effects on fusion reactor materials, which is to be sited at Rokkasho, Japan. A-FNS should be designed so that radiation safety is assured for workers and the public with compliance of the Japanese radiation safety regulation, that was revised in 2017 after the accidents at the Fukushima Daiichi nuclear power plant and the J-PARC hadron experimental facility. The revision to be accounted for the A-FNS conceptual design is that "safety provisions should be made *a priori* for emergency situations." The A-FNS safety design is *the first design case adopted to the new Japanese radiation licensing* of large-scale radiation facilities in Japan.

We present safety requirements, design strategy and associated safety analysis of the A-FNS conceptual design. We have newly developed safety requirements for the A-FNS conceptual design, which are in compliance not only with the revised Japanese regulation but also to the international nuclear safety standards. Numerical project guidelines are presented, which satisfy the safety requirements for the workers and public radiation safety. We have analyzed a "bounding accident case" of A-FNS at Rokkasho, in which all the amount of the major mobilizable radioactive materials, i.e. tritium and beryllium-7 in the target liquid lithium loop, are released to the environment. We performed a site-specific dose analysis using meteorological data observed at Rokkasho. The simulation result indicates that the early dose to the public (for a week) 500 m away from the release point is 2 orders of magnitude smaller than the IAEA guideline for emergent evacuation, 100 mSv. The result also indicates that the early dose of the site-specific case is 1 order of magnitude smaller than the dose evaluated previously under the generic conservative meteorological condition [1].

[1] M.M. Nakamura, K. Ochiai, Fusion Eng. Des. 118, 104 (2017).

Keywords: *fusion neutron source, A-FNS, safety requirements, radiation protection, licensing*

Corresponding author: *nakamura.makoto@qst.go.jp*

Initial Development of MELCOR 2.2 for Fusion

O4-2.3

Paul W. Humrickhouse *, Brad J. Merrill

Idaho National Laboratory, Idaho Falls, USA

MELCOR is a computer code for modeling the progression of transients and accidents in nuclear reactors. Originally developed by Sandia National Laboratories for application to light water-cooled fission reactors, a modified version has long been developed and maintained by Idaho National Laboratory specifically for application to fusion energy systems. This modified version provides a number of additional models and capabilities important to the modeling of various fusion energy systems. Among these are a fusion fluids library that includes the properties of liquid Lithium, PbLi, Sn, SnLi, and FLiBe; the ability to model systems including multiple such working fluids; fluid properties that extend to cryogenic temperatures and allow for the freezing of fluids including oxygen and nitrogen on surfaces; steam and air oxidation models for fusion relevant materials such as tungsten, beryllium and graphite; modified aerosol transport models including dust resuspension; and others. These models presently exist in MELCOR 1.8.6 for Fusion. Upstream development of MELCOR 1.8.6 ceased in ~2005, and since that time, MELCOR 2 has been developed by Sandia and includes improved numerical methods, numerous bug fixes, and a variety of new features. In order to take advantage of these, INL has now begun porting fusion modifications to the current branch of MELCOR, presently at version 2.2. This paper describes the status of MELCOR 2.2 for Fusion, outlines which models have now been implemented, and provides some comparisons with earlier versions of the code. Some priorities and plans for continuing development are also outlined.

Keywords: *MELCOR, Fusion Safety*

Corresponding author: *paul.humrickhouse@inl.gov*

Optimizing the EU-DEMO Pellet Fuelling Scheme

O4-2.4

Peter T. Lang [a]*, F. Cismondi [b], C. Day [c], E. Fable [a], C. Gliss [b], B. Pégourié [d], B. Ploeckl [a]

[a] *Max-Planck-Insitut für Plasmaphysik, Garching, Germany*
[b] *EUROfusion Consortium, PPPT, Garching, Germany*
[c] *Karlsruhe Institute of Technology (KIT), Karlsruhe,Germany*
[d] *CEA, IRFM, St. Paul-lez-Durance Cedex, France*

Efficient fuelling will be an essential task in the EU-DEMO. The basic requirement is to establish the target plasma core density with a minimum particle flux in order to maximize plasma performance while keeping the burden on the fuel cycle low. Injection of mm-size pellets formed from solid hydrogen fuel from the vessel inboard will be the main actuator for core density control. Hence, efficiency maximization has to be achieved by optimizing pellet parameters and injection geometry taking into account boundary conditions resulting from system integration needs. For this the layout of the guiding tubes required for transferring pellets to the destined launching point into the plasma turned out crucial. Thus, in an initial step design, criteria were elaborated by modelling particle deposition profiles for the different possible injection trajectories. Here, contrary to the smaller devices operated up to date, full high-field side injection is required and the pellet speed component perpendicular to the separatrix contours turned out as most relevant parameter. Design activities integrating the guiding tube into the vessel and the breeding blanket unveiled two possible variants. One with the guiding tube ending at the rear end of the breeding blanket and one with the guiding tube (not cooled) penetrating to a contour with an offset of 0.4 m into the blanket (about half of its thickness). While the first one is technically simpler, the second one requires some extra efforts but bears the benefit of a deeper particle deposition. Considering the benefits of the penetrating solution an actively cooled version of the guiding tube has been studied and can potentially increase the breeding blanket penetration to 0.6m. To evaluate the benefit of the extra efforts, a full closed loop modelling of pellet fuelling was performed to calculate the amount of pellet particle flux required to meet the requirements prescribed by the target plasma. Results provide now a sound basis for a best possible overall solution. Once this selection is made, the developed modelling tool will be employed to analyse the optimized pellet fuelling scheme further, e.g. by taking into account the interplay of the pellet fuelling actuator with burn control requirements.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission

Keywords: *EU-DEMO, Fuelling, Design integration, Pellet*

**Corresponding author:* *peter.lang@ipp.mpg.de*

Study and Characterization of Potential Adsorbent Materials for the Design of the Hydrogen Isotopes Extraction and Analysis System

O4-2.5

Amit Sircar *, V. Gayathri Devi, Deepak Yadav, Jyoti Shankar Mishra, Ranjana Gangradey, J. Gayathry, Rahul Tomar, Pragnesh B. Dhorajiya, Purvi Dave

Institute for Plasma Research, Bhat, Gandhinagar, Gujarat, India

Selection of appropriate adsorbents for the Hydrogen Isotopes Removal System (HIRS) is important for its effective functioning. The two main systems of HIRS, viz., Atmospheric Molecular Sieve Bed (AMSB) adsorber and Cryogenic Molecular Sieve Bed (CMSB) adsorber remove ppm levels of water vapour, hydrogen isotopes, oxygen and nitrogen gas from Helium gas. Adsorbents, viz., Activated Carbon, Zeolites MS 3A, 4A, 5A and 13X and Activated Alumina have been studied in detail at room and cryogenic temperatures to understand their surface characteristics, adsorption potential and adsorbate selectivity. This work discusses the evolution of the potential adsorbents for the AMSB and CMSB.

Keywords: *Adsorbent, cryogenic, hydrogen isotopes, extraction, characterization*

**Corresponding author:* *asircar@ipr.res.in*

R&D of Tritium Systems for CFETR: Progress and Prospect

O4-2.6

Luo Wenhua*, Meng Daiqiao, Chen Chang'an, Kou Huaqin, Luo Junhong, Xiong Renjin, Zhang Zhi, Zhang Guikai, Song Jiangfeng, Luo Deli

China Academy of Engineering Physics, Mianyang, China

China has decided to develop its own fusion engineering test reactor. Tritium plant is one of the key systems of CFETR. A program supposed by China ministry of Science and technology named Conceptual design and key technologies research on tritium plant for fusion reactor was started on September, 2011. After several years of research, we have finished the design of TEP, SDS, WDS, ISS and tritium safety system. The key technologies such as hydrogen storage materials for SDS, catalysts for WDS, palladium alloy membranes for TEP are under research. In this paper, the progress and prospect of tritium technology for R&D of CFETR is introduced.

Keywords: *tritium, CFETR, TEP, SDS, ISS*

**Corresponding author:* *iterchina@163.com*

Study on Hydrogen Adsorption and Desorption Using Large-Scale Cryogenic Molecular Sieve Bed

O4-2.7

Mu-Young Ahn [a]*, Soon Chang Park [a], Seok-Kwon Son [a], Alice Ying [b], Seungyon Cho [a], Yi-Hyun Park [a], Youngmin Lee [a]

[a] *National Fusion Research Institute, Daejeon, Korea*

[b] *Mechanical and Aerospace Engineering Department, University of California, Los Angeles, USA*

Cryogenic hydrogen adsorption and desorption is considered as a candidate process to separate low concentration hydrogen isotopes from the helium purge gas of breeding blankets and to route them to the inner fuel cycle in fusion reactors. Recently, hydrogen adsorption performance in a Cryogenic Molecular Sieve Bed (CMSB) was evaluated using PGLoop facility at NFRI to confirm its feasibility in large-scale applications such as Tritium Extraction System (TES) of the ITER Test Blanket System (TBS) and beyond. In this paper, desorption performance and parameters are studied for the large-scale applications. Desorption rates of hydrogen and other impurities are measured at the outlet of the CMSB using Quadrapole Mass Spectrometer (QMS). Desorption scenarios and their impacts on operations of tritium circuits in breeding blankets are

investigated with focuses on transition phenomena between adsorption and desorption phases.

Keywords: *Cryogenic Molecular Sieve Bed (CMSB), hydrogen adsorption, hydrogen desorption, molecular sieve 5A*

**Corresponding author: myahn74@nfri.re.kr*

Overview of the IFMIF-DONES Project

O4-3.1

A. Ibarra [a], F. Arbeiter [b], D. Bernardi [c], J. Castellanos [a], W. Krolas [d], M. Cappelli [f], U. Fischer [b], R. Heidinger [e], **Francisco Martin-Fuertes** [a*], G. Micciché [c], A. Muñoz [f], F. S. Nitti [c], T. Pinna [g], J. Quiñones [a] and the full IFMIF-DONES team

[a] *CIEMAT, Madrid, Spain*
[b] *KIT, Karlsruhe, Germany*
[c] *ENEA, Brasimone, Italy*
[d] *IFJ PAN, Krakow, Poland*
[e] *F4E, Garching, Germany*
[f] *Empresarios Agrupados, Madrid, Spain*
[g] *ENEA, Frascati, Italy*

The International Fusion Materials Irradiation Facility - Demo Oriented NEutron Source (IFMIF-DONES) is a proposed research infrastructure for irradiation of the same type of materials to use in a fusion reactor. The facility would provide a unique neutron source of energy spectrum and flux level representative of those expected for the first wall containing future fusion reactors. Materials irradiation data under such conditions are of fundamental interest for the fusion community to feed and validate modelling tools for materials radiation damage. IFMIF-DONES has been identified as one of the key facilities required for the development of the EU Fusion Roadmap, recently released, and in the critical path to DEMO. The design is presently being developed in the framework of a specific work package of the EUROfusion Consortium and it is strongly linked to the engineering work and validation results being obtained in the IFMIF/EVEDA (Engineering and Validation Engineering Design Activities) project under the framework of the EU-JP Bilateral Agreement to the Broader Approach to Fusion. It has been proposed to build the facility in Spain (in the Granada area).

The main specifications of the facility are as follows: IFMIF-DONES is based on a 40 MeV, 125 mA in continuous wave mode (CW) deuteron accelerator (5 MW beam average power) hitting with a rectangular beam size (approx. 20 cm x 5 cm) a liquid Li screen target flowing at 15 m/s to absorb the beam power and generating a 10^{18} m^{-2} s^{-1} neutron flux by means of stripping nuclear reactions. Materials placed very close to the target are irradiated to obtain damage rates up to 15 atomic displacements per year (dpa/year), and irradiation takes place under temperature controlled conditions. After a long irradiation period, modules containing the samples will be partially dismantled while samples themselves will be transported and characterized in separate laboratories.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement Nº 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission, Fusion for Energy, or of the authors' home institutions or research funder

Keywords: *neutron source, materials irradiation, nuclear physics, isotopes production*

**Corresponding author: francisco.martin-fuertes@ciemat.es*

On the Role of Integrated Computer Modelling in Fusion Technology

O4-3.2

Sergey Smolentsev [a], Gandolfo Alessandro Spagnuolo [b], Arkady Serikov [b], Jens Juul Rasmussen [c], Anders H. Nielsen [c], Volker Naulin [c], Jaime Marian [d], Matti Coleman [f,g], **Lorenzo Malerba** [h*]

[a] *Mechanical & Aerospace Eng. Department, UCLA, Los Angeles, USA*
[b] *KIT, Eggenstein-Leopoldshafen, Germany*
[c] *PPFE, Department Physics, Technical University of Denmark, Lyngby, Denmark*
[d] *Materials Science & Eng. Department, UCLA, Los Angeles, USA*
[f] *UKAEA, Culham Science Centre, Abingdon, UK*
[g] *University of Cambridge, Cambridge, UK*
[h] *CIEMAT, Energy Materials Division, Madrid, Spain*

Computer modelling plays an increasingly important role in all science and technology domains. Once suitable computer simulation tools are in place, different reactor and component concepts can be subjected to various operational and accidental conditions *in silico*, providing an obvious advantage to the design of nuclear energy devices. The potential is immense: from fewer and more targeted experimental validations for complex multi-physics models, right up to integrated reactor designs codes, modelling can significantly improve productivity in a broad range of fusion R&D activities. Simulation can both handle very complex systems and help to distinguish among the individual effects of variables that in experiments would not be easily separated or studied. Accordingly, computer modelling is expected to play an increasingly important role in fusion design and technology, where the complexity of the physical processes involved (plasma, materials, engineering), and the highly interconnected nature of systems and components ("system of systems" design), call for support from sophisticated and integrated computer simulation tools. In this presentation, we review the contribution of coupled computer modelling to the design of the breeding blanket in terms of neutronics, materials behaviour, plasma-materials interaction and radiation effects, as well as compatibility with fluids, mechanical performance and diagnostics, supplemented by simulations of plasma transport out of the confinement region to determine heat and particle loads on plasma facing components. The current possibilities and levels of maturity of existing simulation tools are critically analysed, having in mind the possibility of integrating several tools in the future and highlighting the difficulties of such an endeavour.

Keywords: *fusion reactor, computer modelling, neutronics, materials, plasma, model integration*

**Corresponding author:* *lorenzo.malerba@ciemat.es*

Conceptual Design of a Breeding Blanket for Laser Fusion Power Plants with Tunable Tritium Breeding Ratio (TBR) Capabilities

O4-3.3

Aldo A. Fierro [a]*, F. Sordo [b], I.A. Carbajal Ramos [c,d], J.M. Perlado [a,e], A. Rivera [a,e]

[a] *Instituto de Fusión Nuclear, Universidad Politécnica de Madrid, Madrid, Spain*
[b] *Consorcio ESS-Bilbao. Polígono Ugaldeguren III, Zamudio, Spain*
[c] *Department of Chemical Engineering, Kansas State University, USA*
[d] *Consejo Nacional de Investigaciones Científicas y Técnicas (CONICET)-Centro Atómico Bariloche, Argentina*
[e] *Departamento de Ingeniería Energética, ETSI Industriales, Universidad Politécnica de Madrid, Spain*

Nuclear fusion power plants cannot operate without an appropriate breeding blanket, mainly for neutron thermalization and tritium breeding through (n,t) reactions with lithium isotopes. Tritium shortage makes impossible plant operation and excess tritium production is a safety issue. It has been recognized that tritium demand will vary during the plant lifetime. Furthermore, calculations of tritium breeding ratio and overall inventories are nowadays not free of uncertainties. A solution is to design a blanket with capabilities to tune the tritium breeding ratio (TBR) during operation. Despite this is a serious point in nuclear fusion technology it is still unaddressed. In this talk, we describe for the first time a conceptual design of a breeding blanket based on lithium ceramics with beryllium multiplier and surrounded by a (heavy) water neutron reflector. Varying the reflector thickness (equivalent to vary the filling level of a tank), the tritium breeding ratio can be tuned at will to fulfill the requirements of a power plant. The resulting design turns out to be compact and does not require lithium enrichment.

Keywords: *fusion, tritium breeding, lithium, neutron irradiation*

**Corresponding author:* *aa.fierro@estudiantes.upm.es*

Service Joining Strategy for the EU DEMO

O4-3.4

Tristan Tremethick *, Simon Kirk, Keelan Keogh, Alexander O'Hare, Emily Harford, Ben Quirk

United Kingdom Atomic Energy Authority, Culham Science Centre, Abingdon, UK

As part of the European Research Roadmap to the Realisation of Fusion Energy, the DEMO reactor aims to show the feasibility of a fusion power plant. Due to the loss of revenue created by downtime and the potential for a breakdown to render a reactor inoperable, maintenance is "mission critical" for a power plant. The harsh environment of a fusion reactor dictates that maintenance must be carried out remotely, which requires the development of new strategies and technologies. There are many challenges to be solved, one of which is how to manage service connections. Within DEMO, the plasma-facing, first-wall components will be the most challenging to connect services to, due to the number of connections and operational environment. High speed, highly reliable cutting and welding tools are required to minimise downtime and mitigate the danger of rendering the reactor inoperable. Uniquely, these tools are required to operate wholly in-bore to allow the current pipe density in the DEMO architecture, something that is not available from industry.

Here, the development of a Service Joining System to meet the DEMO scenario is presented. The joining strategy is discussed along with the substantiation of the design solution. The results of proof-of-principle trials to date are discussed and their implications for the strategy considered. Having discussed feasibility for DEMO, a roadmap is presented for the development of the Service Joining System to an appropriate Technology Readiness Level.

Keywords: *DEMO, Welding, Cutting, Laser, Pipe, Services*

**Corresponding author:* *tristan.tremethick@ukaea.uk*

Recovery from a Hot Water Leakage at the Tokamak ADEX Upgrade

O4-3.5

Volker Rohde *, A. Herrmann, M. Balden, D. Bösser, K. Hunger, T. Vierle and ASDEX Upgrade Team

Max Planck Institut für Plasmaphysik, Garching, Germany

In most fusion devices water is used to cool down the plasma facing components (PFCs), which requires operation of high pressured water infrastructure inside a vacuum vessel. At mid-size tokamak ASDEX Upgrade (AUG), which is in operation since 1988, the same system is also used for vessel baking to 150 °C after a vent. In November 2017, during a regular vessel baking, a gasket of a water pipe failed, which leads to the ingress of hot water steam into the vacuum of the vessel. The hot steam developing reacted chemically with the in-vessel surfaces consisting out of tungsten and stainless steel, which were partly coated by boron hydride layers for wall conditioning, forming W-oxide, rust and boric acid.

Whereas most of the vessel was at 150 °C during the event, some more remote locations, as cable feed through and some diagnostics, i.e. instruments which measure properties of the plasma, were at a reduced temperature, leading to condensation of water at these locations. In total about 1000 flanges are installed at AUG, mostly for diagnostics access.

After removal of the water, and the chemical reaction products on the PFCs, venting with dried air was used to dehumidify all components. At some remote locations this procedure did not work, which causes chemical reactions on a longer time scale, as the formation of molybdenum oxide at electrical feedthrough pins, Al_2O_3 at Helicoflex gaskets and installations made out of Al. Additional some kinds of brazing used in electrical feed through were damaged. These reactions caused subsequently the failure of some components and especially vacuum leakages.

In these contributions the observed damages, the repair strategies and the adopted solutions are described. Finally a strategy to minimize this kind of damages in future events is presented.

Keywords: *Tokamak, vacuum, plasma facing components, diagnostics*

****Corresponding author:*** *Rohde@ipp.mpg.de*

Automated Maintenance of Nuclear Fusion Power Plants

O4-3.6

Guy Burroughes *, Samuel Jimenez

RACE, UKAEA, Oxfordshire, UK

As nuclear fusion reactors evolve towards the higher power levels needed for commercial electricity production they will become increasingly inhospitable to human entry, so that the scope of remote maintenance will increase significantly. Versatile teleoperation approaches are suitable to maintain experimental machines, which undergo constant modifications to meet the needs of their scientific campaigns, but future power plants will require much shorter maintenance shutdown durations if commercially viable plant availability is to be achieved. Hence, the maintenance systems must become faster and more efficient. Automation emerges as a potential solution, its benefits amply demonstrated in conventional industrial practice. However, the design of maintenance strategies and systems is an inherently multi-domain problem that spans the depth of the complexity hierarchy, interfacing with every aspect of the plant. Implementing automated maintenance therefore requires communication with many different stakeholders with widely disparate expertise, which can lead to different interpretations of automation concepts and mixed expectations of its benefits and limitations. Thus, a first step towards the development of automated maintenance is to set out clearly what automation is, and what it is not, in the context of a fusion reactor. Hence, this paper presents a conceptual framework which lays the foundation on which to build an automated maintenance strategy that is suitable for future fusion power plants. The aim is to provide a clear and common starting point from which to define goals, manage expectations, guide research, and assist the development of suitable standards and regulations.

Keywords: *remote handling, safety, maintenance, standards*

****Corresponding author:*** *guy.burroughes@ukaea.uk*

Pulsed Neutron Sources Based on Plasma Focus Chambers

O4-3.7

Yuriy V. Mikhailov *, B.D. Lemeshko, I.A. Prokuratov, A.K. Dulatov

Federal State Unitary Enterprise "All-Russian Research Institute of Automatics n.a. N.L. Dukhov" (VNIIA), Moscow, Russia

In this paper, the features of the generation of pulsed ionizing radiation by sources based on plasma focus (PF) chambers are considered. The pinch effect arising in the PF chambers leads to the generation of neutron and X-ray radiation of nanosecond duration. A number of perspective installations with PF chambers are considered in this work: generators of neutron radiation of an industrial design, that used as closed and safe sources of neutrons with energy of 14 MeV; a source of pulsed hard X-ray based on a PF chamber with an discharge current amplitude of up to 1.5 MA.

At the developed devices with PF chambers, the generation of ionizing radiation in pinch was considered. The consideration was carried out under the assumption of the beam-target mechanism of neutron generation - the results of measurements of the average energy of the ion beam, leading to the generation of neutrons, are given for various experimental conditions. The measurements were carried out according to the time-of-flight method using scintillation detectors. With the deuterium filling of the PF chambers, the average energy of the ion beam was 60–90 keV. The probabilities of the generation of several pulses of neutrons and X-ray with one switching on of the PF chamber are also considered. The histograms of the probability distribution for the occurrence of one, two, or three pulses of neutrons and hard X-ray based on experimental data are given. The greatest instability is observed for hard X-ray pulses: thus, for a PF chamber with a discharge current amplitude of ~ 300 kA in 25% of the responses, there was no hard X-ray pulse, and a double hard X-ray pulse was observed in 15% of the operations. Moreover, in the case of a double pulse, the form and duration of each separate hard X-ray pulse actually coincided with each other, $t1 / t2 = (1.00 \pm 0.04)$, which may speak in favor of the hypothesis about the development of several instabilities of the same type in pinch PF chambers.

Keywords: *plasma focus, pinch, pulsed neutron source*

Corresponding author: *bogolubov@vniia.ru*

Poster Presentations

European DEMO Divertor Cassette: Study of an Alternative Path of the Cooling Pipes Inside the Cassette Body Considering Piping Manufacturing Assessment

P1-001

Ugo Bonavolontà [a]*, Christian Bachmann [c], Domenico Coccorese [a], Giuseppe Di Gironimo [a], Vito Imbriani [a], Domenico Marzullo [a], Giuseppe Mazzone [b], Christian Vorpahl [c], Jeong-Ha You [d]

[a] *CREATE/University of Naples, Industrial Engineering, Napoli, Italy*
[b] *ENEA Frascati, Fusion & Technology for Nuclear Safety, Frascati, Italy*
[c] *Power Plant Physics & Technology, EUROfusion, Garching, Germany*
[d] *Max-Planck-Institute for Plasma Physics, Garching, Germany*

The pre-conceptual design of the DEMO divertor cassette with a novel, alternative path of the main cooling pipes inside cassette body is presented in this paper, focusing on cassette design and Plasma Facing Components (PFC) integration.

The divertor cassette design is reviewed, considering recent updates in the DEMO configuration model as presented by the Programme Management Unit (PMU) in 2018. The new configuration requires the cooling pipes to be integrated inside the cassette body. The components affected by these changes and the impact on the divertor design are analyzed. The study focuses on a new integration system between cassette and cooling pipes. The paper describes the integration on the new cassette geometry and the divertor sub-systems. The design activities related to this system are discussed in detail in terms of CAD modeling and considerations with respect to manufacturing such as welding technologies and non-destructive testing.

Keywords: *DEMO, divertor cassette, divertor target, cooling pipes*

Corresponding author: *ugo.bonavolonta@consorziocreate.it*

Manufacture of an ITER Full Scale First Wall Panel Prototype

P1-002

Delphine Bossu [a]*, Bruno Boireau [a], Humberto Miguel [a], Brice Pelletier [a], Robin Shuff [b], Stefano Banetta [b], Patrick Lorenzetto [b], Paul Edwards [c], René Raffray [c]

[a] *Framatome, Centre technique, Le Creusot cedex, France*
[b] *Fusion For Energy, Barcelona, Spain*
[c] *ITER Organization, Saint Paul Lez Durance, France*

This poster describes the main steps realized for the manufacturing of a full-scale Normal Heat Flux First Wall panel for ITER. This full-scale prototype (FSP) is foreseen to be delivered in 2019 to F4E in order to perform high heat flux tests. The dimensions of this prototype are 1360 mm x 850 mm x 500 mm. It consists of a bi-metallic support structure made from 15-25 mm thick CuCrZr alloy in which are embedded 316L(N)- IG tubes and which is bonded to a 40-50 mm thick 316L(N)-IG backing plate with cooling channels. The CuCrZr surface is coated with 784 Beryllium tiles.

The Technical Center of Framatome in Le Creusot used its competencies in conception and manufacturing engineering as well as lessons learned from the previous mocks-up to successfully manufacture and assemble this FSP. Processes using a number of technologies were carried out including diffusion bonding in solid state by Hot Isostatic Pressing (stainless steel to stainless steel, stainless steel to CuCrZr alloy), stainless steel and CuCrZr alloy machining (deep drilling, milling, etc.), welding, and bending.

Currently, the FSP has reached the stage whereby the entire structure has been fabricated and joined. The assembly progress as well as the preceding manufacturing steps will be highlighted. The final control stages involving helium leak testing, ultrasonic tests and especially 3D scan measurements will also be described.

Keywords: *first wall panel, prototype, manufacturing*

Corresponding author: *delphine.bossu@framatome.com*

The Simulation of the Quasi-Snowflake Divertor Configuration with the EAST New Upgrade Lower Tungsten Divertor Shape

P1-003

Chen Zhang [a], Chaofeng Sang [a*], Liang Wang [a,b], Zhengping Luo [b], Guosheng Xu [b], Daoyuan Liu [a], Dezhen Wang [a] and EAST team

[a] *Key Laboratory of Materials Modification by Laser, Ion and Electron Beams (Ministry of Education), School of Physics, Dalian University of Technology, Dalian, China*

[b] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*

Currently the graphite lower divertor of EAST limits the achievement of the long-pulse discharge due to its low heat flux toleration which need to be solved. The update of its graphite divertor with tungsten (W) as the plasma-facing material (PFM) is carried out. One of the most important goals for the divertor physical design is to reduce the heat flux as well as the plasma temperature to protect the target. To this end, two methods are adopted: 1. Design the divertor shapes to increase the divertor closure thus enhance the power dissipation capacity [1]; 2. Develop the advanced magnetic configuration to increase the flux expansion/connection length and reduce the peak heat flux [2]. In this work, the design and feasibility of the quasi-snowflake (QSF) divertor equilibrium for the EAST new designed lower divertor shape is presented and compared with the conventional magnetic configuration by using SOLPS-ITER simulation [3]. The performance of heat flux control, power radiation and impurities shielding of the QSF divertor are studied. The results indicate that QSF could significantly increase the connection length and flux expansion, thus reduce the heat flux loads. Since there are almost no intrinsic impurities in the W divertor, the impurity-seeding (Ne) is also considered in this work. The behaviors of the energy dissipation by the impurity, impurity transport and shielding in the QSF configuration is also presented.

[1] *C. F. Sang et al., Nucl. Fusion 59 (2017) 025009.*

[2] *Z. P. Luo et al., Fusion Eng. Des. 128 (2018) 90.*

[3] *X. Bonnin et al., Plasma Fusion Res. 11 (2016) 1403102.*

Keywords: *Divertor, QSF, heat flux, power radiation, impurity*

Corresponding author: *sang@dlut.edu.cn*

Design of the RFX-mod2 First Wall

P1-004

Mauro Dalla Palma [a*], Giovanni Berton [a], Alessandra Canton [a], Roberto Cavazzana [a], Giulio Gambetta [a], Paolo Innocente [a], Simone Peruzzo [a], Marco Siragusa [a,b], Silvia Spagnolo [a], Monica Spolaore [a]

[a] *Consorzio RFX, Padova, Italy*

[b] *Università degli Studi di Padova, Padova, Italy*

RFX-mod2 is the last upgrade of the reversed field pinch machine operated at Consorzio RFX. A significant modification consists of the replacement of the first wall tiles, proposed as a key factor for the improvement of the gas density with the reduction of the hydrogen retention, and designed in coherence with the magnetic front-end modification that foresees the tiles supported by the existing MHD passive stabilising shell. The main choices in the design of the new first wall tiles are the polycrystalline graphite as bulk material, the use of the existing fixing keys with fastening bayonets, and the tile width that shall be less than the diameter of the larger port holes to allow remote handling operations for the maintenance of the first wall; the latter fix the number of tiles to 2016 as in the original configuration.

The expected decrease of the plasma-wall interaction determined the first wall design with expected power densities up to 50 MW/m^2 considering the deformation of the last magnetic surface in both reversed field pinch and tokamak configurations. At the other side, the need of sensor integration and shielding of the passive stabilising shell from the plasma addressed the tile thickness and surface extension. Local prominences have been modelled on tiles, based on visual inspections of actual surfaces after previous operations, in order to limit the plasma in regions far from openings and supporting structures.

The tile resistant section has been increased coherently with all the interfaces and constraints, so decreasing the maximum stress at 3.5 MPa calculated from finite element analysis that simulates the operating condition. This low stress level together with a measurement of the experimental loads during next RFX-mod2 operations could qualify the use of extruded graphite for a further first wall change in the future. Indeed, extruded graphite is considered attractive given its high directional thermal diffusivity (about 50% better then polycrystalline graphite) to enhance the heat transmission and so improving the gas density control, and the low stress induced may allow this mechanically less performing grade of graphite.

Keywords: *polycrystalline graphite, plasma-wall interaction, thermal diffusivity, remote handling compatibility, high heat flux*

Corresponding author: *mauro.dallapalma@igi.cnr.it*

On the Numerical Assessment of the Thermal-Hydraulic Operating Map of the DEMO Divertor Plasma Facing Components Cooling Circuit

P1-005

Pietro Alessandro Di Maio [a]*, R. Gaglio [a], M. Giardina [a], J. H. You [b], G. Mazzone [c], E. Vallone [a]

[a] *Department of Engineering, University of Palermo, Palermo, Italy*
[b] *Max Planck Institute of Plasma Physics (E2M), Garching, Germany*
[c] *Department of Fusion and Technology for Nuclear Safety and Security, ENEA C.R. Frascati, Frascati (Roma), Italy*

Within the framework of the Work Package DIV 1 - "Divertor Cassette Design and Integration" of the EUROfusion action, a research campaign has been jointly carried out by University of Palermo and ENEA to investigate the thermal-hydraulic behaviour of the DEMO divertor cassette cooling system, focussing the attention on the latest configuration of the Plasma Facing Components (PFCs) circuit consistent with the DEMO baseline 2017. The research campaign has been carried out following a theoretical-computational approach based on the finite volume method and adopting the commercial Computational Fluid-Dynamic (CFD) code ANSYS CFX.

A steady-state CFD analysis has been carried out for the PFCs cooling circuit under nominal conditions and its thermal-hydraulic performances have been assessed in terms of coolant total pressure drop, flow velocity and Critical Heat Flux (CHF) margin distributions among the Plasma Facing Units (PFUs) channels, to check whether they comply with the corresponding limits. Results obtained have clearly predicted an acceptable total pressure drop (lower than 1.4 MPa) as well as a sufficient margin against CHF onset (higher than 1.4) within all the PFU channels.

Moreover, a parametric study has been performed in order to assess the operating map of the cooling circuit in the phase-space of coolant inlet temperature, pressure and mass flow rate, to be intended as that domain where the circuit thermal-hydraulic performances let it stay within the prescribed requirements.

Models, loads and boundary conditions assumed for the analyses are herewith reported and critically discussed, together with the main results obtained.

Keywords: *DEMO, divertor, Plasma Facing Components, CFD analysis, hydraulics*

***Corresponding author:** pietroalessandro.dimaio@unipa.it*

Plasma-Facing Components Damage and Its Effects on Plasma Performance in EAST Tokamak

P1-006

R. Ding*, D.H. Zhu, R. Yan, **Hai Xie**, S. Sarkar, C.J. Li, B.F. Gao, K. Zhang and J.L. Chen

Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, Anhui, China

During the tokamak operation, plasma-wall interactions can damage the plasma-facing components (PFCs) significantly, which can then degrade the plasma performance due to radiation losses by the impurities from the damaged materials. The PFCs damage severely limit the long pulse plasma operation in EAST. Hot spots due to high heat flux often appear on the PFCs and can then lead to massive impurities burst into the plasma. Since the graphite materials at lower divertor can only sustain 2 MW/m^2 heat loads, hot spots are easily formed during higher heating power operations. The 4.6 GHz LHCD power can also lead to hot spots at the antenna limiter at the midplane due to the fast electrons caused by enhanced RF sheath. The hotspots can produce a number of dusts which easily result in disruptions eventually. With the measurements by fast visible cameras the strong correlation of the dust rate and plasma displacement has been observed. It is also found that dust events were significantly lower with the ICRF heating. For the actively water cooled tungsten upper divertor which can sustain 10 MW/m^2 heat loads, hot spots can still be often observed at the edge of tungsten mono-blocks. After 2017 campaign, the tungsten melting occurred around the strike point on both inner and outer divertor targets for more than 20 locations. All the melting occurred at the edges of cassette bodies, where the misalignment can exceed 1 mm. Due to the balance between the gravity and the electromagnetic force, the migration distance of melted layer is not significant. However, the ejection of W dusts induced by W melting into the plasma and thus disruptions have been observed in some discharges. After the caused disruptions, the plasma performance could be well restored to normal conditions after several discharges. Therefore, the melting behaviors can be tolerated for current power condition. Detailed analysis on the PFCs damage in EAST will be presented, which can help to protect the PFCs for long pulse and high performance operations in the future.

Keywords: *Tungsten, Melting, Dust, Plasma-Facing Component, EAST tokamak*

***Corresponding author:** rding@ipp.ac.cn*

W Enrichment by Preferential Sputtering of EUROFER 97 as Actively Cooled Target

P1-007

Stefan Elgeti [a]*, Martin Balden [a], Bernd Böswirth [a], Henri Greuner [a], Rudolf Neu [a,b], Thomas Schwarz-Selinger [a]

[a] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[b]*Technische Universität München, Garching, Germany*

In the research for future fusion power plants the aptitude of steel as a plasma facing material (PFM) in moderate heat and particle loaded components of the first wall is investigated. Steel is an attractive candidate compared to W materials and offers favourable economic, mechanical, and manufacturing properties. However, the higher erosion rate of iron compared to W is a problematic issue which could be reduced by the effect of preferential sputtering of Fe compared to W contained in the reduced activation ferritic-martensitic steel EUROFER 97 (E97). The aim of this contribution is to investigate the W enrichment in E97 under conditions similar to PFM loading in fusion reactors by loading with an H-beam with an average energy of 10 keV relating to the high energy of charge exchange neutrals as the worst case for the W enrichment ratio. Therefore an actively cooled mockup made from E97 was loaded in sequential fluence steps up to 10^{25} H/m^2 at the high heat flux test facility GLADIS with 2 MW/m^2 resulting in ~480°C surface temperature close to the permissible tested operational temperature for E97. The same marked surface region of the mock-up was analysed iteratively after the fluence steps with scanning electron microscopy by imaging and focus ion beam cross-sectioning as well as energy dispersive x-ray spectroscopy (EDS) maps. The depth of EDS-analysis is minimised to about 50 nm by using an electron beam energy of 5 keV in order to enhance the sensitivity of surface effects. From these EDS maps an enrichment factor of about 4 can be estimated in good agreement with 4.8 which can be simply calculated with sputter yields of pure Fe and W. Rutherford back scattering depth profiling yields an enriched depth zone of 30 nm confirming the approach with EDS. Such enriched depth zone points to a possible stirring effect by incident H ions, which is not expected by simulations showing only atomic surface enrichment.

Keywords: *Eurofer steel, plasma-facing material, erosion, W enrichment, preferential sputtering*

Corresponding author: *stefan.elgeti@ipp.mpg.de*

Design and Manufacturing of Bulk Tungsten Tiles for WEST Outer Limiter

P1-008

Mehdi Firdaouss *, P. Chappuis, M.Diez, S. Gazzotti, M. Lipa, C. Portafaix

IRFM, CEA Cadarache, SAINT PAUL LEZ DURANCE, France

In the WEST tokamak, RF heating systems are protected by a movable limiter designed to avoid that off normal events during the early phase of the discharge, such as runaways electrons, can impact the RF systems. During the start-up phase of a pulse, it is located close to the plasma, in front of the RF heating systems. The limiter is designed to sustain up to 10 MW/m² for a few seconds. It is retracted behind the RF systems when the discharge is stabilized, to minimize the potential W source in the plasma and optimize the RF systems coupling.

The front face of the limiter is made of W coated tiles, which are bolted onto a CuCrZr structure that is water cooled. This structure is assembled on a sliding support.

For the first phase of WEST operation, it has been decided to reuse the CFC tiles coming from Tore Supra, with a 100µm tungsten coating on a 100µm molybdenum interlayer, in order to comply with the metallic environment.

Strong degradation of the coatings has been observed after the second experimental campaign. In particular delamination occurred on the lateral faces of some tiles, and large melted areas (up to 10cm²) have been observed on the front face. This melting has been attributed to runaway electron beams hitting the limiter; the control scheme to avoid runaways was under development during this campaign. Consequently, the most damaged tiles have been replaced during a shutdown. After the third experimental campaign, where runaway electrons were better controlled, fewer damages were observed.

However, as the power coupled to the plasma and the duration of the pulse is expected to increase during the second phase of WEST operation, decision has been made to replace the W coated CFC tiles by bulk W tiles, while keeping the external geometry similar.

This material change is not straightforward. Indeed, as shown by AUG, the main challenge for a W tile bolted to a cooled support, is the appearance of cracks during operations, mainly due to the combination of high thermal loads and large electro-mechanical forces.

To avoid this issue, thermo-mechanical and electro-magnetic simulations have been performed and analyzed for WEST parameters.

The final design includes the use of an elastic clamping system and poloidal castellation. The manufacturing of the 36 tiles has been started, and after reception controls, tiles will be installed in WEST and be ready for operation restart in May 2019.

[1] *A. Herrmann et al., Nucl. Mater. Energy, 12 (2017), pp. 205-209*

Keywords: *bulk tungsten, tungsten coating, plasma facing component, design*

Corresponding author: *mehdi.firdaouss@cea.fr*

The EU DEMO Equatorial Outboard Limiter - Design and Port Integration Concept

P1-009

Thomas Franke [a,b]*, Christian Bachmann [a], Wolfgang Biel [c], Fabio Cismondi [a], Oliver Crofts [d], Aljaz Cufar [e], Gianfranco Federici [a], Winder Gonzalez [c], Richard Gowland [d], Gregory Keech [a], Rocco Mozzillo [f], Francesco Maviglia [a], Massimo Roccella [g], Mikhael Tokar [c], Zsolt Vizvary [d]

[a] *EUROfusion Power Plant Physics and Technology (PPPT) Department, Garching, Germany*
[b] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[c] *Institute of Energy and Climate Research, Forschungszentrum Jülich GmbH, Jülich, Germany*
[d] *Culham Centre for Fusion Energy, Abingdon, Oxfordshire, UK*
[e] *Reactor Physics Department, Jožef Stefan Institute, Ljubljana, Slovenia*
[f] *CREATE, University of Naples Federico II, DII, Naples, Italy*
[g] *LTCalcoli Srl, 23807 Merate (LC), Italy*

The equatorial outboard limiter is an essential part of the DEMO wall protection concept. Limiters are foreseen in different areas of the DEMO first wall, namely in the equatorial ports, on high-field side, in vertical ports and additional small protection limiters between equatorial and lower port. The limiters shall prevent thermal overloads in the first wall of the breeding blanket during all transient and steady state plasma phases, for a) normal plasma contact, like the limiter configuration while plasma ramp-up, b) for all off-normal plasma events, like disruptions and vertical displacement events and c) in case of possible other control or machine failures leading to a loss of plasma control.

The equatorial limiter port plug pre-conceptual design for DEMO and the limiter port integration concept and its rationales behinds are explained taking into account i) thermal, structural and electromagnetic loads, ii) neutronic requirements and related material properties, iii) remote handling considerations, iv) space and mass constraints and v) required alignment precision to allow equal distribution of the thermal heat dissipation of the plasma facing (PFC) limiter components.

While the hot fusion plasma is impinging directly on the limiter, its temperature is raising and by means of infrared (IR) thermography an estimation of the heat flux on the contact point can be made and the temperature of the PFCs can be observed. Also a visual (VIS) spectroscopy shall be implemented in order to ensure the safe limiter operation and a long lifetime. For two different configurations of number and locations of the limiters around the DEMO tokamak torus a performance check of different cases for IR thermography observation area and number of sightlines is presented.

Keywords: *DEMO, equatorial limiter, infrared thermography, wall protection, ramp-up limiter*

Corresponding author: *thomas.franke@euro-fusion.org*

Structural Analysis on EU-DEMO Divertor Cassette Attachments Subjected to Thermal and Electric-Magnetic Loads

P1-010

Paolo Frosi [a]*, C. Bachmann [b], P.A. Di Maio [c], **Domenico Marzullo** [d], G. Mazzone [a], S. Villari [a], C. Vorpahl [b], J.H. You [b]

[a] *Department of Fusion and Technology for Nuclear Safety and Security, ENEA C. R. Frascati, Frascati (Roma), Italy*
[b] *Max Planck Institute for Plasma Physics, Garching, Germany*
[c] *University of Palermo, Viale delle Scienze, Palermo, Italy*
[d] *Department of Industrial Engineering (DII), CREATE Consortium/University of Naples Federico II, Napoli, Italy*

EU-DEMO is under Pre-Conceptual Design Review promoted in the framework of EUROfusion Consortium conceived to continue the study about nuclear fusion undertaken with ITER experiment. The EU-DEMO consists of several topics and phases: the structural analysis is one among them as the reactor components need either the design or the assessment activities. The Divertor is the main plasma-facing component inside vessel as it must withstand and remove the power exhaust and impurity particles. An important challenge of DEMO Divertor design is the cassette attachment system as it must be studied considering the loads evolution during the whole pulse. The dimensioning loads of Divertor Cassette supports are the electric-magnetic forces. This paper deals with the detailed analysis of the structural behavior of the inboard and outboard supports considering the last definition of the cassette geometry and the most up-to-date loads evaluations: all the forces/moments components have been analyzed acting separately and together to distinguish the different contributions. The differential displacement of the cassette and the vacuum vessel is considered in normal operations and baking conditions evaluating the kinematic behavior and the stiffness of the so-called knuckle (cassette outboard support). The real function of hinges, the heat transfer from cassette towards knuckle and the unilateral contact behavior between bodies in outboard cassette region have been reproduced in a realistic manner.

Keywords: *FEM, DEMO, Divertor, structural analysis, thermal analysis, RCC-MRx*

Corresponding author: *paolo.frosi@enea.it*

Development and High-Heat-Flux Test Results of a DEMO Divertor Target Concept with a Thick Graded Bond Interlayer

P1-011

Marc Missirlian [a], F. Gallay [a]*, M. Richou [a], A. Quet [b], I. Chu [c], S. Roccella [d], G. Dose [e], H. Greuner [f], B. Böswirth [f], G. Kermouche [g], M. Lenci [g], E. Visca [d], J.H. You [f]

[a] *CEA, IRFM, Saint-Paul-Lez-Durance, France*
[b] *CEA Le Ripault, Monts, France*
[c] *CEA, LITEN, Grenoble, France*
[d] *Department of Fusion and Technology for Nuclear Safety and Security, ENEA, Frascati, Italy*
[e] *Università di Roma "Tor Vergata", Dipartimento di Ingegneria Industriale, Roma, Italy*
[f] *Max Planck Institute for Plasma Physics, Garching, Germany*
[g] *Ecole des Mines de Saint-Etienne, LGF UMR5307 CNRS, Saint-Etienne, France*

In fusion devices, one of the most critical parts of Plasma Facing Components (PFCs) is the divertor target. In the framework of the EUROfusion DEMO divertor project, some preconceptual design activities have been launched since 2014. One design concept integrates functionally graded material as an interlayer between tungsten monoblock and CuCrZr cooling tube. Thin (~25µm) and thick (~500µm) interlayers composed of functionally graded material are being investigated with some beneficial aspects for the both studied cases. While the use of a thin interlayer reduces armor temperature and may also mitigate plastic fatigue under irradiation, a thick one plays an essential compliant role, especially at the bonding interface where thermal stresses tend to be concentrated.

To study the impact of thick graded interlayer on thermomechanical behaviour, 3 mock-ups were manufactured and tested. This paper presents the recent achievements for the realization and testing of these mock-ups under DEMO relevant heat fluxes (up to 20MW/m²). In this frame, 3 tubes (composed of several layers with different W/Cu compositions) were manufactured via thermal spray and finally bonded in two steps with hot isostatic pressing to form the mock-ups. The influence of W/Cu concentration on the bonding quality was assessed with manufacturing three different configurations of the graded material finishing coat (25%W+75%Cu, 50%W+50%Cu and 75%W+25%Cu). Some optimizations of the bonding process showed that high temperatures (1000°C) are required to bond graded material both to tungsten blocks and to CuCrZr tube. Bond quality of the mock-ups was assessed by infrared thermography (SATIR facility) and ultrasonic non-destructive examinations. For the three manufactured mock-ups, 3 over the 12 bonded W blocks show debonding at the interface. Consistency of the results for the two non-destructive techniques was observed and good bonding quality could be established for all the coating configurations, with the only exception of the one with 75%W+25%Cu. Finally, results of the high heat flux tests carried out in GLADIS facility are reported.

**Corresponding author: franklin.gallay@cea.fr*

Status of the ITER Divertor IVT Procurement

P1-012

Pierre Gavila [a]*, Marco Grattarola[b], Massimiliano Palermo[b], Bruno Riccardi[a], Eliseo Visca[c], Vladimir Kuznetcov[d], Andrew Volodin[d], Alain Durocher[f], Andrey Fedosov[f]

[a] *F4E, Barcelona, Spain*
[b] *Ansaldo Nucleare, Genova,Italy*
[c] *ENEA CR-Frascati, Frascati, Italy*
[d] *NIIEFA, 3 doroga na Metallostroy, Metallostroy, St. Petersburg, Russia*
[f] *ITER Organization, St. Paul Lez Durance, France*

In the frame of the effort to enhance competition among the potential bidders and secure the procurement of the ITER Divertor Inner Vertical Target (IVT), F4E placed a total of four contracts aiming at the manufacturing and testing of IVT Prototypes. The companies involved are Ansaldo Nucleare (Italy), Atmostat-Alcen (France), CNIM (France) and Research Instruments GmbH (Germany).

The Ansaldo Nucleare Prototype manufacturing is under completion. The stainless steel support structure was manufactured by Walter Tosto (Italy) and the Plasma Facing Units (PFUs) were manufactured by ENEA (Italy) by diffusion bonding between the CuCrZr pipe and the W monoblocks equipped with a pure Cu compliance layer (Hot Radial Pressing).

This paper presents some of the learnings of the IVT Prototype manufacturing and provides the main results of the acceptance tests. The outcomes of the PFUs High Heat Flux (HHF) testing performed in the ITER Divertor Test Facility in Efremov institute, Saint Petersburg (Russia) are presented as well.

The main conclusions are the following ones:

- The HRP provides a robust solution for the armor to heat sink joint, in terms of resistance to the HHF testing,
- The tube to tube transition between the Cu alloy and stainless steel pipes by fusion welding meets the IO requirements on mechanical resistance and tightness,
- The tight geometrical tolerances constitute one of the main challenges for the IVT manufacturing. The good results obtained on the ANN Prototype will have to be further improved in order

to avoid the final machining of the W plasma facing surface while ensuring full compliance with the requirements,

- The "self-castellation" appearance previously observed on small scale mock-ups are confirmed on the full scale PFUs at a similar rate. These "self-castellations", which appeared during the testing at 20 MW/m^2, do not affect the monoblocks heat removal capability, but could have a detrimental impact on the monoblocks lifetime due to the creation of leading edges exposed to the direct particles flux in the strike point region,
- The W manufacturing route and surface treatment are confirmed to have a significant influence on the "self-castellations" formation. Some refinement of the specifications to W Suppliers and IVT manufacturers are necessary in view of the IVT Series manufacturing.

Disclaimer: *The views and opinions expressed herein do not necessarily reflect those of the ITER Organization*

***Corresponding author:** pierre.gavila@f4e.europa.eu*

Fabrication of ITER Divertor Cassette Body Prototypes

P1-013

Laurent Guerrini [a*], Bruno Riccardi [a], José Andrade [a], Frederic Escourbiac [b], Montse Felip [a], Laurent Ferrand [b], Patrick Lorenzetto [a]

[a] *Fusion for Energy, Barcelona, Spain*

[b] *ITER Organization, St. Paul-Lez-Durance, France*

Fusion for Energy (F4E), the European Domestic Agency for ITER, is responsible for a significant share of the overall procurement of ITER divertor, namely the inner vertical target (IVT) and the cassette body (CB). Both IVT and CB have been supported by extensive Research and Development (R&D) work programmes, implemented over more than 15 years in Europe, to validate design, develop various fabrication technologies, assess thermohydraulical performances and RH installation and maintainability.

The CB procurement arrangement between F4E and ITER Organization (signed in 2012) foresees, before the launch of the production, the validation of the CB design. The CB tight fabrication tolerances and the requirement to perform 100% volumetric inspection of welds, as per applicable standards, require a formal pre-qualification programme that needs to be performed prior to the start of the series production. A procurement strategy has been implemented by the In-Vessel Project Team at F4E aiming at mitigating technical and commercial risks for the procurement of CB fostering, as far as possible, competition among industrial partners. Following an open call, three framework contracts were awarded at the end of 2013 to the consortium CNIM-SIMIC, VALKON METALLI and WALTER TOSTO for the manufacture of a full-scale prototype to demonstrate their capabilities to fabricate such a component with the required quality. The successful companies will be allowed to bid for the series production. In 2016 VALKON METALLI withdrew from the competition.

The supplying companies chose to manufacture their prototype by conventional fabrication techniques, based on machining and welding of 316L(N)-IG and XM-19 austenitic stainless steel forgings and plates. The CB prototype fabrication has requested many manufacturing steps followed by dimensional surveys to control the distortions after welding and heat treatments. The implementation of ultrasonic and radiographic testing for the specified volumetric inspection of welds required as well the development and the performance of an intensive prior qualification program. Final machining operations lasted for some months due to the stringent fabrication tolerances specified for the CB which is a 2.5m x 3.6m x 0.5m component. At the beginning of 2018 the two companies have successfully completed their full-scale prototypes.

This paper presents in detail the work performed and the technical challenges related to the fabrication of the two CB prototypes, the results of the prototype qualification as well as the lesson learnt and suggested design improvements.

***Keywords:** ITER, Plasma facing components, Divertor Cassette*

***Corresponding author:** laurent.guerrini@f4e.europa.eu*

The Runaway Electron Evolution in the DEMO Reactor Plasmas

P1-014

Yuri Igitkhanov [a,b*], Boris Bazylev [a,b], Sergey Pestchanyi [a]

[a] *Karlsruhe Institute of Technology, Karlsruhe, Germany*

[b] *South Ural State University, Chelyabinsk, Russia*

The runaway electrons (RE) in tokamaks reactor plasma could pose a serious problem for the plasma facing components (PFC) causing a strong erosion of materials in regimes with dominant avalanche generation. The expected PFC damage becomes particularly important, when the RE carry a considerable amount of thermal and magnetic energy. Therefore, the nature of avalanche generation in reactor boundary plasmas is a matter of considerable importance. It was found [1] that highly energetic runaway electrons are able to penetrate the electron shell of partly ionized heavy ions during collisions, for which reason they may be scattered by a positive charge effectively larger than that of a shielded nucleus. This effect increases the Coulomb cross section and can be treated via an effective ion charge Zeff (ε_{kin}) that depends on the energy of the incident electrons ε_{kin}. The increase of effective charge number with increasing electron energy in multi-component plasmas renders qualitatively the same result as high Zeff Coulomb plasmas. Since the generation rate of runaways depends on

Z_{eff}, its production during the mitigation of disruptions by massive gas injection could in some cases decrease owing to a heavy impurity concentration in the boundary tokamak plasma.

In this work we consider the expected evolution of the RE in DEMO during the vertical displacement events and during the massive gas injection expected for reduction of avalanche production due to non-Coulomb collision.

[1] Yu. Igitkhanov, Contrib. Plasma. Physics, 52 (2012) 319.

Keywords: *DEMO, Runaway electrons, Material erosion, vertical displacement, Massive gas injection*

Corresponding author: *juri.igitkhanov@partner.kit.edu*

Divertor Design and Integration Activities for the High-Field ST40 Spherical Tokamak

P1-015

Daniel Iglesias [a]*, Christos K. Stavrou [a], Otto Asunta [a], Peter Buxton [a], Simon Hanks [a], Damian Lockley [a], Jeppe Olsen [b], Ed Pinkney [a], Enrique Ruiz de Villa [a], Glenn Whitfield [a], Jonathan Wood [a], Steve McNamara [a], Mikhail Gryaznevich [a], Rob Slade [a], Paul Thomas [a]

[a] *Tokamak Energy Ltd., Abingdon, UK*

[b] *PPFE, Department of Physics, Technical University of Denmark, Lyngby, Denmark*

Tokamak Energy Ltd, UK, is developing a possible route to fusion power based on spherical tokamaks with high temperature superconductor (HTS) magnets. ST40 is an operational double-null tokamak with copper magnets, intended to operate at fields up to $B_t = 3$ T at a major radius of $R_0 = 0.4$ m. The next device, ST-F1, will use HTS magnets and be capable of producing an industrial scale of fusion power.

The design of the ST40 divertor system is driven by four high-level requirements: to facilitate the plasma performance operation point, to expand the experimental characterization of the SOL at high-field, to evaluate target technologies for next step devices, and to maintain its integrity during the whole experimental campaign with up to 4 MW of plasma auxiliary heating. The design space is further constrained by the interface requirements given by the diagnostics, supports, vacuum chamber, stabilization plates and other tokamak components.

The design consists on a symmetrical up-down semi-closed divertor configuration, with baffles providing the space for efficient divertor pumping. Special target surfaces are located at the experiment sectors for investigating SOL falloff width, spreading, ELM energy density, turbulent transport, and heat flux projection. Specific diagnostics and analysis codes are being developed within an integrated plan for characterizing electron temperature and density, impurity emissions, neutrals pressure, divertor plasma radiation, target surface temperatures and SOL parallel heat flux density. This joint development of design, operations, and analysis tools allows the assessment to all performance, functional, and interface requirements of the divertor system towards the delivery of the ST40 experimental plan and the evaluation of ST-F1 technology.

Keywords: *divertor, spherical tokamaks, integration, experiment planning*

Corresponding author: *daniel.iglesias@tokamakenergy.co.uk*

Thermal and Structural Analyses of Retro-Reflectors of ITER Poloidal Polarimeter That Are Mounted on First Wall Panels

P1-016

Ryota Imazawa [a]*, Takehiro Ono [a], Christopher Watts [b], Masao Ishikawa [a], Takahiko Shimada [a], Kiyoshi Itami [a], Takaki Hatae [a]

[a] *National Institutes for Quantum and Radiological Science and Technology, Ibaraki, Japan*

[b] *ITER Organization, St. Paul-lez-Durance, France*

A poloidal polarimeter system will be installed in ITER to identify the safety factor profile in the plasma core region by measuring polarization change of a probing laser beam passing through plasma. Far-infrared laser beam, injected in a plasma, is reflected by a corner-cube retro-reflector mounted on first wall panels and returns to a diagnostic room. Although the corner-cube retro-reflectors will be exposed to high neutron and radiation fluxes, they will be passively cooled by heat conduction to the first wall panels. The challenging of designing the corner-cube retro-reflector is (1) cooling capability and (2) difference of thermal expansion of different materials. The first wall panel is made of stainless steel, while the corner-cube retro-reflectors is made of tungsten because of high resistivity to plasma sputtering. A special adapter is necessary for tightly fixing tungsten retro-reflectors and strongly touching the stainless-steel wall (the first wall panel) for the sake of high thermal conduction. However, the coefficient of thermal expansion of tungsten and stainless steel is different by a factor of four. It is difficult to touch both the tungsten retro-reflectors and stainless-steel wall with high contact pressure. The authors developed new adapter design consisting of two parts. One part takes care of fixing the position of tungsten retro-reflectors and the other part takes care of good thermal conduction. Thermal and structural analysis results show that the new design can meet the design criteria of ductile damage, plastic damage and thermal creep. A prototype of the new design is under fabrication to demonstrate the manufacturability.

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

Keywords: *ITER, poloidal polarimeter, retro-reflector*

Corresponding author: *imazawa.ryota@qst.go.jp*

Insulated Fixation System of Plasma Facing Components to the Divertor Cassette in Eurofusion-DEMO

P1-017

Vito Imbriani [a]*, Ugo Bonavolontà [a], Giuseppe Di Gironimo [a], Samir El Shawish [b], Mike Fursdon [c], Louis Giannone [d], Domenico Marzullo [a], Giuseppe Mazzone [e], Eliseo Visca [e], Jeong-Ha You [d]

[a] *CREATE/University of Naples, Industrial Engineering, Napoli, Italy*
[b] *Jozef Stefan Institute, Ljubljana, Slovenia*
[c] *Department of Central Engineering, Culham Centre for Fusion Energy, Abingdon, UK*
[d] *Max Planck Institute of Plasma Physics, Garching, Germany*
[e] *Department of Fusion and Technology for Nuclear Safety and Security, ENEA, Frascati, Italy*

The Eurofusion-DEMO divertor is a key in-vessel component with plasma facing components (PFCs) which directly interact with the plasma scrape-off layer.

The PFCs have to cope with high heat loads, neutron irradiation and electromagnetic loads.

The mechanical integrity of the PFCs and water cooling pipes can be jeopardized by currents generated in a disruption event producing forces in the toroidal magnetic field.

It is therefore necessary to study in detail the fixation system of the PFCs to the divertor cassette from the structural point of view in order to ensure electromagnetic loads are limited to safe values.

Thermocurrent measurements with a shunt are needed for feedback control of plasma attachment in Eurofusion-DEMO. To make these measurements possible, some PFC's need to be electrically insulated from the divertor cassette.

This paper presents the design activities of an insulated and not insulated PFC-Cassette Body support developed under the pre-conceptual design phase for Eurofusion-DEMO Work Package DIV-1 "Divertor Cassette Design and Integration" - Eurofusion Power Plant Physics & Technology (PPPT) program. In particular, the possible use of ceramic material (e.g. alumina, silicon nitride) as the insulating layer between the support components is investigated.

Keywords: *Eurofusion-DEMO, Divertor assembly, Divertor Plasma Facing Components fixation system*

Corresponding author: *vito.imbriani@consorziocreate.it*

Hydrogen Permeation Behavior Through Tungsten Deposition Layer Growing on Nickel Substrate by Hydrogen Plasma Sputtering

P1-018

Hideki Ito*, Kazunari Katayama*, Daisuke Mori, Makoto Oya

Department of Advanced Energy Science, Kyushyu University, Kasuga, Fukuoka, Japan

Understanding of hydrogen isotope behaviors in plasma-facing wall is important from viewpoints of fuel control and tritium safety. Tungsten (W) is a candidate material of plasma-facing components. Although a sputtering rate of W is low, a certain amount of W deposition layer would be formed on the plasma-facing wall during a long time operation of a fusion reactor. The present authors have been studying hydrogen isotope retention and desorption behavior in W deposition layer formed by hydrogen plasma sputtering. In the recent work, hydrogen gas driven permeation through W deposition layer formed on nickel (Ni) plate was investigated and it was found that hydrogen diffusivity is smaller than that in W bulk and hydrogen solubility was much larger than that in W bulk. Thus, as the W deposition layer is formed on the plasma facing surface, the fuel accumulation and permeation behavior may be changed with the reactor operating time. In this work, a Ni permeation port was installed at the W deposition position in the hydrogen plasma sputtering device, and the change to the hydrogen permeation flux with the plasma discharge time was observed.

The standard discharge condition was RF power of 100 W, hydrogen pressure of 10 Pa. Under this condition, hydrogen ion flux at Ni position was about 1×10^{19} 1/m^2/s, and the temperature of Ni plate rose to about 80 °C due to heat load from the plasma. The thickness of Ni plate was 20 μm.

For a while after starting plasma discharge, the hydrogen permeation flux gradually increased and showed a peak. After that, it decreased gradually, and eventually continued to reduce in proportion to the discharge time. The peak time is considered to be the time when the thickness of the W deposition layer became thicker than the hydrogen implantation range. After that time, the permeation flux decreased because the hydrogen diffusivity in the W deposition layer is smaller than that in Ni. Eventually, it decreased in proportion to the thickness of the W deposition layer. The permeation flux was on the order of 10^{16} 1/m^2/s which was 1/1000 of the incident ion flux

Keywords: *Tungsten, deposition, sputtering, permeation*

Corresponding author: *kadzu@nucl.kyushu-u.ac.jp*

W_2C-Reinforced Tungsten: A Promising Candidate for High-Heat-Flux Material

P1-019

Petra Jenuš [a]*, Matej Kocen [a,b], Anže Abram [a], Andreja Šestan Zavašnik [b,c], Andrei Galatanu [d], Elena Tejado [e], Jose Ygnacio Pastor [e], Marius Wirtz [f], Gerald Pintsuk [f], Saša Novak [a]

[a] *Department for Nanostructured Materials, Jožef Stefan Institute, Ljubljana, Slovenia*
[b] *Jožef Stefan International Postgraduate School, Ljubljana, Slovenia*
[c] *Center for Electron Microscopy, Jožef Stefan Institute, Ljubljana, Slovenia*
[d] *National Institute of Materials Physics, Magurele, Romania*
[e] *Departamento de Ciencia de Materiales-CIME, Universidad Politécnica de Madrid, Spain*
[f] *Institute for Energy and Climate Reseach, Forschungszentrum Juelich GmbH, Juelich, Germany*

Tungsten has good high-temperature physical properties. However, it is also associated with a reduction in its mechanical properties in particular at and above the recrystallization temperature. The main aim of this work is to improve the tungsten's properties to sustain plasma-facing conditions in the divertor. Among the available options, we selected the reinforcement of tungsten with carbide nanoparticles (W_2C), wherein the reinforcement should not chemically react with the matrix.

Carbide particles in W-xW_2C composites (0 ≤ x ≥ 40 wt%) were formed *in-situ* during the thermal treatment of powder mixtures consisting of W and WC particles at various ratios with a field assisted sintering technique (FAST). If more than 4 at % of carbon in the form of WC nanoparticles was added to the starting mixture of powders, only two phases were detected in the sintered composites, namely cubic W and hexagonal W_2C. In addition to the microstructural and phase analysis, thermo-mechanical properties at room and elevated temperature, as well as high-heat-flux tests, were carried out. Results of thermo-mechanical measurements indicate that small amount (i.e. 5 wt %) of W_2C reinforcements improves mechanical properties of the material, while the thermal conductivity of the composite does not drop below 100 W/m K also at elevated temperatures (up to 1000 °C).

Additionally, the as-sintered samples were aged at 1250 °C and 1600 °C for 24 h. The microstructural analysis confirmed that the presence of small W_2C grains enhances densification of tungsten produced via FAST and inhibits grain growth at temperatures up to 1600 °C.

Keywords: *divertor, tungsten, mechanical properties, high-heat-flux tests, ageing*

Corresponding author: *petra.jenus@ijs.si*

Simulation Study of Evolution of Plasma-Related Defects in W

P1-020

Minyou Ye *, Jie Zhan

Department of Engineering and Applied Physics, School of Physical Sciences, University of Science and Technology of China, Hefei, Anhui, China

Tungsten (W) is considered as a primary candidate material for plasma facing components, which endures high fluence plasma in future fusion reactor. Extremely insoluble helium (He) introduced into W exposed to high fluence He plasma tend to agglomerate into bubbles, which play a significant role in the radiation-induced micro-structural evolution and properties degradation. Therefore, to understand the relevant underlying physics, especially the micro-structural evolution process dominated by He defects interactions, is of fundamental importance to the development of plasma facing materials and components. Because in-situ observation is difficult by present experimental techniques, molecular dynamics (MD) simulations have been widely employed to study the atomic-level evolution process of He bubbles in W. Moreover, the high thermal conductivity of W (175 $W \cdot m^{-1}k^{-1}$ at room temperature) is one of main reasons for its selection. However, the recent experiments indicated that the thermal conductivity of W irradiated by He plasma was reduced by 80% [1]. Atomistic simulations have been employed to investigate the atomic-level evolution process of He bubbles in bulk W and their influence on lattice thermal conductivity. In this presentation MD simulations were used to study the effects on the early stage of He bubbles formation with the He concentration up to 1at% and temperature (300-2100 K). The results show that the size of He-V clusters increases with increasing in the irradiation temperature and He concentration. The sizes of He bubbles obey the normal distribution. He defects interactions result in W interstitial atoms and vacancies with different types and sizes. Furthermore, it was firstly performed to study the equilibrium He/V ratios of nano-sized He bubble in bulk W from pressure perspective. It is revealed that the equilibrium He/V ratios related with its size and irradiation temperature are generally ranged from 1.6 to 2. We also observed the He atoms in small He bubble may be trapped into the large He bubble in the case of their distance below 2 lattice parameters, which would cause the decrease of He/V ratio of small bubble. Moreover, we evaluate the influence of nano-sized He bubbles with different shapes (ellipsoid and sphere) and He/V ratio up to 3 on the lattice thermal conductivity of W using Muller-Plathe non-equilibrium MD simulations. The lattice thermal conductivity of W is also compared with the effective thermal conductivity estimated by Maxwell's effective medium model. It is found that the lattice thermal conductivity is strongly related with the shape and He/V ratio of He bubbles in bulk W. These results can help to the understanding for nucleation and growth of He bubble as well as the influence on thermal conductivity in bulk W that observed by the experiment techniques [1-3].

[1] S. Cui, M. Simmonds, W. Qin, et al., J. Nucl. Mater. 486, 267 (2017).

[2] M.Y. Ye, N. Ohno, S. Takamura, J. Plasma Fusion Res. Ser. 3 (2000) 265.

[3] M.Y. Ye, S. Takamura, N. Ohno, J. Nucl. Mater. 241–243 (1997) 1243.

Keywords: *tungsten, plasma-related defects, thermal conductivity, molecular dynamics*

***Corresponding author:** *yemy@ustc.edu.cn*

Experimental Study of Hypervapotron and Cylindrical Channel for Divertor Cooling by One-side, Electric Joule Heating System

P1-021

Gi Cheol Lee, **Jihwan Lim**, Don Koan Hwang, Mingyu Park, Hungyo Oh, Moo Hwan Kim*

Pohang University of Science and Technology, Nam-Gu, Korea

Beyond ITER, demonstration of fusion reactor should be investigated for production of electrical power and commercialization. One of the technical issues on operating DEMO reactor is removing the high heat flux applied to Divertor system. In this study, we developed an electric joule heating system capable of one-side heating over 10 MW/m^2 with area of 27 mm * 100 mm for evaluating a cooling channel in Divertor system. The heating material is selected as FeCrAl (400 μm thickness) on owing to its melting point, thermal expansion rate, oxidation resistance and thermal conductivity. For the electric insulation of the CuCrZr channel, thin Al_2O_3 insulator is inserted between channel and heating element. Compression plates (Al_2O_3 and SiO_2) and damper system were used to minimize the thermal resistance between heating element and cooling channel. Developed one-side, electric joule heating system successfully applied the heat flux to cooling channel over 10 MW/m^2 and finally we conducted the subcooled flow boiling experiments with water coolant on Hypervapotron and cylindrical channel.

Keywords: *Critical Heat Flux, High Heat Flux Heater System, Mechanical Compression, Thermal Resistance*

***Corresponding author:** *mhkim@postech.ac.kr*

Improved Safety for DEMO by Advanced Tungsten Alloys as First Wall Armor

P1-022

Felix Klein [a]*, A. Litnovsky [a], M. Gilbert [b], J. Schmitz [a,c], K. De Lannoye [a,d], J. Gonzalez-Julian [a], T. Wegener [a], Ch. Linsmeier [a], J. W. Coenen [a], M. Bram [a]

[a] *Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung, Jülich, Germany*

[b] *CCFE, United Kingdom Atomic Energy Authority, Culham Science Centre, Abingdon, Oxfordshire, UK*

[c] *Department of Applied Physics, Ghent University, Ghent, Belgium*

[d] *Physics Department, Vrije Universiteit Brussel, Elsene, Belgium*

Tungsten (W) is a prime candidate as first wall armor material of future fusion power plants, as W withstands extreme particle and heat loads. Neutrons from the plasma activate the W during regular operation. This activation is a safety issue in case of a loss-of-coolant accident (LOCA) where the cooling systems fail and air ingress into the vacuum vessel occurs. Temperatures between 1200 K and 1450 K for several weeks are predicted due to the nuclear decay heat in such an accident. The radioactive W oxidizes and sublimates, posing a severe safety hazard for the environment.

Current research focusses on W-based alloys with the alloying elements chromium (Cr) and yttrium (Y) consolidated by Field Assisted Sintering Technology (FAST). The Y concentration is optimized for the first time on FAST samples. W with ~12 weight % Cr and 0.6 weight % Y appears to be the optimum. At 1273 K in humid air the sublimation rate is suppressed by more than one order of magnitude as compared to that of pure W – significantly mitigating the radioactive hazard.

Neutron fluxes will alter the properties of the material when employed in DEMO. Extensive studies were published on W. Here, the influence of alloying elements on the neutron transport and transmutation of tungsten is studied by simulating the exposure of spatially heterogeneous highresolution models of the W-Cr-Y alloys to 14 MeV fusion neutrons. The rhenium (Re) production and the decay heat generation in case of a LOCA are addressed. Further, the influence of Re on the oxidation resistance of the W-Cr-Y alloy is investigated experimentally; this includes studies on the potential release of radioactive oxides formed in the oxidation process.

Keywords: *tungsten alloys, DEMO, safety, sublimation, oxidation*

***Corresponding author:** *fe.klein@fz-juelich.de*

Surface Damage of Beryllium Armor Materials under Extreme Plasma Heat Loads

P1-023

Igor Kupriyanov [a]*, A. Markin [a], A. Zhidilev [a], V. Podkovyrov [b], N. Klimov [b], D. Kovalenko [b], V. Efimov [c], V. Safronov [d], A. Gervash [f]

[a] *A.A. Bochvar High Technology Research Institute of Inorganic Materials, Moscow, Russia*
[b] *TRINITI, Troitsk, Moscow, Russia*
[c] *National Research Nuclear University 'MEPHI', Moscow, Russia*
[d] *Project Center of ITER, Moscow, Russia*
[f] *Efremov Research Institute, St. Peterburg, Russia*

The first wall panels of the ITER main chamber will be completely armored with beryllium. During plasma operation in the ITER, beryllium will be suffered by high transient heat loads, such as ELMs, disruptions, VDE, etc. (off normal events). These transient loads cause rapid heating of beryllium surface and can result in significant changes in surface and near-surface regions, such as material loss, melting, cracking, evaporation and formation of beryllium dust as well as hydrogen isotopes retention both in the armor and in the dust. It is expected that the damage of beryllium under ELMs and disruptions will have significant impact on lifetime of the ITER first wall.

This paper presents the results of recent experiments on QSPA-Be plasma gun facility with two ITER beryllium grades: TGP-56FW and S-65C. The special Be/CuCrZr and Be mock-ups with beryllium armor plates of different dimensions were tested by deuterium plasma streams (6 cm in diameter) with pulse duration of 0.5 ms and heat loads of 1.5-2.5 MJ/m^2 and maximum quantities of plasma pulses up to 50 shots. The angle between plasma stream direction and mock-ups surface was 90^o. During the experiments the temperature of Be plates before each plasma pulse has been maintained at 250 and 500°C that corresponds to the temperature of the ITER Be first wall armor. After 50 shots, the beryllium mass loss/gain and erosion rate were studied as well as a surface cracking and a microstructure of beryllium plates. Influences of beryllium plate dimensions, the size of the gaps between plates, plasma heat loads and surface temperature on the Be erosion and surface damage are discussed. The study of deuterium retention in erosion products by thermal desorption spectroscopy will be also presented.

Keywords: *beryllium, fusion reactor, plasma facing material, plasma heat loads, erosion*

Corresponding author: *igkupr@gmail.com, IBKupriyanov@bochvar.ru*

Tritium Distribution Analysis on Be Limiter Tiles from JET-ITER Like Wall Campaigns using Imaging Plate Technique and β-Ray Induced X-ray Spectrometry

P1-024

Sun Eui Lee [a]*, Y. Hatano [a], M. Hara [a], S. Masuzaki [b], M. Tokitani [b], M. Oyaizu [c], H. Kurotaki [c], D. Hamaguchi [c], H. Nakamura [c], N. Asakura [c], Y. Oya [d], A. Widdowson [e], S. Jachmich [f], M. Rubel [g], J. Likonen [h] and JET contributors

[a] *University of Toyama, Toyama, Japan*
[b] *National Institute for Fusion Science, Toki, Japan*
[c] *National Institutes for Quantum and Radiological Science and Technology (QST), Rokkasho Aomori, Japan,*
[d] *Shizuoka University, Shizuoka, Japan*
[e] *Culham Centre for Fusion Energy, Culham Science Centre, Abingdon, UK*
[f] *Association Euratom-Etat Belge, ERM-KMS, Brussels Belgium*
[g] *KTH Royal Institute of Technology, Stockholm, Sweden*
[h] *VTT Technical Research Centre of Finland, VTT, Finland*

The Joint European Torus (JET) has been operated since year 2011 with the ITER-Like Wall (ILW): Be in the main chamber and tungsten in the divertor: bulk W and W-coated carbon tiles. Following three ILW campaigns in 2011-2012 (ILW-1), 2013-2014 (ILW-2) and 2015-2016 (ILW-3) a number of selected tiles exposed to plasma were retrieved for post-mortem analysis. In this study, tritium (T) distributions on plasma-facing surfaces (PFS) and in the castellation grooves of several Be tiles were examined using imaging plate (IP) technique and β-ray induced x-ray spectrometry (BIXS). Samples of the castellated tiles were cut from limiters: inner-wall guard (IWGL), outer poloidal (OPL) and dump plate (DP) retrieved after ILW-1 and then shipped to the International Fusion Energy Research Center (QST, Rokkasho, Japan) for IP and BIXS analysis.

The highest T concentration is observed at the central part of the plasma-facing surface of the OPL. The penetration depth and T retention at this position was evaluated to be ~4 μm and 50 kBq/cm^2, respectively. The T concentrations on the PFS of IWGL and DP are significantly lower than that above quoted value. Analysis of surfaces located in the grooves of castellation has shown the T deposition was extended up to ~5 mm into the grooves. The T concentrations in the first 1–2 mm from the entrances of grooves vary depending on location, though similar profiles are observed for all analyzed surfaces in the deeper regions (3–5 mm). In the presentation, the T distributions on ILW-1 tiles will be compared with those on the tiles retrieved after ILW-3.

See appendix to paper of X. Litaudon, Nucl. Fusion 57 (2017) 102001

Keywords: *JET-ILW, tritium, Be limiter, castellation*

***Corresponding author:** m1841222@ems.u-toyama.ac.jp*

Smart Tungsten-Based Alloys for a First Wall of DEMO

P1-025

Andrey Litnovsky [a]*, Janina Schmitz [a,b], Felix Klein [a], Karen De Lannoye [a,c], Arkadi Kreter [a], Marcin Rasinski [a], Jan W. Coenen [a], Christian Linsmeier [a], Jesus Gonzalez-Julian [a], Andreas Mutzke [d], Ivan Povstugar [a], Duc Nguyen-Manh [e], Mark Gilbert [e], Damjan Sobiraj [f], Jan Wróbel [f]

[a] *Forschungszentrum Jülich GmbH, Jülich, Germany*
[b] *Department of Applied Physics, Ghent University, Ghent, Belgium*
[c] *Vrije Universiteit Brussel, Elsene, Belgium*
[d] *Max-Planck-Institut für Plasmaphysik, Greifswald, Germany*
[e] *CCFE, United Kingdom Atomic Energy Authority, Culham Science Centre, Abingdon, Oxfordshire, UK*
[f] *Warsaw University of Technology, Warsaw, Poland*

During the accident with loss-of-coolant and air ingress in DEMO, the temperature of tungsten first wall cladding may exceed 1000°C and remain for months leading to tungsten oxidation. The radioactive tungsten oxide can be mobilized to the environment at rates 10 - 150 kg per hour. Smart tungsten-based alloys are under development to address this issue. Alloys are aimed to function as pure tungsten during regular plasma operation of DEMO. During an accident, alloying elements will create the protective layer, suppressing release of W oxide.

Bulk smart alloys were developed by using mechanical alloying and field-assisted sintering technology. Smart alloys and tungsten were tested under variety of DEMO-relevant plasma conditions. Both materials demonstrated similar sputtering resistance to deuterium plasma. Seeded plasmas impose issues for both materials, as predicted using SDTrimSP code. Under accidental conditions alloys feature a 40-fold reduction of W release compared to that of tungsten. Atom probe tomography revealed remarkable stabilization of protecting layer in agreement with modeling. The research and development program on smart first wall for DEMO will be presented.

Keywords: *DEMO, first wall, advanced tungsten alloys, safety, reduced oxidation*

***Corresponding author:** a.litnovsky@fz-juelich.de*

Substantial Upgrades of the TCV Tokamak for Improved Divertor and Heating Capabilities

P1-026

Yves Martin *, Stefano Alberti, Hammam Elaian, Ambrogio Fasoli, Alexander Karpushov, Florian Ramseyer, Holger Reimerdes, Miguel Silva, Matthieu Toussaint

Ecole Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), Switzerland

The TCV tokamak and its 16 independently driven poloidal shaping coils has allowed intensive developments of plasma shapes, in particular with the aim of reducing the heat flux onto the plasma facing components. In order to approach regimes of interest for future fusion power plants, and to explore ways to further reduce the heat load onto these components, a stronger separation between the radiative divertor region and the plasma itself could be advantageous, since it would allow to increase the pressure in the divertor, to radiate a larger power fraction, without spoiling the neutral pressure around the plasma. Therefore, a set of protruding tiles forming a kind of diaphragm or baffle is being installed into TCV to implement closure between the plasma and divertor region. The baffle is made of 32 and 64 modified tiles installed on the High and Low field side, respectively, of the TCV vessel. Several pieces are equipped with magnetic or Langmuir probes to diagnose the plasma at the vicinity of the baffle tips. A series of diagnostics had to be implemented or modified to address the physics issues linked with the installation of the baffles.

In addition, the ECH and NBI heating systems are upgraded to get closer to fusion reactors conditions: new dual frequency, 1MW gyrotrons are installed, a new routing of microwaves is implemented as well as new launchers on TCV ports. A second, 1MW NBI will improve the capability of heating the plasma ions without generating total particle momentum since both beams will aim at opposite toroidal directions.

Keywords: *TCV tokamak, divertor, heating, ECH, NBI*

***Corresponding author:** yves.martin@epfl.ch*

Preliminary Engineering Assessment of Alternative Magnetic Divertor Configurations for EU-DEMO

P1-027

Domenico Marzullo [a*], Roberto Ambrosino [b], Antonio Castaldo [b], Giuseppe Di Gironimo [a], Samuel Merriman [c]

[a] *Consorzio CREATE, Univ. of Naples Federico II, Napoli, Italy*
[b] *Consorzio CREATE, Univ. Napoli Federico II, Napoli, Italy*
[c] *CCFE, Culham Science Centre, Abingdon, UK*

One of the main challenges in the roadmap to the realization of fusion energy is to develop a heat and power exhaust system able to withstand the large loads expected in the divertor of a fusion power plant.

The challenge of reducing the heat load on the divertor targets is addressed, within the mission 2 'Heat-exhaust systems', through the investigation of divertor configurations alternative to the standard Single Null (SN), such as the Snowflake (SF), Double Null (DN), X and Super-X (SX) divertors.

This paper focuses on a preliminary engineering assessment of the alternative configurations proposed for the EU DEMO reactor. Starting from the description of the optimized plasma shape developed for each configuration, the 3D geometrical description of the Magnet System and of the main Mechanical Structures (Vacuum Vessel and in-vessel components) is presented. Based on the 3D geometry, the compatibility of the location and dimensions of ports with Remote Maintenance needs is discussed. The possibility to define non-standard Toroidal Field (TF) coils is investigated performing linear elastic static structural stress analysis against Start of Flat Top and End of Flat Top Electro-Magnetic (EM) loads. Possible design optimizations are proposed both for the Magnets system and the mechanical structures design. Finally, the various configurations are compared with regard to the engineering and feasibility aspects.

Keywords: *EU-DEMO, Alternative divertor configurations, FEM Structural Analysis, CAD design*

Corresponding author: *domenico.marzullo@unina.it*

In-Situ Measurement of Surface Modifications of Tungsten Exposed to Pulsed High Heat Flux for Divertor Design in Tokamak-Type Fusion Nuclear Reactors

P1-028

Yuki Matsuda *, Shohei Yamashita, Yusei Miyamoto, Daichi Motoi, Takufumi Okita, Eiji Hoashi, Kenzo Ibano, Yoshio Ueda

Graduate School of Engineering, Osaka University, Osaka, Japan

Tungsten is a candidate as a surface material of plasma facing components in tokamak-type fusion nuclear reactors. In the reactors, tungsten is exposed to pulsed high heat flux caused by plasma disruption and Type I Edge Localized Mode(ELM). It is known that such transient thermal loads causes melting and solidification of tungsten repeatedly. The repetitive phase change can degrade surface characteristics of divertor and reduce the thermal resistance. It is thus significant to investigate and understand surface melting and solidification procedure in detail for the divertor design. In previous studies, it was confirmed that the surface after the laser irradiation had a crater with a center cone. The in-situ observation of tungsten during laser irradiation using a high speed video (HSV) camera was tested but the surface modification of tungsten could not be fully observed due to a halation by the strong luminescence during melting and vaporization. Then, in this study, by modifying the optical and the lens systems, the in-situ observation of the surface modification dynamics of tungsten was succeeded. It could be observed that luminescence at the center of the irradiated area fluctuated violently in association with the phase change of molten tungsten. In addition, 3-Dimensional (3D) behavior of molten tungsten during high heat load was evaluated dynamically using stereo photography technique (SPT). 3D images of molten tungsten could be taken successfully and the results indicated that the surface fluctuated and gathered to the center gradually after the laser irradiation.

Keywords: *Tungsten divertor, Pulsed high heat flux, Surface modification, Stereoscopic photography*

Corresponding author: *y-matsuda@stu.nucl.eng.osaka-u.ac.jp*

Eurofusion-DEMO Divertor - Cassette Design and Integration

P1-029

Giuseppe Mazzone [a*], Jeong-Ha You [b], Christian Bachmann [c], Ugo Bonavolontà [d], Valerio Cerri [a], Domenico Coccorese [d], Danilo Dongiovanni [a], Davide Flammini [a], Paolo Frosi [a], Laurent Forest [g], Giuseppe Di Gironimo [d], Gennaro Di Mambro [e], Vito Imbriani [d], Antonio Maffucci [e], Domenico Marzullo [d], Pietro Alessandro Di Maio [f], Maria Teresa Porfiri [a], Eugenio Vallone [f], Rosaria Villari [a], Eliseo Visca [a], Christian Vorpahl [c]

[a] *Department of Fusion and Technology for Nuclear Safety and Security, ENEA, Frascati, Italy*
[b] *Max Planck Institute of Plasma Physics, Garching, Germany*
[c] *EUROfusion, PMU-PPPT, Garching, Germany*
[d] *CREATE/University of Naples, Industrial Engineering, Napoli, Italy*
[e] *CREATE – University of Cassimo, Cassino, Italy*
[f] *Department of Engineering, University of Palermo, Viale delle Scienze, Palermo, Italy*
[g] *DEN-Service d'études mécaniques et thermiques (SEMT), CEA, Université Paris-Saclay, Gif sur Yvette, France*

The Eurofusion-DEMO design will complete the Pre Conceptual Design phase (PCD) with a PCD Gate, named G1, scheduled to take place in Q4 2020 that will focus on assessing the feasibility of the plant and its main components prior to entering into the Conceptual Design phase. In the paper first an overview is given of the Eurofusion-DEMO Divertor Assembly including design and interface description, systems & functional requirements, load specification, system classification, manufacturing procedures and cost estimate. Then critical issues are discussed and potential design solutions are proposed, e.g.:

- Neutron material damage limits of the different (structural) materials present in the divertor assembly (as CuCrZr, Eurofer) and in the vacuum vessel (AISI 316L(N)-IG);
- Temperature hot spots in parts of the divertor assembly exposed to high nuclear heating and/or high heat radiation (from the plasma core or the separatrix) causing difficulties for active or passive cooling (e.g. cassette body structure, liner support structures, mechanical supports, divertor toroidal rails);
- Arrangement and design of plasma-facing components and liner with pumping slot in the divertor cassette to enable pumping of exhaust gases from the lower port.

Keywords: *Eurofusion-DEMO, Divertor Assembly, Divertor Cassette, Divertor Toroidal Rails, Eurofer*

Corresponding author: *giuseppe.mazzone@enea.it*

Neutron Tomography of "Thermal Break" Divertor Mock-Ups Before and After High Heat Flux Exposure

P1-030

Frank Schoofs [a*], Triestino Minniti [a], Winfried Kockelmann [b], Llion M. Evans [a,c], Mike Fursdon [a], Thomas R. Barrett [a], Heather Lewtas [a]

[a] *United Kingdom Atomic Energy Authority, Culham Science Centre, Abingdon, UK*
[b] *ISIS - Pulsed Neutron and Muon Source, Rutherford Appleton Laboratory, Harwell Campus, Didcot, UK*
[c] *College of Engineering, Swansea University, Bay Campus, Swansea, UK*

The function of divertor components is to extract heat from the fusion plasma and to do so reliably over several cycles. The lifetime of these components is limited by the thermally induced fatigue, which is expected to occur at dissimilar material boundaries. Neutron tomography data is collected on a "thermal break" EU-DEMO divertor concept small-scale mock-up (MUP), in the as-manufactured state and after high heat flux exposure at the GLADIS facility. The MUPs were exposed to 1 cycle at 25 MW/m^2 and 100 cycles at 10 MW/m^2. Neutron tomography has been used as a non-destructive method to inspect the quality of the entire component with the aim of detecting microscopic manufacturing defects like cracks, interface flaws and delamination. Furthermore, a selected region of interest within the specimen near the W/Cu interface has been investigated by energy-resolved neutron radiography to study the distribution of residual strains with high spatial resolution. Such experimental outcomes have been compared against finite element analysis (FEA) simulations and used to validate the model.

Keywords: *divertor, neutron imaging, energy-resolved neutron imaging, neutron radiography, neutron tomography*

Corresponding author: *frank.schoofs@ukaea.uk*

Design of Electron Cyclotron Resonance Heating Protection Components for First Plasma Operations in ITER

P1-031

Alessandro Moro [a*], Alessandro Bruschi [a], Olivier Darcourt [b], Francesco Fanale [a], Daniela Farina [a], Lorenzo Figini [a], Franco Gandini [b], Mark Henderson [b], Ryan Hunt [b], Carsten Lechte [c], Paola Platania [a], Burkhard Plaum [c]

[a] *Istituto di Fisica del Plasma IFP-CNR, Milano, Italy*
[b] *ITER Organization, Saint-Paul-lez-Durance, France*
[c] *Universität Stuttgart, IGVP, Stuttgart, Germany*

The ITER first plasma operations will occur with no full blanket and divertor components installed. Machine protection is required and First Plasma Protection Components (FPPC) have been designed to shelter the vacuum vessel and other components from the plasma itself and from high power Electron Cyclotron Resonance Heating (ECRH), foreseen for first plasma breakdown. Dedicated ECRH protection components will be installed to protect in-vessel structures from direct and stray radiation as EC beams will be used to assist the plasma breakdown. This paper is focused on the design of these specific components for ITER first plasma operations (B=2.65 T, I_p<1 MA), where a bundle of EC beams will be injected from the ECRH upper launcher (P~0.8 MW per beam, Δt=0.3-1s pulse length) towards the central column of the vacuum vessel. Dedicated mirrors will shape and redirect the beams to the EC resonance location in the poloidal magnetic field null region and then into a beam dump located in an equatorial port, where not absorbed EC radiation will be trapped and dumped (targeting absorption>90%). As the entire system will be used only for first plasma operational regime, its design aims at cost effectiveness, easy assembly and fulfillment of plasma operations. Two mirrors and one grating mirror have been designed to provide the required shaping and directions for the beams coming from the upper launcher towards EC resonance and dump. The ECRH beam dump consists of four large plates affixed to the equatorial port mounts. The plates are differently shaped, oriented and covered by a distributed absorbing coating according to trapping requirements. We describe here the guidelines that drove the design of the quasi-optical system for ECRH operations, the characteristics of the mirrors, resulting launched beams and concept developed for the beam dump. Evaluation of the expected performances of the system during first plasma operations will be also given.

Keywords: *Electron Cyclotron Resonance Heating, quasi-optics, gratings, coatings*

***Corresponding author:** *moro@ifp.cnr.it*

Behaviour of Actively Cooled ITER Divertor Mock-Ups in High Power ASDEX Upgrade Discharges

P1-032

Rudolf Neu [a,b*], Henri Greuner [a], Albrecht Herrmann [a], Martin Balden [a], Bernd Böswirth [a], Dominik Brida [a], Ralph Dux [a], Stefan Elgeti [a], Michael Faitsch [a], Karl Krieger [a], Peter Leitenstern [a], V. Rohde [a], ASDEX Upgrade Team

[a] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[b] *Technische Universität München, Garching, Germany*

In order to support the development of a qualification strategy for ITER divertor targets, a selection of mock-ups from several manufacturers have been subjected to high heat-flux tests in the GLADIS facility. Actively cooled W monoblock mock-ups by a European supplier have been successfully tested by applying up to 100 pulses at 10 MW/m² and another 300 pulses at 20 MW/m² (10 s each). During the latter pulses surface temperatures of about 2300 °C were reached and as a consequence strong recrystallization and dimensional changes of the monoblocks have been detected. In order to investigate their further behaviour under ITER-relevant plasma and cooling conditions, two mockups have been exposed in the divertor of ASDEX Upgrade (AUG) for a series of discharges with a maximum heating power up to 15 MW. For this purpose they have been mounted on the AUG divertor manipulator, using an active cooling loop allowing to mimic the ITER conditions as close as possible (inlet water temperature of 70 °C, flow of 1 l/s). The mock-ups are equipped with thermocouples and the divertor manipulator is monitored by thermography and spectroscopy. Before the component was mounted in AUG, the status of both mock-ups has been documented and markers have been placed by the high resolution, heavy duty SEM/FIB available at IPP. By comparing these surfaces after exposure in AUG with their status before, information on the initiation and evolution of damages under ITER relevant plasma condition has been extracted. During the AUG exposure the maximum surface temperature on the loaded monoblocks reached values above 2000 °C. Despite the high loads no further degradation of the mock-ups was observed.

Keywords: *tungsten plasma facing components, active cooling, erosion*

***Corresponding author:** *Rudolf.Neu@ipp.mpg.de*

Manufacturing of a HCPB First Wall Demonstrator Using the Additive Manufacturing Process of Metal Powder Application

P1-033

Heiko Neuberger *, Francisco Hernandez

Karlsruhe Institute of Technology (KIT), Institute of Neutron Physics and Reactor Technology (INR)

In 2017 and 2018 an analysis was performed for identification of the main cost driving factors on the roadmap for nuclear fusion as option for commercial electricity production. The study also included the envisaged manufacturing technologies for the First Wall. In parallel to the well-established but costly reference route also options to apply Additive Manufacturing were investigated. An innovative concept basing on Metal Powder Application in combination with HIP welding was identified as option promising a cost effective solution also providing a good dimensional precision, suitability to full-scale Breeding Blanket dimensions and application of channel internal Heat Transfer enhancement structures. Building of dedicated demonstrators was started in 2018, this paper reports the results of qualification of these components.

Keywords: *First Wall, Breeding Blanket, Test Blanket Module, Manufacturing, Additive Manufacturing*

****Corresponding author:*** *heiko.neuberger@kit.edu*

Nuclear Analyses for the Design of the Plasma Facing Components of the DEMO Divertor

P1-034

Simone Noce [a]*, Giacomo Dose [a], Davide Flammini [b], Vito Imbriani [c], Giuseppe Mazzone [b], Selanna Roccella [b], Francesco Romanelli [a], Rosaria Villari [b], Eliseo Visca [b], Jeong-Ha You [d]

[a] *University of Rome Tor Vergata, Industrial Engineering Department, Rome, Italy*
[b] *ENEA FSN-FUSTEC-TEN, C.R. Frascati, Frascati (Rome), Italy*
[c] *CREATE/University of Naples, Industrial Engineering, Napoli, Italy*
[d] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*

The Divertor Plasma Facing Components (PFCs) of a fusion reactor are the most loaded components in terms of high heat fluxes; which combined with high neutron irradiation, can severely compromise their thermal mechanical properties and their heat removal capacity. Therefore, neutronic assessment plays a key role in the design of these critical components. The aim of this work is to perform a detailed nuclear analysis for the European DEMO divertor PFCs, aimed to provide significant outcomes in the PFCs selection concept. In particular, the present assessment is devoted to the reference ITER-like configuration and to a low-activation Chromium-based alternative concept, under study within the EUROfusion WPDIV-PPPT programme. Three-dimensional neutronic and activation analyses have been performed with the MCNP5 Monte Carlo and FISPACT II inventory codes. This work presents high quality neutronics results using much refined mesh, heterogeneous materials constitution and actual geometry of the two PFC concepts. Furthermore, detailed data on the nuclear damage of the PFCs including helium production, assessed for the first time for the latest DEMO design are presented and the impact of the neutronics and activation issues on both PFCs concept design, lifetime, operations and safety are discussed.

Keywords: *PFC, DEMO, neutronics, Divertor, MCNP*

****Corresponding author:*** *Simone.Noce@uniroma2.it*

Plasma Irradiation Experiment on the Metal Pebble Flow in the TPD Sheet IV

P1-035

Takeru Ohgo [a]*, Takuya Goto [a,b], Junichi Miyazawa [a,b], Toshikio Takimoto [c], Akira Tonegawa[c]

[a] *The Graduate University for Advanced Studies, Toki, Gifu, Japan*
[b] *National Institute Fusion Science, 322-6 Oroshi, Toki, Gifu, Japan*
[c] *Tokai University, 4-1-1Kitakaname, Hiratsuka, Kanagawa, Japan*

A renewable divertor concept named the REVOLVER-D (Reactor oriented Effectively VOLumetric VERtical Divertor) has been proposed for the helical reactor FFHR. The REVOLVER-D consists of molten tin shower jets inserted to the ergodic region of the plasma as a limiter/divertor and expected to tolerate a high heat load larger than a few tens of MW per square meter. However, the jets can be deformed due to the strong Lorentz force. To solve this problem, we adopt a tin pebble flow instead of the molten tin flow to cut the electric current path. In addition, the permissible heat load of the divertor target can be increased because of the heat of fusion and the decrease in the initial temperature. A part of tin pebbles become melted by the heat load from the plasma. Therefore, we decided to melt all tin pebbles in the pool that catches the pebbles. Then it also becomes possible to absorb the drop impact of the pebbles. Next to the pool, tin pebbles are manufactured from the molten tin by the shot tower method using a viscous silicone oil. Then pebbles are transported mechanically and dropped to the plasma again. This concept is named the Fusible Metal Pebble Divertor (FMPD). The pebble behavior in the plasma is one of the issues on the FMPD. In the previous study on the ceramic pebble divertor, forces act on the pebble flow (the coulomb force, the Lorentz force, the plasma pressure, and the plasma momentum flux.) have been evaluated. In this presentation, the detailed scenario of the FMPD and experimental results on the pebble behavior in the TPD will be given.

Keywords: *divertor, fusible metal, pebble divertor, plasma experiment*

**Corresponding author: ohgo.takeru@LHD.nifs.ac.jp*

Study on Feasibility of Steady-State Assumption for HCCR-TBM under Transient Thermal Load

P1-036

Seong Dae Park [a], Suk-Kwon Kim [a], **Eo Hwak Lee** [a], Dong Won Lee [a], Mu-Young Ahn [b]

[a] *Korea Atomic Energy Research Institute, Daejeon, Korea*
[b] *National Fusion Research Institute, Daejeon, Korea*

When designing Helium-Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM), structural analysis have been performed considering steady-state conditions to check whether structural requirements are met under various load combinations including thermal load. However, in ITER, heat load is implemented in transient way due to pulse operations which may cause differences in the maximum stress compared to steady-state assumption.

In the present steady, this difference is investigated with preliminary design phase II model of HCCR-TBM to assess the feasibility of the steady-state analyses under transient thermal load. The thermal stress was analyzed under two conditions, steady-state and transient up to 5 thermal cycles in inductive operation I of ITER. First, thermo-hydraulic analysis was carried out for the two conditions. The temperatures are saturated within 3 cycles for the transient condition while the maximum temperatures of the structure, breeder and multiplier are lower than those for the steady condition. Then, structural analysis was performed. In particular for the transient calculation, temperature distributions in 4 time-instants when steep temperature gradients were observed were selected to compare the thermal stress with the steady-state. The results show that the stresses under transient condition are lower than those of the steady-state condition. Conclusively, it is confirmed that the steady-state analysis can show the conservative results compared to the actual transient.

Keywords: *ITER, TBM, thermal-hydraulic, HCCR-TBM*

**Corresponding author: sdpark@kaeri.re.kr*

Survey of the Behaviour of Liquid Metals Targets, Sn and Li, in Reactor Relevant Conditions. DEMO and I-DTT

P1-037

Vincenzo Pericoli Ridolfini [a], Piotr Chmielewski [a], Irena Ivanova-Stanik [a], Michał Poradziński [a], Roman Zagórski [a], Flavio Crisanti [b]

[a] *Institute of Plasma Physics and Laser Microfusion, Warsaw, Poland*
[b] *Unitá Tecnica Fusione, C.R. ENEA Frascati, Frascati, Italy*

The performance of liquid metal divertor targets, of either lithium or tin, in highly powered tokamaks is analyzed for the proposed European DEMO and the projected Italian divertor test tokamak (IDTT). Two codes are used to this purpose, one is TECXY, which is a 2D edge code and the other is COREDIV, which couples a simplified 2D treatment of the edge with a 1D core description. TECXY has the merit to allow a rather fast exploration of the possible operating parameter space of the considered device, at the price of a simplified treatment of the neutral dynamics. COREDIV allows instead judging about the overall mutual core-edge effects for the chosen scenario. Liquid targets temperature, which is a crucial parameter in all these exploration is calculated self consistently. The results show that both materials are compatible with either DEMO or I-DTT under proper setting of the liquid divertor parameters. As a general results operation at high density are always highly recommended especially if detached conditions are desired. Nevertheless sustainable heat loads (<10 MW/m^2) can be obtained also at lower density. Tin performs usually better than lithium in terms of the target load mitigating properties and of the material consumption, which in a DEMO-like device could be very high. Core plasma contamination by tin can be effectively controlled by enhancing radiation in the SOL via impurity seeding, Ar or even Ne for I-DTT. Conversely this technique scarcely affects the Li release rate since apparently the extra power losses replace simply an almost equal amount of those due to Li. More details are provided in the paper

Keywords: *DEMO, DTT, SOL plasma, liquid divertor, heat load mitigation*

**Corresponding author: vincenzo.pericoli@ifpilm.pl*

TOKES Simulations of Mitigated Disruption Thermal Quenches in ITER

P1-038

Sergey Pestchanyi [a], M. Lehnen [b], R.A. Pitts [b], G. Saibene [c]

[a] *KIT, Eggenstein-Leopoldshafen, Germany*

[b] *ITER Organization, St. Paul Lez Durance Cedex, France*

[c] *Fusion for Energy, Barcelona, Spain*

A high level of avoidance and mitigation of disruptions will be mandatory on ITER as a result of the significant potential consequences for the lifetime of certain components due to disruption induced forces and plasma thermal loads. A disruption mitigation system (DMS) is being designed to ensure that thermal, electromagnetic and runaway electron (RE) loads are reduced to tolerable levels. The mitigation strategy relies on the injection of impurities/fuel using a large number of shattered pellet injectors (SPI) situated in three upper and three equatorial port plugs.

The TOKES code has been used for several years to simulate mitigation by massive impurity injection [1]. Estimating the heat loads from the intense line radiation emitted during the mitigated thermal quench (TQ) is the principal focus of this effort. Previous simulations of TQs mitigated by SPI with TOKES were performed for pellets consisting of pure Ne of various size [1].

This paper presents the most recent TOKES simulations of SPI in ITER, similar to [1], but for realistic pellets consisting of deuterium/neon mixtures with the ratio $Ne/(D_2+Ne) = 0 \rightarrow 0.4$. Cylindrical pellets of two different sizes: D=19.7 mm, L/D = 1.5 with $2.6 \cdot 10^{23}$ D_2 molecules and D=28.5 mm, L/D = 2.0 with $1.1 \cdot 10^{24}$ D_2 molecules are considered (diameter D, length L). The results are compared with simulations for injections of the same amount of Ne without D admixture. All pellets containing D/Ne mixtures are found to yield radiation of more than 90% of the core thermal energy. The increase in the radiation efficiency is associated with a large increase in the electron density in the radiating cloud of Ne due to the ionized deuterium.

[1] *https://doi.org/10.1016/j.fusengdes.2017.12.016*

Keywords: *ITER, disruption mitigation, shattered pellet injection, numeric simulation, TOKES*

Corresponding author: *serguei.pestchanyi@kit.edu*

Progress in Commissioning of the HELCZA High Heat Flux Test Facility for ITER First Wall Panel Tests

P1-039

Jan Prokůpek *, Richard Jílek, Josef Sklenka, Karel Samec, Mark Murtazin

Centrum výzkumu Řež, Husinec-Řež, Czech Republic

HELCZA is an experimental complex designed for cyclic heat loading for testing of plasma facing components (PFC) like ITER and DEMO first wall and divertor components and mock-ups. This high heat flux (HHF) test facility allows testing of PFCs in the multi-MW/m^2 range using an 800 kW electron beam gun featuring electromagnetic system providing a beam scanning frequency of 20 kHz at the primary deflection angle up to ±40 degree. The cooling system allows variation of the inlet cooling water temperature in a wide range of temperatures from 25°C up to 320°C and water pressure up to 15 MPa. The HELCZA diagnostics for surface monitoring and sample measurements consist of infrared cameras, high-resolution cameras, X-ray camera, one and two colours pyrometers, thermocouples, temperature sensors, flow meters, and manometers.

The paper presents the recent progress in commissioning of the HHF test facility HELCZA to demonstrate the facility's readiness for testing of PFCs, in particular the first wall panels at high heat loads. For the qualification tests the flat cooper first wall mock-up with representative size for electron beam irradiated windows of all types of first wall panels was used. Moreover, successful HHF tests of ITER-like divertor mock-ups have been carried out at HELCZA in the framework of the EUROfusion Divertor project.

Keywords: *HHF tests, high heat flux, plasma facing components, HELCZA, first wall*

Corresponding author: *jan.prokupek@cvrez.cz*

DEMO Single Module Segment Concept First Wall and Limiter Misalignment Study by 3D Field Line Tracing

P1-040

Maria Lorena Richiusa [a]*, Wayne Arter [a], Dominic Calleja [b], Medhi Firdaouss [c], Jonathan Gerardin [c], Michael Kovari [a], Francesco Maviglia [d], Zsolt Vizvary [a]

[a] *UKAEA-CCFE, Culham Science Centre, Abingdon, Oxon, UK*
[b] *University of Liverpool, Liverpool, UK*
[c] *IRFM CEA Cadarache, Saint Paul lez Durance, France*
[d] *EUROfusion – Programme Management Unit, Garching, Germany*

Within the framework of EURO*fusion* DEMO First Wall and limiter design activities, the protection of the First Wall against power deposition peaks is being considered. During steady-state operation, the radiative power from the plasma could be considered uniformly spread on plasma-facing components. However, the presence of openings (i.e. gaps between segments and ports) and the introduction of limiters breaks the continuity of the wall and opens the possibility of localized high heat flux values on toroidally facing gaps due to charged particles striking the wall. These fluxes can be amplified by misalignments between components upon manufacturing, assembly or under operational conditions.

In this paper, the 3D field line tracing codes SMARDDA and PFCflux are used for studying the impact of misalignment on the heat load distribution for a periodically segmented DEMO First Wall, specifically the Single Module Segment Concept. The work covers normal operation (ramp-up/down and steady-state), H-L transition and off-normal transients such as Upper Vertical Displacement Events (UVDEs), considering both the cases of bare First Wall (without limiters, as reference) and First Wall protected by limiters. Heat flux penalty factor maps have been created to identify the worst cases among the ones analyzed. The related heat flux maps are relevant for thermal assessments of a simplified DEMO model to be performed later on, particularly in terms of maximum temperature values for the current DEMO design.

Keywords: *Misalignment Study, SMARDDA/PFCflux, 3D Field Line Tracing, Single Module Segment Concept.*

Corresponding author: *lorena.richiusa@ukaea.uk*

Ultrasonic Test Results Comparison Before and After High Heat Flux Testing on W-Monoblock Mock-Ups Vertical Target for EU-DEMO

P1-041

Selana Roccella [a]*, G. Dose [b], L.Dupont [c], T. Barrett [d], F. Gallay [c], H. Greuner [e], M. Richou [c], E.Visca [a], J-H. You [e]

[a] *Department of Fusion and Technology for Nuclear Safety and Security, ENEA, Frascati, Italy*
[b] *Dipartimento di Ingegneria Industriale, Università di Roma "Tor Vergata", Rome, Italy*
[c] *CEA, IRFM, Saint-Paul-Lez-Durance, France*
[d] *CCFE, Culham Science Centre, Abingdon, Oxon, UK*
[e] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*

In the framework of the activity of WP-DIV target development phase 2, ENEA carried out an extensive ultrasonic testing (UT) campaign on more than 66 tungsten monoblock mock-ups. The EU-DEMO target development activity concerns primarily the comparison between the reference solution for the divertor targets (ITER-like) and two other concepts, in which the interlayer between monoblocks and tube has been modified with the aim of increasing the component performance (Thermal Break Interlayer, Thin Graded Interlayer). In the same activity, many other aspects were also considered: mock-ups were manufactured to analyze the influence of the tungsten monoblock supplier, interlayer thickness and interlayer realization process (casting or diffusion bonding) on the component performance. During phase 1 of the target development activity, the ENEA ultrasonic technique proved, by comparison with metallographic results, its reliability to detect and size defects for all the design concepts. In this work the comparison between UT results obtained before and after the high heat flux test on the more significant mock-ups for phase 2 are carried out. In particular, the comparison of the different design options is here discussed in terms of UT results, while the other aspects will be addressed in future publications.

Corresponding author: *e-mail: selanna.roccella@enea.it*

Research on the Normal Spectral Band Emissivity Characteristic within 7.5 to 13 µm for Graphite Between 100 and 500 °C

P1-042

Yuzhong Zhang, Wuqiong Zhang, Shuangbao Shu, Zhe Dong, Yan Zhang

School of Instrument Science and Opto-electronics Engineering, Hefei University of Technology, Hefei, China

Graphite is an important alternative material to construct the divertor for Tokamak (EAST) on the first wall. Due to the effect of plasma desorption, sputtering and so on, the first wall, especially the divertor area, will bear high energy to act on the area, and the temperature will be higher and higher. Therefore, the detection and diagnosis of the first wall temperature of Tokamak by non-contact temperature measurement is a very important for the Tokamak project. However emissivity is not a constant and it plays an important parameter that affects the precision of non-contact temperature measurement. The surface temperature of the object, form, material and other factors can lead to the emissivity of the change, which causes big error of temperature measurement. In order to solve the above problems, this paper built a set of emissivity measurement system and a new method for accurate calculation of emissivity is proposed, which effectively eliminates the interference of background radiation to experimental results, and improves the accuracy of emissivity measurement. The system measured normal spectral band emissivity of graphite under different roughness. Results indicate that the emissivity will rise with the rising of temperature, and the growth rate will gradually slow down with the increase of temperature when the temperature exceeds 200°C. Surface roughness has a great influence on emissivity of graphite, the higher the roughness, the higher the emissivity. Moreover, the curves of the result can be fitted using a nonlinear model, and the uncertainty of the measurement under different roughness was calculated and based on the precision of the results of experiment were discussed.

Corresponding author: *zhangyuzhong@hfut.edu.cn*

WITHDRAWN

P1-043

Analysis of Enhanced Heat Flux First Wall Behavior under ITER Pulsed Loads

P1-044

Maxim Sviridenko [a]*, A. Leshukov [a], S. Tomilov [a], Yu. Strebkov [a], M. Khokhlov [b], E. Okuneva [b], A. Gervash [b], V. Safronov [c], R. Raffray [d], R. Eaton [d], S. Gicquel [d], B. Calcagno [d]

[a] *Joint-Stock Company "N.A. Dollezhall Research and Development Institute of Power Engineering", (JSC "NIKIET"), Moscow, Russia*

[b] *Joint-Stock Company "NIIEFA"(D.V. Efremov Institute), St. Petersburg, Russian Federation*

[c] *Institution "Project Center ITER", Moscow, Russian Federation*

[d] *ITER Organization, Saint Paul lez Durance, France*

The First Wall (FW) is a component of the Blanket System facing the plasma. Two types of FW panel design have been developed based on the design heat flux: the Enhanced Heat Flux First Wall (EHF FW) panels designed for a 4.7MW/m^2 heat flux, and the Normal Heat Flux First Wall panels designed for a 2.0 MW/m^2 heat flux. The EHF FW panels procured by the RFDA as part of the Procurement Arrangement 1.6.P1A.RF.01 are manufactured by two institutes: JCS NIIEFA and JSC NIKIET. As part of the technological process qualification, the design of the EHF FW full scale prototype was developed and approved. The so-called "pipe option" was taken as a basis for the FW configuration because during the final assembly the fingers are connected to the beam by orbital welding, allowing for acceptably small welding distortions and while providing for the required NDT examinations. At the same time, the previously considered welded attachment of the fingers to the beam was replaced by a mechanical option. This approach allowed for the removal of the Central Slot Insert (CSI) from the FW beam central zone, thereby simplifying the cooling configuration.

This design was integrated in representative EHF FW panels located on the top and outboard zones of the vacuum vessel and the cooling configuration in the FW beams was developed to accommodate the additional plasma radiation on the front FW beam surface in the absence of the CSI.

This paper summarizes the results of design activity for top and outboard EHF FW panels and presents the results of the thermo-hydraulic and transient thermal analyses which confirm the acceptable performance of the EHF FW panels without the CSI.

Keywords: *Enhanced First Wall, Blanket system, FW beam, Central Slot Insert*

Corresponding author: *sviridenko@nikiet.ru*

Deuterium Retention in the Tungsten Irradiated Detached Plasma

P1-045

Toshikio Takimoto [a]*, Akira Tonegawa [a], Yoshihito Matsumura [a], Kohnosuke Sato [b], Kazutaka Kawamura [a]

[a] *Tokai University, Kanagawa, Japan*

[b] *Tokyo University of Science, Tokyo, Japan*

Quantitative evaluation and control of the residual amount of fuel particles (deuterium and tritium) into the divertor wall is important at the viewpoints of safety and economy. Thus, fundamental studies have been performed on the fuel particles retention characteristics of tungsten by an ion beam irradiation and a plasma irradiation. However, despite the importance of clarifying the fuel particle retention characteristics of tungsten irradiated with the impurity mixed plasma and the detached plasma in a supposition of the actual divertor environment, such experiments have not been performed enough. Therefore, we have been performing the experiments of such plasma irradiation. In a previous study, we had conducted the experiment of the deuterium-helium mixed plasma irradiation and revealed that helium enhances diffusion of deuterium in tungsten [1]. In this contribution, we report results of the deuterium detached plasma irradiation to the tungsten performed by using the linear diverter simulator TPDsheet-U. In the experiment, attached plasma (gas pressure P ~ 0.1 Pa, electron density n_e ~ 5x10^{17} m^{-3}, electron temperature T_e ~ 6 eV, ion flux Γ_i ~ 5x10^{22} m^{-2}s^{-1}) and detached plasma (P ~ 0.6 Pa, n_e ~ 8x10^{16} m^{-3}, T_e ~ 0.8 eV, Γ_i ~ 1x10^{22} m^{-2}s^{-1}) were irradiated to a 1 mm thick tungsten sample of ITER grade. After that, the amount of deuterium retention was evaluated by Thermal Deposition Spectroscopy (TDS). As a result, in the case where the irradiation time was 30 min., even though the incident ion fluence of the detached plasma decreased ~1/5 and the incident energy was reduced ~1/50 from the attached plasma, the deuterium retention amount of the sample was reduced only about half of that of the attached plasma. This suggests that the retention of the fuel particles in tungsten increases under the detached plasma condition.

[1] *T. Hayashi, T. Takimoto, A. Tonegawa, et al., Fusion Engineering and Design, 136 (2018) 545-548.*

Keywords: *detached plasma, deuterium retention, linear plasma device, tungsten*

***Corresponding author:** *2bsp1208@hope.tokai-u.jp*

Inverse Radiation Problem with Infrared Images to Monitor Plasma-Facing Components Temperature in Metallic Fusion Devices

P1-046

Charly Talatizi [a]*, Marie-Helene Aumeunier [a], Fabrice Rigollet [b], Mickael Le Bohec [b], Christophe Le Niliot [b]

[a] *cEA Cadarache/Institut de Recherche sur la Fusion par confinement Magnetique, Saint-Paul-lez-Durance cedex, France*

[b] *Aix Marseille Université, CNRS, IUSTL Marseille, France*

Temperatures infrared measurements are degraded by reflections and uncertainties on materials emissivity in full-metallic fusion devices. A photonic modelling based on a Monte Carlo ray-tracing code has been developed to predict the temperature measurement for a given plasma scenario taking into account these parasitic phenomena [1]. This model showed that the errors due to unknown emissivities and reflections could be greater than 50% [2]. A solution to improve the temperature infrared measurement is to develop an iterative inversion method to retrieve the true surface temperature from experimental images by correcting the effects of reflections and unknown materials emissivity. This method uses a direct model to generate a modelled image and an optimization algorithm that minimizes the differences between the experimental and modelled images. The ray tracing code is too demanding in terms of resources to be used in this method and a faster direct model based on the radiosity method has been chosen. This method has been applied to a numerical simplified model of a tokamak. This paper focuses on the issues raised by the application of the radiosity method to this model and their management. It also presents the results of the iterative inversion method to retrieve the true surface temperature via this method by using the ray tracing Monte Carlo code as input data replacing the experimental ones. The limits of the use of the radiosity method regarding the reflectivity models of the materials (diffuse or specular) and the acceptability of the errors made are also discussed.

[1] *Aumeunier M-H et al. 2012, IEEE Transactions On Plasma Science, 40, 3.*

[2] *Talatizi C et al., Thermique et sciences de l 'information, Tome 1, Actes du Congres annuel de la Societe Franc;aise de Thermique, SFT 2018 (ISBN: 978-2-905267-95-5)*

Keywords: *infrared thermography, emissivity, reflections, Ray Tracing Monte Carlo, Radiosity method, inverse problem*

***Corresponding author:** *charly.talatizi@cea.fr*

Implementation of the FSP Pipe Option Design into Non-Standard FW Panel

P1-047

Sergey Tomilov [a]*, M. Sviridenko [a], A. Leshukov [a], E. Okuneva [b], V. Safronov [c], B. Calcagno [d], R. Eaton [d], S. Gicquel [d]

[a] *Joint-Stock Company "N.A. Dollezhall Research and Development Institute of Power Engineering", (JSC "NIKIET"), Moscow, Russia*
[b] *D.V. Efremov Scientific Research Institute of Electrophysical Apparatus, St. Petersburg, Russia*
[c] *Institution "Project Center ITER", Moscow, Russia*
[d] *ITER Organization, Saint Paul lez Durance, France*

The internal surface of the vacuum vessel of ITER is covered with 440 blanket modules.

Each blanket module consists of FW panel and shield block. The JSC NIKIET and the JSC NIIEFA are responsible for procurement of the FW panels. Each FW includes plasma-facing components (FW fingers), FW beam, mechanical attachment system and electrical connection system with the shield block. FW fingers are mechanically connected to FW beam and at the same time hydraulic connection of the FW beam to FW fingers is realized with milled adapters.

The FW panel full-scale prototype (FSP) was designed and approved by IO in order to demonstrate manufacturability and qualify critical technological processes. Design of other panels is supposed to be based on the FSP design. The JSC NIKIET and IO specialists attempted to implement the design of the FSP cooling circuit and FW fingers mechanical attachment system into non-standard FW panel – FW #14 type NE. Due to complexity of the FW beam shape the cooling circuit had to be modified. In order to justify the design, JSC NIKIET specialists performed hydraulic and thermal analysis of FW beam and FW fingers.

This paper presents the design of the FW panel #14 type NE, based on the FSP design, and proposes modifications for FW beam cooling system in order to satisfy ITER requirements.

Keywords: *Blanket system, EHF First wall, FW plasma-facing component, hydraulic analysis, thermal analysis, cooling circuit*

Corresponding author: *tomilov@nikiet.ru*

Beam Duct for the 1 MW Neutral Beam Injector on TCV

P1-048

Matthieu Toussaint *, Stefano Coda, Frédéric Dolizy, Basil Duval, Ambrogio Fasoli, Alexander Nicola Karpushov, Yves Martin, Roberto Maurizio and the TCV team

Ecole Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), Lausanne, Switzerland

The 1MW power of the neutral beam injector (NBI) enters the plasma of the Tokamak à Configuration Variable (TCV) through a duct. The original beam propagation model predicted a maximal heat flux on the internal faces of the duct below 350 kW/m^2, leading to an acceptable temperature rise for the 2s nominal pulse duration. During commissioning, the NBI showed unacceptable overheating in the duct, indicating a higher power density than expected. Several ion source grids have been tested to mitigate the beam divergence however the overheating of the duct is still problematic according to beam profile measurements. Since then the NBI operates with reduced power and duration to avoid damaging the beam duct.

This paper describes the design and thermal analysis of a new beam duct capable of withstanding up to 2 seconds of full power injection for a range of modeled neutral beam divergences. The design drivers are the high heat flux protection and the accumulated heat dissipation. The concept features a thermal shield composed of a smooth elliptical castellated graphite layer and an actively cooled structure.

Keywords: *TCV, Neutral Beam Heating Injector, Duct, Cooling, Thermal*

Corresponding author: *matthieu.toussaint@epfl.ch*

ITER VSRS First Mirror Plasma Cleaning in RF Gas Discharge – Circuit Design and Plasma Effects

P1-049

Andrey Ushakov [a]*, Ad Verlaan [a], Ulf Stephan [b], Olaf Steinke [b], Maarten de Bock [c], Matthew Maniscalco [a], Peter Verhoeff [a]

[a] *TNO, Delft, Netherlands*
[b] *FAP GmbH, Dresden, Germany*
[c] *ITER-IO, Saint-Paul-lez-Durance, France*

The line averaged measurement of the visible continuum emission of the core plasma (VSRS.55.E6) is going to be one of the first optical diagnostics in the ITER tokamak chamber. For the VSRS diagnostic system, the 100 mm diameter first mirror (FM) located at the front end of the optical system is expected to experience contamination with beryllium and tungsten. Small levels of contamination (on the order of

10 nm) can significantly degrade optical performance of the mirror. In the VSRS design, a plasma FM cleaning system will be implemented to maintain or restore optical performance. An RF discharge operating in a 30-60 MHz frequency range in an inert gas such as He Ar, or Ne is a candidate technology for the VSRS FM cleaning. In the present report, we determine the design requirements for such system and analyze the composition of the high frequency electrical circuit with a plasma load. Characteristics of critical components for the RF generator, RF matching unit, RF cables outside and inside the vacuum vessel, air-to-vacuum interfaces and the RF electrode are considered. The anticipated effect of the plasma cleaning is discussed based on the numbers of ion fluxes and energies obtained in the similar research program on the FM plasma cleaning development for the UWAVS ITER diagnostics. The proposed analysis may be useful for other diagnostic systems using RF plasma with first mirrors electrode facing plasma and connected to long RF feed and water cooling lines.

Keywords: *Plasma cleaning, first mirror, RF discharge, VSRS, plasma diagnostics*

Corresponding author: *andrey.ushakov@tno.nl*

High Heat Flux Tests of Divertor Mock-Ups at the HELCZA Facility for the EUROfusion DIVERTOR Project

P1-050

Jan Prokůpek, Richard Jílek, Josef Sklenka, **Ladislav Vála** *, Mark Murtazin, Karel Samec

Centrum výzkumu Řež, Husinec-Řež, Czech Republic

The experimental complex HELCZA is a high heat flux (HHF) test facility designed for cyclic heat loading for testing of plasma facing components, for instance ITER and DEMO first wall and divertor components and mock-ups. Plasma facing components can be tested in the multi-MW/m^2 range using an 800 kW electron beam gun, which electromagnetic system provides a beam scanning frequency of 20 kHz at the primary deflection angle up to ±40 degree. The HELCZA cooling system allows variation of the inlet cooling water temperature in a range from room temperature up to 320°C and water pressure up to 15 MPa. The cooling system provides an optimal flow rate up to 40 m^3/h independent of the pressure. Diagnostics for surface monitoring and mock-up measurements consist of infrared cameras, high-resolution cameras, X-ray camera, one and two colours pyrometers, thermocouples, temperature sensors, flow meters, and manometers.

In the framework of the EUROfusion DIVERTOR project, screening and long-pulse cycling tests on ITER-like phase 2 mock-ups are being performed at the HELCZA HHF facility. The later type of tests is performed in order to study a cycle-to-cycle behaviour of the mock-up performance, surface conditions and heat flux removal. The applied water-cooling parameters are: inlet temperature of 130°C, coolant velocity of 16 m/s velocity and pressure of 4 MPa.

The paper presents the HHF tests carried out in the framework of the EUROfusion DIVERTOR project together with results of the HHF tests performed with the mock-ups (tungsten, CuCrZr alloy and TZM molybdenum alloy).

Keywords: *divertor, HHF tests, high heat flux, plasma facing components, HELCZA*

Corresponding author: *ladislav.vala@cvrez.cz*

On the Thermal-Hydraulic Performances of the DEMO Divertor Cassette Body Cooling Circuit Equipped with a Liner

P1-051

P. A. Di Maio [a], R. Forte [a], R. Gaglio [a], J. H. You [b], G. Mazzone [c], **Eugenio Vallone** [a]*

[a] *Department of Engineering, University of Palermo, Palermo, Italy*
[b] *Max Planck Institute of Plasma Physics (E2M), Garching, Germany*
[c] *Department of Fusion and Technology for Nuclear Safety and Security, ENEA C.R. Frascati, Frascati (Roma), Italy*

In the framework of the Work Package DIV 1 - "Divertor Cassette Design and Integration" of the

EUROfusion action, a research campaign has been jointly carried out by University of Palermo and ENEA to investigate the steady-state thermal-hydraulic performances of the DEMO divertor cassette cooling system. The research activity has been focussed onto the most recent design of the Cassette Body (CB) cooling circuit, consistent with the DEMO baseline 2017 and equipped with a liner, whose main function is to protect the underlying vacuum pump hole from the radiation arising from plasma. The research campaign has been carried out following a theoretical-computational approach based on the finite volume method and adopting the commercial Computational Fluid-Dynamic (CFD) code ANSYS-CFX.

The CB thermal-hydraulic performances have been assessed in terms of coolant and structure temperature, coolant overall total pressure drop and flow velocity distribution, mainly in order to check coolant aptitude to provide a uniform and effective cooling to both CB and Liner structures. Moreover, the margin against coolant saturation has been evaluated in order check whether any risk of its bulk vaporisation is prevented.

The outcomes of the study have shown some criticalities, mainly in terms of structure maximum temperature and coolant vaporization occurrence within the Liner, that have suggested some design variations

whose effectiveness has been numerically assessed.

Models, loads and boundary conditions assumed for the analyses are herewith reported and critically discussed, together with the main results obtained.

Keywords: *DEMO, divertor, cassette body, CFD analysis, thermofluid-dynamics*

**Corresponding author:* *eugenio.vallone@unipa.it*

An Alternative Design Concept of DEMO Relevant Liquid Lithium Divertor Target Based on Capillary Structures

P1-052

Alexey Vertkov [a]*, I. Lyublinski [a,b], M. Zharkov [a], A. Berlov [a], A. Komov [c], A. Zakharenkov [c]

[a] *SC "Red Star", Moscow, Russia*
[b] *NRNU MEPhI, Moscow, Russia*
[c] *National Research University "Moscow Power Engineering Institute" , Moscow, Russia*

The alternative design of DEMO relevant liquid lithium divertor target is proposed and discussed. The heat exhaust opportunity under DEMO conditions at power flux of 10–20 MW/m^2 is analyzed on the base of comprehensive consideration the following aspects: liquid metal properties, CPS structure parameters, structural material properties, cooling scheme and coolant properties. It has been shown that thin-walled target design with Li CPS based plasma-facing material and advanced composite structural material of 67%W fiber+ 33% Eurofer type steel including cooling system with promising coolant on the base of gas-water spray provides heat exhaust opportunity at target surface temperature not higher of 700°C. It was find out that power flux limit for Sn or Sn-Li CPS design is not satisfy of DEMO demands for unacceptable liquid metal corrosion activity. Experimental validation results on gas-water coolant heat removal efficiency, estimation of Li CPS divertor target safety and design of liquid lithium divertor target of T-15U tokamak are presented.

Keywords: *liquid metals, capillary porous structure, plasma facing material, plasma facing component, design, DEMO*

**Corresponding author:* *avertkov@yandex.ru*

Thermal and Mechanical Analyses of W7-X Plasma Facing Components for Operation Phase 2

P1-053

Zhongwei Wang *, Felix Schauer

Max-Planck-Institut fuer Plasmaphysik, Garching, Germany

After the successful short pulse operation phase 1 (OP1) of the stellarator Wendelstein 7-X with maximal plasma energies of 200 MJ, the upcoming long pulse OP2 aims at stepwise higher energies up to 18 GJ. With the knowledge of partly unexpected experimental heat load distributions on plasma facing components (PFC), their allowable loads were re-evaluated which consequently lead to some adaptation work.

The divertor target modules TM5h and TM6h were loaded more than expected during OP1. This triggered a detailed transient and stationary analysis which revealed a high sensitivity of the thermomechanical response to the heating power distribution.

At the baffles, cracks were detected just before assembly for OP1 at the braze joints between CuCrZr heat sinks which carry the graphite tiles, and stainless steel (SS) cooling pipes. Due to large temperature gradients between parts of the pipes and support structure the braze becomes heavily stressed during the high loads in OP2. Detailed crack initiation and propagation analyses led to the decision to release thermomechanical stresses by loosening the connections between the heat sinks and steel structures of the modules.

Finally, new SS wall protection panels to be positioned behind the divertor pumping gaps for stationary OP2-loads of 100 kW/m² are being manufactured. Calculations were performed to optimize the intricate cooling channels and to lead as well as confirm the mechanical design for all working cases, from factory tests to normal and abnormal operation.

The findings of these analyses lead to adaptation of the plasma operating instructions for OP2.

Keywords: *Stellarator, Wendelstein 7-X, plasma facing components, thermomechanical analysis*

**Corresponding author:* *zhongwei.wang@ipp.mpg.de*

Microstructure Characterization of the Key Material Interfaces in ITER First Wall Components

P1-054

Pinghuai Wang [a]*, Jiming Chen [a], Qian Li [a], Danhua Liu [a], Pan Huang [a], Yanyu Chen [a], Zhenxing Wei [a], Kun Wang [b]

[a] *Southwestern Institute of Physics, Chengdu, China*

[b] *China International Nuclear Fusion Energy Program Execution Center, Beijing, China*

China is manufacturing the enhanced heat flux first wall components for ITER. Hot iso-static pressing (HIP) and explosion bonding techniques are chosen for joining the plasma facing material - beryllium tiles to the CuCrZr-IG alloy heat sink and the joining of the alloy to the 316L(N)-IG structural material, respectively. The properties of the bonding interfaces are considered critical for the performance of the components under 15000 cyclic heat flux loading up to 4.7MW/m^2. In this work, the micro-structure of the two interfaces were studied. It is found that several kinds of intermetallic compound layers formed in the Be/CuCrZr interface during HIP joining them at 590°C and 150MPa for 2h. The Ti/Cu coating on the Be tiles as Be/Cu diffusion barrier, the inter-layer metals for thermal stress relieving and the Post-HIPing heat treatment to reduce residual stress strongly affect the formation and evolution of the interfacial microstructure, and the joining properties of the components as a consequence. For the CuCrZr/316L(N) interfaces, the adiabatic shear line at the 316L(N) side, the vortex structure at explosive bonding interface and the Zirconium segregation on the CuCrZr side were observed after solution annealing, which may cause brittle cracks especially under pulsing high heat flux. Several measures are taken to improve the properties of CuCrZr/316L(N) bonding.

Corresponding author: *wangph@swip.ac.cn*

3D Modelling of the Toroidally-Localized Lithium Powder Injection Experiments on EAST with EMC3-EIRENE

P1-055

Tian Xie [a], S.Y. Dai [a]*, Z. Sun [b], G.Z. Zuo [b], L. Wang [b], J.S. Hu [b], Y. Feng [c], D.Z. Wang [a]*

[a] *Key Laboratory of Materials Modification by Laser, Ion and Electron Beams (Ministry of Education), School of Physics, Dalian University of Technology, Dalian, China*

[b] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*

[c] *Max-Planck-Institut für Plasmaphysik, Greifswald, Germany*

Three dimensional (3D) edge plasma fluid and kinetic neutral transport code package EMC3-EIRENE has been employed to study lithium transport in the scrape-off layer (SOL) of EAST with the toroidally-localized lithium powder injection. The simulated profile of Li ions density shows that the Li^{1+} ions mainly populate at the upper private region, while the Li^{2+} and Li^{3+} ions distribute at the upper X-point and upstream SOL regions. The synthetic 3D line-integrated LiII emission (emitted by Li^{1+} ions) image obtained by EMC3-EIRENE is in reasonable agreement with the experimental data from the CCD camera system installed on EAST. The impacts of the poloidal locations of Li injection have been studied to evaluate the variation of the Li impurity transport and emission behaviors, which shows a strong dependence on the Li injection position. It is found that the Li injection at the midplane is the most effective position to radiate the power and reduce the heat flux deposition on the lower divertor targets compared to other poloidal injection locations.

[1] *Feng Y. et al 2004 Contrib. Plasma Phys. 44 57–69*

[2] *Reiter D. et al 2005 Fusion Sci. Technol. 47 172*

[3] *Dai S.Y. et al 2016 Nucl. Fusion 56 066005*

[4] *Dai S.Y. et al 2018 Nucl. Fusion 58 096024*

[5] *Xie T. et al 2018 Fusion Eng. Des. 136 699-705*

[6] *Xie T. et al 2018 Nucl. Fusion 58 106017*

Keywords: *Impurity transport, Lithium powder, Edge plasma, EMC3-EIRENE, Impurity emission*

****Corresponding author:*** *daishuyu@dlut.edu.cn, wangdez@dlut.edu.cn*

Magnetohydrodynamic Effect of Multilayer Liquid Metal Film Flow on the Inclined Surface

P1-056

Juan-Cheng Yang [a], Tian-Yu Qi [b], Bai-Qi Liu [b], Ming-Jiu Ni [a,b]*

[a] *State Key Laboratory for Strength and Vibration of Mechanical Structures, School of Aerospace, Xi'an Jiaotong University, Xi'an, Shaanxi, China*

[b] *College of Engineering Science, University of Chinese Academy of Sciences, Beijing, China*

The utilization of liquid metal film flows to protect the plasma-facing components is a promising method in fusion devices. However, inside the fusion device, the extreme conditions such as a strong magnetic field, high temperature difference will influence the stability of the flowing liquid metal layer. Much more effort needs to be done to understand the magnetohydrodynamics (MHD) of such flows in a magnetic field. In the present paper, we design a test section to generate multilayer liquid GaInSn film flowing on a PMMA surface and carry out a systematic experiment to discover the flow characteristics of the liquid GaInSn film under the influence of transverse magnetic field. By introducing three Laser Profilometers (LPs), we obtain the instantaneous surface contour of GaInSn film flow at three fixed lines. The experimental results show that the surface fluctuation of the liquid film increase obviously with the increase of Reynolds number. The magnetic field can effectively inhibit the surface fluctuations along the magnetic lines and leave the waves alone in the flow direction. The thickness of GaInSn film increase non-monotonically with the increase of the strength of magnetic field, which means the existence of a competition between Lorentz force and inertial force when magnetic field introduced. Moreover, results also indicate that the multilayer film flow device can effectively improve the spreading of liquid metal on a solid surface.

Keywords: *liquid metal, Multilayer film flow, magnetohydrodynamic, plasma facing components*

Corresponding author: *mjni@ucas.ac.cn, yangjc@xjtu.edu.cn*

Fracture Mechanical Analysis of Divertor Plasma-Facing Component Designs Using Tungsten Particle-Reinforced Copper Composite Heat Sink

P1-057

Kuo Zhang [a]*, Alexander von Müller [a,b], Henri Greuner [a], Jeong-Ha You[a]

[a] *Max Planck Institute for Plasma Physics, Garching, Germany*

[b] *Technische Universität München, Garching, Germany*

Tungsten-copper (W-Cu) composite materials are currently considered as advanced heat sink materials for highly heat loaded plasma-facing components (PFCs). Against this background, design concepts of such W-Cu composites are being investigated within the framework of the EUROfusion DEMO divertor project [1]. One class of such materials are W particle-reinforced Cu composites that can be joined to pure tungsten tiles as plasma-facing armour. Depending on the material composition, the W-Cu composite material can exhibit a notably reduced coefficient of thermal expansion (CTE) compared to Cu materials, which in turn can reduce thermal stresses at the W to heat sink joint. High stresses near the free edge of a bonding interface in a flat-tile PFC design can cause failure of or near the interface, which is why a reduction of the CTE mismatch is considered a major design advantage.

In this regard, PFC mock-ups with a W particle-reinforced Cu composite heat sink of 70 wt% W - 30 wt% Cu has been fabricated and high heat flux tested. The high heat flux tests with cyclic loading of 20 MW/m^2 were performed at the neutral beam facility GLADIS (IPP Garching). In this study, the fracture and fatigue behaviour of this design concept is simulated by means of FEM studies to assess the reliability and robustness of this design concept, focusing both on joint interface and bulk material, by introducing initial crack and investigating the crack propagation under thermal-induced loading due to static as well as cyclic heat-flux. The advantages and disadvantages of this design concept are discussed and compared to other concepts of plasma-facing components.

[1] *J.H. You, "European divertor target concept for DEMO: Design rationales and high heat flux performance", Nuclear Materials and Energy 16 (2018) 1-11*

Keywords: *DEMO divertor, W-Cu composite, FEM study*

Corresponding author: *kuo.zhang@ipp.mpg.de*

Molecular Dynamics Simulation Study of the Interaction between Energetic Incident D Species with Rough Tungsten Surface

P1-058

Hongyu Zhang [1], **Jizhong Sun** [1*], Ali Y. Hamid [1,2], Thomas Stirnerb [3]

[1] *School of Physics and Optoelectronic Technology, Dalian University of Technology, Dalian, China*
[2] *Department of Physics, University of Nyala, Nyala, Sudan*
[3] *Department of Applied Natural Science, University of Applied Sciences Deggendorf, Deggendorf, Germany*

Tungsten has been used for plasma facing materials of tokamaks. Under severe work conditions in future tokamaks, the tungsten surface will soon turn rough, resulting from events such as erosion, blistering, blister bursting, cracking, and melting. In this paper, the effects of energetic D species impinging tungsten substrate are studied using molecular dynamics simulation. Results indicate that the retention rate, depth distribution and energy deposition of deuterium increase with the enhanced roughness. Further analysis show that the atomic binding energy drops as the surface goes rough, explaining why the tungsten atoms of rougher surface are easier to be sputtered out. Besides, deuterium irradiation leads to the reduction of the vacancy formation energy, shedding light to the cause why the D atoms retained in the tungsten are pumped out more easily during continuous D irradiation, which suggests that the tritium retained in plasma facing walls could be replaced by isotope exchange The later result provides the research basis for our future work on extraction of tritium retained in materials.

This work is supported by the National Key R&D Program of China under Grant No. 2017YFE0301100 and the National Science Foundation of China under grant Nos. 11575039 and 51671045. It is also partially supported by the Fundamental Re6 search Funds for the Central Universities (DUT18GF112) and by Supercomputing Center of Dalian University of Technology.

Keyword: *rough tungsten, tokamak, deposit*

****Corresponding author:*** *jsun@dlut.edu.cn*

Laminated Composites Using Potassium Doped Tungsten

P1-059

Shuhei Nogami [a*], Takefumi Hazama [a], Amon Takahashi [a], Ryota Kanamaru [a], Hiroyuki Noto [b], Takuya Nagasaka [b], Akira Hasegawa [a]

[a] *Tohoku University, Sendai, Japan*
[b] *National Institute for Fusion Science, Toki, Japan*

Tungsten (W) is promising as a plasma facing material of fusion reactor divertor. However, there remain some drawbacks of the mechanical properties of W, which are low temperature brittleness, high ductile-to-brittle transition temperature (DBTT), and recrystallization-induced embrittlement.

It is known that highly deformed W foils show ductility even at room temperature. However, their thickness (volume) is not enough for divertor application. To satisfy both low temperature ductility and material volume, laminated composite using pure W foil was developed, which is a composite material consisting of stacked pure W foils and interlayer material. However, it was concerned that most of highly deformed materials show low resistance to recrystallization.

Potassium (K) doping is known as a method for dispersion-strengthening and suppression of recrystallization of W materials. K-doped W contains nano-bubbles including K atoms, which are mainly dispersed at the grain boundaries. Because K bubbles can hinder the motion of grain boundaries and dislocations, they lead to strengthening at a high temperature and suppression of recrystallization compared to the pure W.

The laminated composites using K-doped W materials (cold-rolled foils and hot-rolled plates) have been developed, which are expected to have a higher resistance to recrystallization compared to the pure W based laminates. A laminate using a K-doped W foil, which was fabricated at 1250 °C, showed room temperature ductility. In this presentation, their deformation and fracture behaviors will be discussed.

Keywords: *Tungsten, Laminated composite, Potassium doping, Ductility, Pseudo ductility*

****Corresponding author:*** *shuhei.nogami@qse.tohoku.ac.jp*

Corrosion Behavior of Multi-layer Ceramic Coatings in Liquid Lithium-Lead

P1-060

Erika Akahoshi [a], Moeki Matsunaga [a], Keisuke Kimura [a], Kazuki Nakamura [a], Kazuki Saito [b], Hikari Fujita [c], Martin Balden [d], Yoshimitsu Hishinuma [e], Takumi Chikada [a]*

[a] *Shizuoka University, Suruga-ku Shizuoka, Japan*
[b] *The Graduate University for Advanced Studies (SOKENDAI), Toki, Gifu, Japan*
[c] *The University of Tokyo, Tokyo, Japan*
[d] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[e] *National Institute for Fusion Science, Toki, Gifu, Japan*

Tritium permeation through structural materials in fusion reactor blanket systems is a critical issue from the viewpoints of radiological safety and fuel loss. Ceramic coatings have been investigated as a tritium permeation barrier and showed high permeation reduction performance. In liquid blanket concepts, however, corrosion of the coatings by liquid tritium breeders such as lithium-lead (Li-Pb) is an unavoidable concern. It was found that two-layer coatings using erbium oxide (Er_2O_3) and zirconium oxide (ZrO_2) indicated higher Li-Pb corrosion resistance than single layer coatings. In this study, further multi-layer coatings were fabricated and exposed to Li-Pb under static conditions to investigate the effects of layer structure on Li-Pb compatibility.

Er_2O_3 and ZrO_2 coatings were fabricated by metal-organic decomposition for 4 times on each reduced activation ferritic/martensitic steel F82H substrate to obtain 2–4-layer structures, e.g., F82H-Er_2O_3-Er_2O_3-ZrO_2-ZrO_2 and F82H-Er_2O_3-ZrO_2-Er_2O_3-ZrO_2. After static Li-Pb exposure for 500–1000 h at 500–600 °C, the surfaces of the coatings and cross-sections through the coating were analyzed by scanning electron microscopy with energy dispersive X-ray spectroscopy with the aid of a focused ion beam system.

After exposure tests at 500 and 550 °C, all the coatings showed no cracks and peelings, and the surfaces were smooth, indicating corrosion did not occur. In the cases of the exposure at 600 °C, many cracks and peelings were observed in 4-layer coatings and all the coatings seemed to be corroded. The crack formation was attributed to the difference in coefficients of thermal expansion between Er_2O_3 and ZrO_2, which allows invasion of Li-Pb into Er_2O_3-ZrO_2 coating interfaces resulting in peeling. Results of deuterium permeation measurements for the coatings in contact with Li-Pb will be included in the presentation.

Keywords: Lithium-lead, Corrosion, Coating, Erbium oxide, Zirconium oxide

**Corresponding author: chikada.takumi@shizuoka.ac.jp*

Design and Preliminary Analyses of the New Water Cooled Lithium Lead TBM for ITER

P1-061

Julien Aubert [a]*, G. Aiello [a], D. Alonso [e], T. Batal [b], R. Boullon [a], S. Burles [b], B. Cantone [b], F. Cismondi [c], A. Del Nevo [d], L. Maqueda [e], A. Morin [a], E. Rodríguez [e], F. Rueda [e], M. Soldaini [b], J. Vallory [f]

[a] *DEN-DM2S, CEA, Université Paris-Saclay, Gif-Sur-Yvette, France*
[b] *CEA/IRFM, St. Paul lez Durance Cedex, France*
[c] *EUROfusion Consortium, Programme Management Unit, Garching, Germany*
[d] *ENEA FSN-ING, CR Brasimone, Camugnano (BO), Italy*
[e] *ESTEYCO S.A., Madrid, Spain*
[f] *Fusion for Energy (F4E), Barcelona, Spain*

In the European strategy, DEMO is the intermediate phase between ITER and a commercial fusion power plant. In this framework, one of the goal of DEMO is to be a Breeding Blanket test facility. The Breeding Blanket, which is not present in ITER, is one of the key components for the future deployment of nuclear fusion electricity as it accomplishes the functions of tritium breeding and nuclear to thermal power conversion.

Due to time constraints lead by the construction schedule of DEMO, a new strategy to consider in DEMO a "driver" Breeding Blanket that needs limited technological extrapolation has been chosen, while "advanced" Breeding Blanket concepts will be tested in the next phases. In this context, ITER will allow to test technologies to provide relevant contributions in terms of Return of eXperience to the DEMO "driver" Breeding Blanket project by the mean of Test Blanket Modules (TBM) to be installed in different ITER Vacuum Vessel Ports.

Among the possible "driver" Breeding Blanket, the Water Cooled Lithium Lead (WCLL) concept comes out. In this framework, a realignment of the DEMO Breeding Blanket and TBM programs has started in 2017, leading to a new TBM development relevant of the DEMO WCLL BB. The WCLL TBM is therefore an essential component in ITER that will provide crucial information for the development of the DEMO "driver" blanket.

This paper aims at presenting the development process and design status of WCLL TBM set. After recalling the main features of the WCLL TBM set, conceptual design analyses are presented and discussed.

Keywords: ITER, WCLL, TBM, Breeding Blanket

**Corresponding author: Julien.aubert@cea.fr*

Long-Term Corrosion Behavior of EUROFER RAMF Steel in Static Liquid Pb-16Li at 550°C

P1-062

Serena Bassini *, Vincenzo Cuzzola, Andrea Antonelli, Marco Utili

ENEA FSN-ING Division, C.R. Brasimone, Camugnano, Italy

Liquid Pb-16Li eutectic alloy is the tritium breeding material in several breeding blanket concepts for European DEMO reactor, which are namely the Helium-Cooled Lithium-Lead (HCLL), the Water-Cooled Lithium-Lead (WCLL) and the Dual-Coolant Lithium-Lead (DCLL). RAMF steels (Reduced-Activation Martensitic/Ferritic steels) are considered as structural materials for breeding blankets in fusion reactors and EUROFER steel (9Cr-1.1W-0.2V) is the European current choice for DEMO. Moreover, the WCLL European Test Blanket Module (TBM) to be installed in ITER will be manufactured in EUROFER. Here, the chemical compatibility of EUROFER in Pb-16Li is a major issue due to the dissolution of the material at high temperature in the liquid metal. In this work the corrosion behavior of EUROFER steel in static Pb-16Li melt was investigated at 550°C up to 8000 hours of exposure with intermediate time spans at 2000 and 4000 hours. For each time spans SEM and EDX analysis were performed on corroded samples to assess microstructural and compositional changes occurring during the exposure. The corrosion was numerically evaluated by weight loss determination on samples exposed to the various exposure times and the corrosion rate for one year of exposure (8760 hours) was extrapolated. The analysis by SEM-EDX shows that EUROFER mostly dissolved uniformly with a depletion of chromium only at the near interface and the weight loss evaluation indicates a corrosion rate of ≈ 20 μm/yr at 550°C.

Keywords: *Pb-16Li, EUROFER steel, corrosion*

**Corresponding author:* *serena.bassini@enea.it*

Advancements in the HELIAS 5-B Breeding Blanket Structural Analysis

P1-063

Gaetano Bongioví *, A. Häußler and the W7-X team

Karlsruhe Institute of Technology (KIT), Institute for Neutron Physics and Reactor Technology (INR), Eggenstein-Leopoldshafen, Germany

Within the framework of EUROfusion consortium, the Work Package S2 aims at developing the HELIcal-axis Advanced Stellarator (HELIAS) as a possible long-term alternative to a tokamak DEMO.

From the plasma physics standpoint the most promising concept is the HELIAS 5-B machine, a large 5 field period stellarator reactor directly extrapolated from Wendelstein 7-X. An intense research campaign has been launched at KIT in order to attain a preliminary design of the HELIAS 5-B breeding blanket (BB), taking into account as initial input the design experience acquired in the pre-conceptual design phase of the tokamak DEMO BB. To this end, the Helium-Cooled Pebble Bed (HCPB) breeding blanket concept has been considered, focusing on the investigation of the suitability of its main structural features to the stellarator geometry. Design requirements coming from the Remote Maintenance have been taken into account in order to orient the blanket segmentation and a more sophisticated assessment of the BB modules structural behavior has been performed. Attention has been also paid to the refinement of the numerical models so far adopted, investigating the impact of some assumptions on the obtained results. The achieved advancements in this HELIAS 5-B BB design are herewith presented and critically discussed, indicating a possible way for the follow up of this activity.

Keywords: *HELIAS, stellarator, breeding blanket, thermomechanics, FEM analysis*

**Corresponding author:* *gaetano.bongiovi@kit.edu*

Development of a WCLL DEMO First Wall Design Module in the SYCOMORE System Code Interfaced with the Neutronic One

P1-064

Rémi Boullon *, Julien Aubert, Jean-Charles Jaboulay, Giacomo Aiello

DEN-DM2S, CEA, Université Paris-Saclay, Gif-Sur-Yvette, France

The pre-conceptual design of the DEMOnstration reactors has already started and several tokamak configurations have to be tested to find the best design by exploring different design parameters. Fast simulations involving the different components behavior must be performed. Within the European framework, SYCOMORE (SYstem COde for MOdelling tokamak REactor) is developed by CEA for this purpose. The Breeding Blanket facing the plasma is a key component in DEMO ensuring tritium self-sufficiency, shielding against neutrons and heat extraction for electricity production. Several Breeding Blanket concepts are being studying, among which the Water Cooled Lithium Lead (WCLL) one. SYCOMORE includes several specific modules linked together, one of which has been developed to define a suitable design of the WCLL Breeding Blanket and is presented in this paper.

The method to define automatically the WCLL First Wall (FW) design using analytical design formulae starting from thermo-hydraulic and thermo-mechanical considerations as well as design criteria coming from Codes & Standards is presented. Moreover, the FW thermo-hydraulic

limits – due to the deterioration of water thermal properties versus high HF – are taken into account in order to avoid any Departure from Nucleate Boiling (DNB) due to Critical Heat Flux (CHF). In parallel, the Onset of Nucleate Boiling (ONB) is also verified. Furthermore WCLL FW design obtained with SYCOMORE is compared to 3D FW computed with Cast3M FEM code.

Finally, a coupling between thermo-mechanical and neutronic is implemented, several iteration are necessary to obtain a converged design. Neutronic module evaluates the BB tritium production, and the nuclear heating in the FW and the Breeding zone (used by thermo-mechanical module). Thermo-mechanical module gives the design data (FW thickness, compositions, etc.) to the neutronic module.

Keywords: *DEMO, SYCOMORE, System Code, WCLL, Breeding Blanket*

Corresponding author: *remi.boullon@cea.fr*

Experimental Investigation of Liquid Metal MHD Flow Entering a Flow Channel Insert

P1-065

Leo Bühler *, Chiara Mistrangelo, **Hans-Jörg Brinkmann**

Karlsruhe Institute of Technology, Karlsruhe, Germany

Liquid lead lithium (PbLi) is foreseen as breeder material, neutron multiplier and heat transfer medium in dual coolant (DCLL), helium-cooled (HCLL) and in water-cooled (WCLL) lead lithium blankets. In all these applications, the liquid metal has to be circulated under the influence of the strong plasmaconfining magnetic field. The interaction with the magnetic field induces electric currents that cause strong electromagnetic Lorentz forces resulting in very high pressure drops, especially when duct walls are electrically conducting. Currents and pressure drops may be reduced by using electrically insulating flow channel inserts (FCI) that decouple the fluid region from the well conducting walls. FCIs have been proposed for DCLL blankets, which rely on convective heat transport by the liquid metal flow. However, in HCLL and in WCLL blankets, FCIs could be beneficial as well for supplying lines and manifolds, in which the velocities are higher than in the breeding zones.

In the present paper, we consider the entrance flow of an electrically conducting fluid into a FCI under the influence of a strong uniform magnetic field. The abrupt change in wall conductivity leads to 3D effects that are investigated by measurements of pressure and electric potential distribution along the duct. It is shown that FCIs may reduce pressure drop by at least one order of magnitude. Moreover, the obtained experimental data will be useful for future comparison with numerical simulations for validation of predictive computational tools.

Keywords: *liquid metal experiments, magnetohydrodynamics (MHD), blankets, MHD pressure drop*

Corresponding author: *leo.buehler@kit.edu*

An Integrated Hydrogen Isotopes Transport Model for the TRIEX-II Facility

P1-066

Luigi Candido [a]*, Mattia Cantore [a], Edoardo Galli [a], Raffaella Testoni [a], Marco Utili [b], Massimo Zucchetti [a]

[a] *Dipartimento Energia "Galileo Ferraris" – Politecnico di Torino, Torino, Italy*

[b] *ENEA FSN-ING-QMN – C.R. Brasimone, Camugnano, Italy*

The experimental facility TRIEX-II (TRItium EXtraction) has been designed and installed at ENEA Brasimone Research Centre to verify the efficiency of the hydrogen isotopes extraction unit (TEU) from the flowing metallic alloy Pb-15.7Li in the range of operating conditions foreseen for the European Test Blanket System WCLL-TBS.

The model proposed presents an integrated modelling tool able to dynamically quantify the hydrogen isotopes (Q_2) mass balance for the Gas/Liquid Contactors (GLCs), in particular the packed-column technology. This tool, starting from component-detail level models developed in COMSOL for both the saturation column and the extraction column mock-up, is able to describe, using MATLAB/Simulink, the whole TRIEX-II circuit at system level, in order to quantify the Q_2 retention, leakages and permeation and to verify the hydrogen isotopes mass balance, and, then, the theoretical extraction efficiency of the mock-up. The integration is carried out by implementing the COMSOL-developed components into the discrete S-functions of MATLAB/Simulink, preserving the process flow diagram of the loop. In this way, it is possible to have a comparison with the experimental results and to validate the integrated model in view of design improvements and safety-related issues for the tritium cycle of ITER.

Results deriving from the developed model, input and boundary conditions are illustrated in detail within the paper.

Keywords: *WCLL, ITER, hydrogen isotopes, transport, MATLAB/Simulink, COMSOL*

Corresponding author: *luigi.candido@polito.it*

Parametric Study of the Influence of Double-Walled Tubes Layout on the DEMO WCLL Breeding Blanket Thermal Performances

P1-067

Ilenia Catanzaro [a]*, Pietro Arena [b], Alessandro Del Nevo [b], Pietro Alessandro Di Maio [a], Francesco Edemetti [c], Emanuela Martelli [b], Ruggero Forte [a]

[a] *Department of Engineering, University of Palermo, Viale delle Scienze, Palermo, Italy*

[b] *ENEA FSN, ENEA CR Brasimone, Camugnano, Italy*

[c] *Department of Astronautical, Electrical and Energy Engineering, Sapienza University of Rome, Roma, Italy*

Within the framework of the EUROfusion activities regarding the EU-DEMO Breeding Blanket (BB) concept, the University of Palermo is long-time involved, as ENEA linked third-party, in the design of the Water Cooled Lithium Lead (WCLL) BB, that is one of the two concepts under consideration for the DEMO reactor. It is mainly characterized by a liquid lithium-lead eutectic alloy acting as breeder and neutron multiplier, as well as by subcooled pressurized water flowing as coolant under PWR-like conditions (pressure of 15.5 MPa and inlet/outlet temperatures of 295 °C/328 °C). Two separate circuits are deputed to cool down the Segment Box (First Wall - Side Walls) and the Breeder Zone (BZ). The former consists in a system of radial-toroidal-radial C-shaped squared channels, where countercurrent water flow occurs, whereas the latter relies on the use of bundles of Double Walled Tubes (DWTs) submerged within the breeder. Both circuits have to guarantee that the structural material maximum temperature does not overcome the prescribed EUROFER allowable limit of 550 °C. A research campaign has been recently carried out to study the potential influence of the DWTs layout on the BZ thermal behaviour. In particular, a parametric study has been performed to assess the effects of several geometrical DWT parameters onto the blanket thermal response, assuming a cooling circuit composed of horizontal DWTs, arranged along toroidal-radial planes. The study has been carried out following a theoretical-numerical approach based on the Finite Element Method (FEM) and adopting the quoted ABAQUS v. 6.14 commercial code. Model, assumptions and results obtained are herewith presented and critically discussed.

Keywords: DEMO, WCLL breeding blanket, DWT, FEM

**Corresponding author: ilenia.catanzaro@unipa.it*

Damage of Titanium Beryllide under High-Dose Neutron Irradiation

P1-068

Vladimir Chakin [a]*, Rolf Rolli [a], Ramil Gaisin [a], Masaru Nakamichi [b]

[a] *Karlsruhe Institute of Technology, Institute for Applied Materials, Eggenstein-Leopoldshafen, Germany*

[b] *Quantum and Radiological Science and Technology, Rokkasho Fusion Institute, Aomori, Japan*

Titanium beryllide $Be_{12}Ti$ is a promising material as neutron multiplier for the DEMO breeding blanket because it has comparative advantages such as lower swelling, higher stability in water steam, etc. to pure beryllium. Neutron irradiation of advanced beryllium materials in a material testing nuclear reactor up to helium and tritium productions close to the DEMO blanket parameters can provide essential results for the DEMO blanket design. In this study, titanium beryllide and pure beryllium pellets fabricated by the arc-melted method were irradiated in the HFR at temperatures of 438, 525, 664, 768 °C up to 430, 550, 625, 653 appm tritium, 4144, 5142, 5757, 5992 appm helium, corresponding to 23.3, 30.9, 35.6, 37.5 dpa, respectively, with following post-irradiation examinations (PIE's). The PIE's included the temperature-programmed desorption (TPD) tests focusing on tritium and helium release and retention, measurements of micro-hardness, study of cross-sections of irradiated pellets by the optical microscope.

The total tritium release (or retention) from irradiated titanium beryllide at three highest irradiation temperatures is significantly lower than that from pure beryllium. In particular, at temperatures of 664 and 768 °C, the total tritium release from titanium beryllide is close to zero while from pure beryllium that reaches of 3000 MBq/(g·s). The optical images show a two-phase structure in the titanium beryllide pellets such as coarse grains of $Be_{12}Ti$ phase and thin layers of beryllium phase between them. The relative part of the beryllium phase decreases on increasing irradiation temperature from 18.7 % at 438 °C to 16.1 % at 768 °C. Micro-hardness HV0.1 of $Be_{12}Ti$ phase is much higher than that of beryllium phase, 1350-1450 to 380-480, respectively. The enhanced tritium release from titanium beryllide pellets after irradiation at high temperatures can be explained by the formation of high porosity that leads to the formation of open structural channels for tritium release through them.

Keywords: titanium beryllide, neutron irradiation, tritium, porosity, thermal desorption

**Corresponding author: vladimir.chakin@kit.edu*

Simulations toward the Ultimate Parameter of Liquid Metal Fusion Blanket, Part I: High Hartmann Number MHD Flow

P1-069

Long Chen [a], Sergey Smolentsev [b], Ming-Jiu Ni [a]*

[a] *School of Engineering Science, University of Chinese Academy of Sciences, Beijing, China*

[b] *Mechanical and Aerospace Engineering Department, University of California at Los Angeles, Los Angeles, USA*

Numerical simulations of liquid metal magnetohydrodynamic (MHD) flows in a prototypical DCLL (dual coolant lead lithium) blanket module, where environmental conditions and operation parameters are close to the real system with Ha=1.1×10^4, Re = 6.5×10^5, have been performed to study the flow distribution and to predict the pressure drops and velocity distributions. The complex induced electric currents and 3D MHD phenomena, occur in distributing and collecting liquid metal manifolds, reveal the great influence on the flow distribution in parallel poloidal ducts. Moreover, MHD flows are studied for different design options of a flow channel insert (FCI) placed position with the aim of reducing the MHD pressure drops. The result shows that the pressure drop can be greatly decreased (reduced by 79.40%) when FCIs are placed inside the inlet duct and outlet duct, while undesirable flow imbalance will be formed.

Keywords: *magnetohydrodynamic, DCLL blanket, FCI, pressure drop, high Hartmann number*

Corresponding author: *mjni@ucas.ac.cn*

Compatibility of Tritium Permeation Barrier Coatings with Ceramic Breeder Pebbles

P1-070

Takumi Chikada [a]*, Matthias Kolb [b], Hikari Fujita [c], Kazuki Nakamura [a], Keisuke Kimura [a], Marcin Rasinski [d], Yoshimitsu Hishinuma [e], Regina Knitter [b]

[a] *Shizuoka University, Shizuoka, Japan*

[b] *Karlsruhe Institute of Technology, Karlsruhe, Germany*

[c] *The University of Tokyo, Tokyo, Japan*

[d] *Forschungszentrum Jülich GmbH, Jülich, Germany*

[e] *National Institute for Fusion Science, Gifu, Japan*

Strict control of tritium migration is an essential requirement for every blanket concept in a fusion reactor. Tritium permeation barriers (TPBs) have been developed in particular for liquid lithium-lead blankets; however, recent DEMO reactor design activities initiate an argument that a TPB is necessary also in solid breeder blankets. The previous studies showed that solid breeder lithium ceramics reacted with reduced activation ferritic/martensitic steels at elevated temperature, indicating the lithium reactivity of solid breeders should be taken into consideration. In this study, compatibility tests for TPB coatings with lithium ceramic pebbles have been carried out in order to assess the chemical stability of TPB coatings in the helium cooled pebble bed blanket concept. Erbium oxide (Er_2O_3) coatings fabricated by filtered vacuum arc deposition and metal-organic decomposition, and chromium oxide (Cr_2O_3) layers formed on reduced activation ferritic/martensitic steel F82H substrates by heat treatment were used as TPB coating samples. Ceramic breeder pebbles of lithium orthosilicate with 30 mol% of lithium metatitanate were put on the samples and then annealed for 2–32 days at 550 °C under 20 standard cubic centimeter per minute He with 0.1 vol% H_2 flow. No change in microstructure was confirmed for the Er_2O_3 coatings, while the Cr_2O_3-formed samples showed drastic changes in surface and cross-sectional structures including oxidation of F82H, reduction of iron oxide and chemical reactions by lithium. Further discussion on chemical reactions of the Cr_2O_3 layer and stability of Er_2O_3 coatings at 700 °C will be included in the presentation.

Keywords: *ceramic breeder, tritium, permeation, corrosion, coating*

Corresponding author: *chikada.takumi@shizuoka.ac.jp*

Investigation of the DEMO WCLL Breeding Blanket Cooling Water Activation

P1-071

Pierluigi Chiovaro [a]*, S. Ciattaglia [b], F. Cismondi [b], A. Del Nevo [c] P.A. Di Maio [a], G. Federici [b], C. Frittitta [a], I. Moscato [a], A. Spagnuolo [d], E. Vallone [a]

[a] *Dipartimento Ingegneria, Università di Palermo, Palermo, Italy*

[b] *EURO fusion Consortium, Garching, Germany*

[c] *ENEA Brasimone I, Camugnano, Italy*

[d] *Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany*

Within the framework of the activities foreseen by the EUROfusion action on the cooling water activation assessment for a DEMO reactor equipped with a Water Cooled Lithium Lead Breeding Blanket (WCLL BB), the University of Palermo is involved in the investigation of dose rates induced by the decay of nitrogen radioisotopes produced by water activation, nearby the main components (e.g. isolation valves) of both First Wall (FW) and Breeder Zone (BZ) cooling circuits.

In particular, the aim of this work is to assess the spatial distribution

of nitrogen isotopes (^{16}N and ^{17}N) in the WCLL BB cooling circuits. To this purpose, a coupled neutronic/fluid-dynamic problem is solved following a theoretical-numerical approach and adopting an integrated computational tool mainly relying on the use of MCNP6 and ANSYS CFX codes. The operative procedure adopted foresees the assessment of nitrogen isotopes production rate distribution within FW and BZ cooling channels and tubes by means of a totally heterogeneous neutronic analysis. A fully 3-D approach is, then, used to compute the nitrogen isotopes concentration within the In-Vessel complex flow domain, while a lumped parameters 1-D approach is adopted to calculate its distribution along the Ex-Vessel BB Primary Heat Transfer System.

The results obtained, herewith presented and critically discussed, provided the necessary data to perform dedicated neutronic and photonic transport analyses and, hence, to assess the dose rates in the aforementioned target locations.

Keywords: *DEMO, WCLL blanket, water activation, neutronics, CFD*

**Corresponding author:* *pierluigi.chiovaro@unipa.it*

Developments on the Tritium Extraction and Recovery System for HCPB

P1-072

Ion Cristescu [a]*, Mirela Draghia [b]

[a] *Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany*
[b] *B&C Tech Solution, Timisoara, Romania*

The tritium release from the ceramic materials of the DEMO Breading Blanket (BB) is a complex mechanism and the atomic form produced inside the ceramics structure is also converted into tritiated water at the surface of the breeder material due to interaction with residual moisture / water layer on the ceramics and free oxygen atoms after Li burn-up. Therefore, in contrast to the liquid BB where tritium is produced as Q2 solely, the Tritium Extraction and Recovery (TER) from any solid BB must handle both the molecular tritium and the tritiated water. Most of the processes considered for T extraction from the purge gas foresee to handle Q2O and Q2 separately and consecutively, making necessary a two-stage process for full tritium removal.

The recent developments on the TER system, covering the technology for handling the highly tritiated water that is produced at the level of ceramic pebbles bed, the sizing and the main design features of the key components will be introduced. The impact on the tritium recovery when considering He purge gas containing steam has been evaluated and the selection of the technologies for the full tritium recovery as tritiated water from the BB has been made. In addition, the preliminary modelling of the tritium desorption process from the cryogenic molecular sieve aiming to mitigate the gas flow rate fluctuation at the interface with the DEMO Tritium Plant will be introduced as well.

Keywords: *tritium recovery, modeling, Reactive molecular sieve beds, cryogenic trapping*

**Corresponding author:* *ion.cristescu@kit.edu*

Control System Design for the Primary Loop of CFETR WCCB

P1-073

Qinglan Cui, Yun Guo, Changhong Peng *, Zhi Chen

Department of Engineering and Applied Physics, University of Science and Technology of China, Hefei, Anhui, China

Chinese Fusion Engineering Test Reactor (CFETR) is an ITER like Tokamak fusion reactor. Water Cooled Ceramic Breeder blanket (WCCB) is one of the conceptual designs of CFETR blanket. Blanket is the key component to achieve the function of energy transfer. It is significant to analyze the thermalhydraulic properties of the blanket, and design control systems to maintain the stability of important parameter.

The object of this paper is CFETR WCCB. In the first place, the original complicated design of the blanket is simplified reasonably. Then a control system model of WCCB primary loop is developed using RELAP 5. A steady-state simulation is carried out and the results are compared with its design parameters. The WCCB primary loop includes WCCB sector blanket, pressurizer, steam generator and main pump. In the control system, the blanket power control is applied to regulate average coolant temperature, and the pressurizer is used to adjust the coolant pressure. The main signal of blanket power control is the difference of average coolant temperature and reference temperature obtained from steam generator load. A PI controller is selected in the blanket power control. To improve the response speed, a mismatch signal obtained by the difference of the blanket power and target power is introduced as the auxiliary control signal. Different load patterns are introduced to test the control system performance, it keeps the variation of average coolant temperature within 10°C when steam generator load changed by 50%, which indicates that the control system is effective.

Keywords: *CFETR, WCCB, control system*

**Corresponding author:* *pengch@ustc.edu.cn*

Design Features of the RMSB for Tritium Recovery as Tritiated Water from Helium Purge Loop of the TER HCPB

P1-074

Mirela Draghia [a]*, I. Cristescu [b], Gheorghe Pasca [c]

[a] B&C Tech Solution, Timisoara, Romania
[b] Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany
[c] I.S. TECH, Timisoara, Romania

Several technologies for the development of the Tritium Extraction and Recovery System aiming to recover tritium from the helium purge loop of the HCPB breading blanket have been assessed. The adsorption/desorption process using reactive molecular sieve beds (RMSB's) is considered one of the technologies that can be successfully used for tritium recovery as tritiated water from helium purge loop of the TER HCPB. The main advantage of using the RMSB consists in increasing the tritium activity in the water vapors at levels that usually in liquid tritiated water cannot be achieved due to the safety constrains. During regeneration the tritium is transferred, from the highly tritiated vapors adsorbed on the RMSB, by isotopic exchange to a hydrogen/deuterium stream that is further process in view of final tritium recovery.

One of the most challenging issues in the design and operation of the RMSB's is the temperature control along the bed due to the fact that water vapors shall circulate in a controlled manner. In order to maximize the isotopic exchange and to avoid releasing of the highly tritiated vapors from the bed during regeneration, a continuously desorption and adsorption shall take place. The paper aims to present the design features of a reactive molecular sieve bed that has provisions to control the adsorption and desorption process along the bed.

A proposal for the configuration of the TER HCPB based on RMSB technology together with the estimation of the amount of the required adsorbent, the sizing of the RMSB's vessels and the number of the adsorbents beds that should be used to have a suitable regeneration time will be introduced as well.

Keywords: *reactive molecular sieves beds, tritium recovery, assessment, configuration.*

Corresponding author: mirela.draghia@istech-ro.com

On the Impact of the Heat Transfer Modelling Approach on the Prediction of DEMO WCLL Breeding Blanket Thermal Performances

P1-075

Francesco Edemetti [a]*, Emanuela Martelli [b], Alessandro Del Nevo [c], Fabio Giannetti [a], Gianfranco Caruso [a], Pietro Arena [c], Ruggero Forte [d], Pietro Alessandro Di Maio [d]

[a] Department of Astronautical, Electrical and Energy Engineering, Sapienza University of Rome, Roma, Italy
[b] ENEA FSN-FUSTEC-TES, ENEA CR Frascati, Frascati, Italy
[c] ENEA FSN-ING-PAN, ENEA CR Brasimone, Camugnano, Italy
[d] University of Palermo, Palermo, Italy

The Water-Cooled Lithium-Lead (WCLL) Breeding Blanket (BB) is a key component in charge of ensuring Tritium self-sufficiency, shielding the Vacuum Vessel and removing the heat generated in the tokamak plasma. The last function is fulfilled by two independent cooling systems namely: First Wall (FW) and Breeding Zone (BZ).

Several layouts of BZ coolant system have been investigated in the last years to identify a configuration that might guarantee Eurofer temperature below the imposed limit (823 K) and good thermal-hydraulic performances (i.e. water outlet temperature equal to 601 K). In the past year, a research activity has been focused on the WCLL configurations. The main aim of this work is to compare different modelling approaches in the simulation of the heat transfer occurring within the BZ liquid metal, assessing their impact on the numerical prediction of the WCLL BB overall thermal performances. In particular, the first approach will rely on the simulation of convective and diffusive heat transfer processes taking place within the liquid metal by means of a computational thermofluid-dynamic tool based on the Finite Volume Method. Conversely, the second approach will roughly assume a pure diffusive heat transfer mechanism within the breeder, due to the very slow velocities envisaged for its flow field. In this case the heat transfer performances will be preferably assessed by means of a commercial code based on the Finite Element Method.

The analyses have been carried out modelling with both approaches the WCLL BB 2018 V0.6 equatorial unit cell. Advantages and issues from the thermal-hydraulic point of view are identified and the impact of the imposed boundary conditions and heat transfer properties, with the implemented correlations, on the respective results are critically discussed.

Keywords: *WCLL, CFD, FEM, Breeding Blanket, Blanket Engineering*

Corresponding author: Francesco.edemetti@uniroma1.it

Deuterium Permeation Behavior Through Yttria-Stabilized Zirconia Coating Fabricated by Magnetron Sputtering

P1-076

Riho Endoh [a], Kazuki Nakamura [a], Hikari Fujita [b], Moeki Matsunaga [a], Keisuke Kimura [a], Johann Riesch [c], Yoshimitsu Hishinuma [d], Takumi Chikada [a]*

[a] *Shizuoka University, Shizuoka, Japan*
[b] *The University of Tokyo, Tokyo, Japan*
[c] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[d] *National Institute for Fusion Science, Gifu, Japan*

Since hydrogen isotopes diffuse fast through metals in the operation temperature range of fusion reactors, tritium contamination in the surrounding environment and decrease in fuel efficiency are great concerns due to hydrogen isotope permeation from structural materials. In order to deal with this problem, tritium permeation barrier coatings have been investigated. Our previous studies showed that zirconium oxide (ZrO_2) coatings had an excellent performance on reduction of deuterium permeation at low temperature range, but deteriorated at 600 °C. On the other hand, it is known that yttria-stabilized zirconia (YSZ), which is produced by addition of a small amount of yttrium oxide (Y_2O_3) to ZrO_2, can suppress crack propagation because of the volume change with the phase transformation. In this study, YSZ coatings were prepared for the modification of high-temperature property, and characterization including deuterium permeation behavior in the coatings have been performed.

YSZ coatings were fabricated on reduced activation ferritic/martensitic steel F82H plate substrates by reactive magnetron sputtering without substrate heating. The thickness of the coatings was approximately 800 nm. After surface observation by scanning electron microscopy and microstructure analysis by X-ray diffraction and X-ray photoelectron spectroscopy, gas-driven deuterium permeation measurements for the coated samples were performed. Deuterium permeation through the samples was detected by a quadrupole mass spectrometer with the driving pressure of 1.00×10^4–8.00×10^4 Pa in the temperature range of 300–600 °C, and then coating characterization of the samples was conducted.

The surface of the coating was smooth, and no cracks were found. The XRD spectrum indicated the coating was oriented in the direction of <100>. In the deuterium permeation tests at 300 °C, the permeation flux was almost the same as that of the substrate and decreased at the temperature range of 400–500 °C. From the results, the primary crystal structure of the sample was not dense after fabrication without heat treatment, while crystallization proceeded with increasing test temperature. In the second series of the permeation tests at 400–600 °C, the permeation reduction factor of about 100 was reproducible, indicating no degradation of the coating occurred at 600 °C. These results prove that the YSZ coating was stable at high temperature in comparison with ZrO_2 coatings.

Keywords: *tritium, permeation, coating, yttria-stabilized zirconia*

Corresponding author: *chikada.takumi@shizuoka.ac.jp*

Remarks on the Performance of the EU DCLL Breeding Blanket Adapted to DEMO 2017

P1-077

Iván Fernández-Berceruelo *, Iole Palermo, Fernando R. Urgorri, David Rapisarda, Belit Garcinuño and Ángel Ibarra

Fusion National Laboratory, CIEMAT, Madrid, Spain

The capability of the breeding blanket to efficiently fulfill its main requirements: tritium breeding, power extraction and neutron shielding is strongly sensitive on the variation of the geometric constraints imposed by plasma physics and the integration with the rest of the plant systems.

Within the Power Plant Physics and Technology Programme, EUROfusion is developing a Dual Coolant Lithium-Lead (DCLL) breeding blanket based on the multi-module segment approach. In the current DEMO reactor specification released in 2017, the outboard blanket thickness has been decreased with the aim of providing better stability to the plasma and increasing the elongation. Besides, the first wall has been shaped in order to optimize the radiative and charged particles heat fluxes from the plasma. This work briefly describes the adaptation of the EU DCLL breeding blanket to the characteristics of DEMO 2017 and assesses the influence of the changes on its thermal-hydraulic and thermomechanical performances, mainly focusing on the outboard equatorial module.

Keywords: *DEMO, DCLL, blanket*

Corresponding author: *ivan.fernandez@ciemat.es*

Preliminary Design of the Cap Regions of DEMO Water-Cooled Lithium Lead Breeding Blanket Segments

P1-078

Ruggero Forte [a*], Pietro Arena [b], Ilenia Catanzaro [a], Alessandro Del Nevo [b], Pietro Alessandro Di Maio [a], Eugenio Vallone [a]

[a] *Department of Engineering, University of Palermo, Viale delle Scienze, Palermo, Italy*
[b] *ENEA FSN-ING-PAN, ENEA CR Brasimone, Camugnano, Italy*

Within the framework of EUROfusion R&D activity, a research campaign has been carried out in close cooperation with ENEA labs, in order to preliminary design the top and bottom cap regions of the DEMO Water-Cooled Lithium Lead (WCLL) breeding blanket segments. Due to the high thermal and mechanical loads acting on such systems, their design results particularly demanding and a specific multi-physics approach is needed, covering several aspects from neutronics, to thermalhydraulics and structural mechanics.

Preliminary CAD models of the top and bottom cap regions of the WCLL breeding blanket outboard central segment have been set-up, equipped with proper cooling circuits as well as manifolds and attachment systems. Realistic numerical models have been developed for these systems with the aim of simulating both their thermal-hydraulic and thermo-mechanical behaviour, assuming the nuclear heating distribution already assessed by dedicated neutronic and photonic analyses.

The thermal-hydraulic performances of the top and bottom cap regions have been numerically investigated to verify their cooling circuit's aptitude to effectively extract nuclear deposited heat power while complying with the prescribed requirements (outlet temperature, maximum temperature, pressure drop, etc). Finally, the thermo-mechanical behaviour of the top and bottom cap regions has been assessed in terms of stress and temperature distributions, verifying that the structural material maximum temperature stays below its prescribed value (550°C) and its integrity results ensured by the fulfilment of the RCC-MRx design rules.

Keywords: *DEMO, WCLL breeding blanket, cap, thermal-hydraulics, thermo-mechanics*

Corresponding author: *ruggero.forte@unipa.it*

Industrial-Scale Manufacturing Experience of Titanium Beryllide Block for DEMO Blanket Application

P1-079

Ramil Gaisin [a*], Vladimir Chakin [a], Pavel Vladimirov [a], Francisco Alberto Hernandez Gonzalez [a], Sergei Udartsev [b], Maxim Kolmakov [b]

[a] *Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany*
[b] *Ulba Metallurgical Plant (UMP), Ust-Kamenogorsk, Kazakhstan*

Recently, solid beryllide blocks have been proposed to be used as neutron multiplier in the updated helium-cooled breeding blanket for the DEMO fusion reactor. Intermetallic titanium beryllide ($Be_{12}Ti$) was chosen instead of pure Be as a neutron multiplier material, since it has higher working temperature, shows lower swelling and retains lower amount of tritium under irradiation. Switching from Be to $Be_{12}Ti$, logically leads to use $Be_{12}Ti$ in the form of a hexagonal block (Ø144mm×150mm) with a hole in the center. The key issue for the implementation of the new breeding blanket design is the lack of industrial technology for the production of massive $Be_{12}Ti$ blocks. This contribution describes the first results on the manufacturing of such blocks obtained in cooperation between KIT and UMP.

Be PTB-56 (UMP) and Ti powders were taken as the starting materials. After mixing, the powders were compacted by cold isostatic pressing to a density of 73% of the theoretical one. The synthesis of $Be_{12}Ti$ was performed using hot isostatic pressing (HIP) of a capsule with a compacted powder after degassing. X-ray diffraction analysis showed that HIP at 1150°C resulted in single-phase $Be_{12}Ti$ structure formation. However, the obtained density was lower than the theoretical one owing to the considerable porosity. The $Be_{12}Ti$ workpieces were ground and milled to obtain a powder. Vacuum hot pressing was performed at high temperature to consolidate the beryllide powder. Finally, the flat surfaces of the hexagon and the hole in the middle were cut using an electrical discharge machining. The advantages of $Be_{12}Ti$ blocks manufacturing according to the presented technology are discussed.

Keywords: *titanium beryllide, neutron multiplier material, hot isostatic pressing, powder metallurgy*

Corresponding author: *ramil.gaisin@kit.edu*

Electric Resistivity Behavior of Alumina Flow Channel Inserts in PbLi

P1-080

Maria González [a]*, M. Kordac [b]

[a] CIEMAT, Madrid, Spain

[b] Research Centre Řež, Husinec - Řež, Czech Republic

In order to minimize the pressure drop induced on dynamic liquid metals flowing in strong magnetic fields, the idea of inserting ceramic insulators in the PbLi flow pipes of a BB fusion is being considered. These components are known as Flow Channel Inserts. To complete the resistivity measurements already performed at CIEMAT, FCI alumina crucibles immersed in liquid PbLi at close to operational conditions were tested. The aim of the experiments was to monitor the alumina insulator's behaviour with time and temperature, using static liquid PbLi as the electrodes ensuring an intimate contact with the inner and the outer crucible walls. At the same time, the chemical compatibility between alumina and the PbLi alloy was also examined by means of the microstructural study of cut pieces after testing. Rounded and squared crucibles were supplied for the dielectric measurements at CV-Rez, Czech Republic, where several set of cooling-heating cycles between 250 and 550ºC were programmed while recording the alumina resistance.

Under 100 V polarization conditions and temperatures above 500ºC, excellent insulating properties of alumina crucibles have not deteriorated in contact with static PbLi, the calculated resistivity value being comparable with that obtained in vacuum. Thermal and electron irradiation treatments previously applied to crucibles do not either influence the alumina behavior under this experimental conditions. Furthermore, the PbLi in contact with the alumina crucibles has not damaged the alumina microstructure, which is found unaltered as concluded from SEM and EDX analytical results.

Keywords: *alumina, FCI, PbLi, electrical resistivity, chemical compatibility*

Corresponding author: *maria.gonzalez@ciemat.es*

Evaluation on Electromagnetic Analysis of Cylindrical WCCB TBM

P1-081

Wenhai Guan *, Hyoseong Gwon, Hisashi Tanigawa, Takanori Hirose, Yoshinori Kawamura

National Institutes for Quantum and Radiological Science and Technology, Rokkasho, Aomori, Japan

Water-cooled ceramic breeder (WCCB) Test blanket module (TBM) fabricated by reduced activation ferritic/martensitic steel of F82H is being developed by National Institutes for Quantum and Radiological Science and Technology in Japan. The design of TBM container has been changed from a box-shaped container to a cylindrical because of its advantage of pressure resistance in case of water ingress into the container. The structural integrity of the TBM-set consisting of TBM, shield and shield case under the electromagnetic (EM) loads induced by eddy current during plasma disruption should be taken into accounted. This study aims to evaluate the structural integrity of the cylindrical WCCB TBM-set suffered from different types of EM loads.

A 20 degrees symmetrical finite element (FE) model provided by ITER organization was adopted to analyze EM field. The model of TBM-set with its surrounding void region mesh was created by WCCB TBM team and installed into the FE model. Three types of applicable disruption events calculated by DINA as an input current data was loaded. A transient nonlinear analysis with BH curve of F82H was applied for the analyses. Summarized EM force including Lorentz and Maxwell forces were used for mechanical analysis to investigate the stress distribution and structural integrity. In the time variation of Lorentz force under the most critical disruption event, the maximum value along the radial direction of cylindrical TBM was observed. The maximum Lorentz force was almost 50% lower than box-shaped TBM, which was caused by the reduction of weight of the structural material compared with box-shaped container. The effect of Maxwell force, BH curve and stress distribution of support key with all expected disruption events will be presented.

Keywords: *Plasma disruption, Structural integrity, BH curve, Lorentz force, Maxwell force*

Corresponding author: *wenhai.guan@qst.go.jp*

Influence of Heat Transfer on Tritium Transport in Liquid Metal Blankets at High Hartmann Number

P1-082

Han Jiajia *

Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China

Compared with solid breeder materials, liquid breeder materials have many advantages. And liquid metal (LM) blanket is a topic of great interest in the fusion reactor blanket design. However, the flow of the LM will cause complex magnetohydrodynamic (MHD) effect under the strong magnetic field in the blanket. The dynamic viscosity of LM is impacted by heat transfer, resulting in the change of velocity distribution and tritium concentration distribution. The influence of heat transfer on tritium transport in a rectangular duct at high Hartmann number is investigated by a coupling method. By this method, the velocity field is calculated through a second-order projection method, coupled with the temperature distribution and tritium concentration distribution computed by a finite-volume method. The numerical result without temperature influence is validated by Hunt's and Shercliff's analytical solutions, which shows very good accuracy. On the basis of this numerical code, the tritium concentration distribution is simulated. The simulation result indicates that heat transfer has a significant effect on the concentration distribution of tritium.

Keywords: *Tritium transport, magnetohydrodynamic (MHD), finite-volume methods, velocity distribution*

Corresponding author: *hjaza_f@163.com*

Consolidated Design of the HCPB Breeding Blanket for the Pre-Conceptual Design Activities of the EU DEMO and Harmonization with the ITER HCPB TBM Program

P1-083

Francisco A. Hernández [a]*, Pavel Pereslavtsev [a], Guangming Zhou [a], Salvatore D'Amico [a], Heiko Neuberger [a], Jörg Rey [a], Qinlan Kang [a], Evaldas Bubelis [a], Lorenzo V. Boccaccini [a], Béla Kiss [b], Ivo Moscato [c], Gábor Nádasi [d], Fabio Cismondi [e], Yves Poitevin [f]

[a] *Karlsruhe Institute of Technology, Karlsruhe, Germany*
[b] *Budapest University of Technology and Economics, Budapest, Hungary*
[c] *Università degli Studi di Palermo, Palermo, Italy*
[d] *Wigner Research Center for Physics, Budapest, Hungary*
[e] *EUROfusion Programe Management Unit, Garching, Germany*
[f] *Fusion for Energy, Barcelona, Spain*

From the period 2014-2020, the pre-Conceptual Design Activities (pre-CDA) of the EU DEMO have taken place. These pre-CDA differ from past exercises in their strong Systems Engineering methodology, as well as for the pragmatic approach in their technology choices. The Helium Cooled Pebble Bed (HCPB) is one of the 2 candidates as driver blanket for the EU DEMO in the pre-CDA. Several design iterations have been required during the pre-CDA to adjust the design to the demanding DEMO requirements, to the very challenging systems integration and to the need to keep near-term technologies. To this respect, the design has evolved to a so-called fuel-breeder pin architecture built in single-module segments. The pins are filled with a pebble bed of a ceramic breeder mixture of Li_4SiO_4 + 35mol % Li_2TiO_3 and are embedded in prismatic blocks of $Be_{12}Ti$ acting as neutron multiplier. He gas at 8 MPa is used as coolant with a temperature window of 300-520 °C. This architecture has proven to achieve a large tritium breeding performance (≈1.20), a remarkably low plant circulating power (<100 MW) and its design for manufacturing paves the way for a better industrialization of its components and functional materials. This paper describes the consolidated design of the HCPB for the pre-CDA, shows its main performance figures and presents the ongoing realignment of the DEMO pre-CDA HCPB with the ITER HCPB TBM program.

Keywords: *EU DEMO, HCPB, TBR, breeding blanket*

Corresponding author: *francisco.hernandez@kit.edu*

Survey on Lithium-6 Availability and Enrichment Strategies

P1-084

Julia M. Heuser [a*], Regina Knitter [a], Sehila M. Gonzalez de Vicente [b]

[a] *Institute for Applied Materials (IAM), Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany*
[b] *Department of Nuclear Sciences and Applications, International Atomic Energy Agency (IAEA), Vienna, Austria*

For future fusion reactors, tritium and deuterium are intended to be used as fuel. While deuterium is abundant in sea water and can be implemented in the fusion reaction directly, tritium has to be bred from lithium. Therefore, a sufficient supply of lithium will be of crucial importance for all tritium breeding materials. For fusion technology materials enriched with the lighter isotope, Li-6, are required to produce tritium by an exothermic reaction with thermal neutrons. Li-6 enriched breeding material is essential to ensure an adequate tritium breeding ratio of slightly above one for a self-sustained fusion reaction. Since natural lithium contains only 7.6 % Li-6 beside 92.4 % Li-7, efficient Li-6 enrichment and procurement strategies are needed.

In this survey, the availability and need of Li-6 for future fusion technology is discussed focusing on the EU solid breeder concept and the possible usage of advanced ceramic breeder materials consisting of Li_4SiO_4 with additions of Li_2TiO_3 in helium cooled pebble beds (HCPB). Furthermore, selected Li-6 enrichment strategies will be described. Considering the advantages and disadvantages, the different methods are compared and evaluated with respect to fusion technology.

This study will highlight an often neglected topic, which must be faced for future fusion technology.

Keywords: *Li-6 availability, Li-6 enrichment, advanced ceramic breeder*

Corresponding author: *julia.heuser@kit.edu*

Preparation of Irradiation Rig for Test of Al_2O_3 Coatings in Contact with Liquid PbLi

P1-085

Karel Samec, M. Iarmonov, P. Matouš, T. Melichar, L. Vála, M. Kordač

Centrum výzkumu Rez, Husinec-Řež, Czech Republic

DEMO blanket concepts reliant on liquid metal, PbLi, as breeding material require the resolution of several issues. Among the many challenges to be solved feature most prominently structural material corrosion by liquid metal and tritium losses due to permeation of the structural materials. A possible solution to these matters is to develop anti-permeation and anti-corrosion barriers in the form of coatings. Alumina coatings are one of the potential candidate materials, whose performance under model conditions is being tested in several institutions. These tests show promising behaviour if the various key parameters are isolated: corrosion, hydrogen permeation, hydrogen permeation under ion-irradiation, however as yet no data exists for combinations of these parameters. This is a blind area of the data set as the key parameters with will occur concurrently in practice. The aim of the task presented, is therefore to perform a combined test of the coatings performance in the LVR15 reactor – i.e. tritium permeation under neutron irradiation through barriers, which will be in contact with PbLi during the irradiation test. The current contribution shows progress in the irradiation test rig preparation and its critical subsystem performance in out-of pile conditions without irradiation.

A dedicated testing apparatus has been produced to demonstrate the reliability of safely operating a liquid metal PbLi-filled capsule, which is cooled by flowing water without exceeding critical design parameters of the components (e.g. maximum steel temperature for structural strength). A gas-filled thermal gap concept for regulating inner temperature was operated under variable Nitrogen/Helium gas composition. Cooling performance both at steady state and during simulated decay heat changes was recorded. The limitations of the cooling systems were studied and the experimental feedback was gainfully used to modify the design of the liquid metal irradiation capsule to make it more suitable for the test of the Al_2O_3 coatings. Overall, the out-of-pile thermal tests have demonstrated the feasibility of an irradiation rig by demonstrating the operation of a PbLi-filled capsule with simulated gamma heating of 5 kW being cooled by water flowing at 0.5 m/s at a bulk temperature of 50°C. During the experiments, the PbLi temperature was regulated with temperatures between 300 and 500°C obtained by changing the parameters of the thermal gap.

Keywords: *PbLi, neutron irradiation, thermal gap, reactor test rig*

Corresponding author: *Mikhail.Iarmonov@cvrez.cz*

Effect of Post Weld Heat Treatment on Mechanical Properties of Gas Tungsten Arc Welded 316L(N)-IG (X2CrNiMo 17-12-2) Stainless Steel

P1-086

Sawoong Kim *, Byungil Park, Heejin Shim, Yonghee Lee

ITER Korea Domestic Agency, National Fusion Research Institute, Daejeon, Korea

The International Thermonuclear Experimental Reactor (ITER) Blanket Shield Block (BSB) shall be required to accommodate the interfaces with tight tolerances to all the components located in the Vacuum Vessel such as the First Wall (FW), In-Vessel Coils, Diagnostics and so on. The ITER require mandatory performance of stress relieving after welding of the cover plates on the SB to ensure a dimensional stability and mechanical properties. Post Weld Heat Treatment (PWHT) is generally performed to reduce the residual stress and to recover the microstructural change. Rates of heating, holding times and temperatures, and rates of cooling are all importance variables that need to be controlled and monitored precisely, or the desired effects may not be achieved. The variable conditions in PWHT is specified as per RCC-MR, Sect. 5, RF-8410 "Dimensional Stabilization Treatment on Austenitic Stainless Steel Components".

In order to investigate the effect of PWHT, the mechanical properties were evaluated with manually Gas Tungsten Arc Welded joints of 316L(N)-IG stainless steel before/after PWHT, and reviewed the relationship between variable conditions in PWHT and mechanical properties. Hardness of weldment after PWHT decreased with increasing the temperature and holding time. Scanning Electron Microscope (SEM) was used to characterize the microstructure and fracture modes.

Keywords: *Shield Block, Tolerance, Weld, Post Weld Heat Treatment, Mechanical Properties*

****Corresponding author:*** *swkim12@nfri.re.kr*

Numerical Simulation of 3D Magnetohydrodynamic Liquid Metal Flow in a Spatially Varying Solenoidal Magnetic Field

P1-087

Viktor Klüber *, Chiara Mistrangelo, Leo Bühler

Karlsruhe Institute of Technology, Karlsruhe, Germany

The flow of lead lithium (PbLi) in liquid metal breeding blankets or manifold systems is subject to strong magnetic fields, which induce currents in the electrically conducting fluid. These currents in turn interact with the magnetic field, which leads to Lorentz forces, resulting in dominant magnetohydrodynamic (MHD) effects such as high pressure drop and modified velocity profiles compared to hydrodynamic flows. In the present work, the 3D flow of an electrically conducting fluid entering a strong magnetic field is investigated by numerical simulations using a code based on the OpenFOAM® library. The use of unstructured grids for resolving complex fluid domains and walls requires integrated correction schemes to guarantee the quality of the numerical solution, especially for strong magnetic fields. The considered MHD flows in non-uniform magnetic fields are relevant for the piping system in fusion reactors, where the fluid enters regions of the strong plasmaconfining magnetic field before it is distributed to the blanket modules. For this type of flow, also experimental data is available for code validation. While previous simulations for MHD flows in non-uniform magnetic fields used the simpler but physically incomplete single-component representation of the magnetic field to compare results with the ALEX-experiment [1], the present work considers a realistic solenoidal multi-component magnetic field as measured in experiments in the MEKKA facility [2]. A comparison of numerical and experimental results is presented.

[1] B. F. Picologlou and C. B. Reed, "Experimental investigation of 3-D MHD flows at high Hartmann number and interaction parameter," in Liquid Metal Magnetohydrodynamics, J. Lielpetris and R. Moreau, Eds., Dordrecht, Kluwer, 1989, pp. 71-77.

[2] L. Bühler, H.-J. Brinkmann and C. Mistrangelo, "Experimental investigation of liquid metal pipe flow in a strong non-uniform magnetic field," in Proceedings of the 11th International PAMIR Conference - Fundamental and Applied MHD, July 01 - 05, 2019, Reims, France, 2019.

Keywords: *Liquid metals, numerical MHD, code validation, breeding blankets*

****Corresponding author:*** *viktor.klueber@kit.edu*

Experimental Validation of Tritium Recovery System from Liquid LiPb by Vacuum Sieve Tray Concept

P1-088

Satoshi Konishi [a]*, Fumito Okino [a], Juro Yagi [a], Keisuke Mukai [a], Teruya Tanaka [b], Akio Sagara [b]

[a] *Institute of Advanced Energy, Kyoto University, Uji Kyoto, Japan*
[b] *National Institute of Fusion Science, Japan*

Liquid Lithium Lead is a major candidate of breeder for near term fusion reactor such as ITER/TBM and DEMO, as well as expected to advance for higher temperature applications upto 900 degree C. Several tritium recovery methods are considered, and the authors have developed the Vacuum Sieve Tray (VST) concept where tritium dissolved in the LiPb breeder is extracted from falling droplets from nozzles under vacuum. Based on the earlier studies that proved enhanced tritium transport in liquid at two orders of magnitude faster than the diffusion in static liquid, tritium extraction from falling droplets in VST device provides extremely efficient recovery with minimal mechanical and material constraints. The authors have also developed an innovative technology to spray liquid metal in vacuum with spiral nozzle that forms a number of fine droplets with simple mechanical configuration with reduced risk of clogging.

As the next step to demonstrate this tritium recovery technology, mockup device is manufactured and installed in the liquid metal loop Oroshhi-2 at the NIFS facility. Typical operation conditions are: temperature 375-450 degree C, droplet speed 1.5 m/s – 3.0 m/s, diameter and number of nozzles 0.6mm, 1-19, and vacuum pressure 10^{-3} Pa. Liquid LiPb is equilibrated with D_2 gas in the loop, and its extraction performance is evaluated in steady state operation upto 24 hours from the difference of concentration at the inlet and outlet of the VST device. In a preliminary test, formation of droplets with prescribed radii was confirmed with satisfactory observation and diagnostics of droplets within the chamber. The VST device was then attached to the Oroshhi-2 loop in March 2019 and will demonstrate the steady state operation to show the feasibility of the process to efficiently recover pure bred tritium from LiPb stream continuously at engineering scale in 2019. The result is expected to provide a realistic technical solution for fusion blankets and fuel cycle systems.

Keywords: *tritium, blanket, liquid metal, vacuum sieve tray, fuel cycle*

Corresponding author: *s-konishi@iae.kyoto-u.ac.jp*

Modeling of PbLi Purification from Volatile Contaminants

P1-089

Michal Kordač *

Centrum Výzkumu Rez, Husinec-Řež, Czech Republic

DEMO blanket concepts reliant on liquid metal, PbLi, as breeding material require the resolution of several issues. Among the many challenges to be solved feature most prominently occurrence of nuclear reactions products with various degree of volatility, namely He, Hg and Po. Accumulation of these product would cause issues related to safety – mainly Po – or even significantly influence the performance of the breeding blanket by formation of gas pockets over time – He. A possible solution to these matters is to develop a purification technique, which will allow selective removal of the contaminants. Based on the fact, that the contaminants in question show high volatility, evaporation from a free surface seems to be a feasible technique. To intensify the process, a gas-liquid contactor may be used either in a form of a sieve tray column, in which liquid droplets are in free fall.

A dedicated testing apparatus consisting of a sieve tray column with circulating liquid PbLi, which allows following the process of species evaporation has been produced to demonstrate the feasibility of the concept and to obtain engineering data for scaling up. In preparation of the evaluation of the experimental campaign, a mathematical model of the process is developed to study its parametric sensitivity. The model is based on unsteady diffusion inside of a spherical droplet including the gas-phase resistance, which limits the evaporation rate during the initial phase of the fall. The effect of using multiple stages inside a single column is shown in comparison to a single stage process inside the same geometry. The effect of a sweep gas on overall purification efficiency is examined.

Keywords: *vacuum degassing, PbLi purification*

Corresponding author: *Michal.Kordac@cvrez.cz*

Qualification of Al-Based Coatings Processed by ECX-Process for Application as Corrosion and T-Permeation Barriers

P1-090

Wolfgang Krauss [a]*, Sven-Erik Wulf [a], Nils Holstein [a], Teresa Hernández [b]

[a] *Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany*
[b] *CIEMAT National Laboratory for Magnetic Fusion, Madrid, Spain*

Low activation ferritic martensitic steels (RAFM) such as Eurofer are considered for application in breeding blankets in future fusion reactors. In some of the designed blanket concepts the steels will be in direct contact with flowing Pb-16Li, which acts as breeding material and partly as coolant medium, at demanding operation temperatures of around 550°C. Under these conditions the structural material suffers from strong corrosion attack and corrosion rates of up to 400 µm per year were reported depending on flow rate and temperature. The corrosion products will be transported with the flowing breeder and will be precipitated at cooler loop sections with the risk of line plugging by precipitates. To overcome these problems aluminum-based coatings on RAFM steels are considered as barriers to reduce/prevent corrosion of the structural material. On the other hand these barriers are envisaged to provide a tritium permeation reduction into the steel structures, too.

The presentation will review the coating processes which were developed in the past at KIT and will highlight the progress achieved on aluminum-based coatings manufactured by the development of the ECX process. By this process, the coatings are made via controlled electrodeposition of thin pure aluminum layers from an ionic liquid on RAFM steels. Subsequently, the pure aluminum layers are transformed into protective Fe-Al with superficial Al_2O_3 via a special multi-step oxygen-controlled heat treatment procedure. Their corrosion resistance was analyzed in flowing Pb-16Li in PICOLO loop of KIT. After optimization ECX coatings survived harsh testing conditions (550°C, 0.1 m/s) for more than 12,000 h without visible attack. The analyses included diameter and surface investigations by SEM/EDX before and after exposure to Pb-16Li. ECX coatings were also deposited on Eurofer discs for analyzing the issue reduction of tritium permeation. First results for one side coated discs showed a permeation reduction factor (TRF) in the range of 100. Both tested properties of the ECX coatings, the corrosion resistance and the TRF, are promising for fusion application.

Keywords: *Al-based barriers, Corrosion, Permeation, Blanket, Liquid breeder Pb-16Li*

Corresponding author: *wolfgang.krauss@kit.edu*

Performance Test of the Key Components for the HCCR Breeding Blanket Cooling System and the Validation of the GAMMA-FR Code

P1-091

Eo Hwak Lee [a]*, Suk-Kwon Kim [a], Hyung Gon Jin [a], Chang Wook Shin[a], Dong Won Lee [a], Seungyon Cho [b]

[a] *Korea Atomic Energy Research Institute, Daejeon, Korea*
[b] *National Fusion Research Institute, Daejeon, Korea*

A helium cooling system (HCS) is applied to remove the heat from the helium cooled ceramic reflector (HCCR) breeding blanket. To test the performance of the key components (helium circulator, printed-circuit heat exchange (PCHE) type recuperator/cooler, and helium heater) of the HCS and to use as experimental equipment for the validation of the safety analysis code (GAMMA-FR), helium supplying system (HeSS) was constructed at KAERI. Recently helium circulator was upgraded for better performance and better reliability and the independent mechanical running test was successfully performed up to 8 MPa of helium. Then, it was installed in HeSS and the performance of the key components was tested under normal operation conditions. The results were analyzed for developing and validating the components model in the GAMMA-FR code.

Keywords: *HCCR, breeding blanket, HCS, fusion power plant, GAMMA-FR, Helium circulator, HeSS*

Corresponding author: *ehl@kaeri.re.kr*

Numerical Investigation of Purge Gas Flow Through Binary-Sized Pebble Beds Using Discrete Element Method and Computational fluid Dynamics

P1-092

Youngmin Lee [a], Dongwoo Sohn [b]*, Mu-Young Ahn [a], Yi-Hyun Park [a], Seungyon Cho [a]

[a] *National Fusion Research Institute, Daejeon, Korea*
[b] *Department of Mechanical Engineering, Korea Maritime & Ocean University, Busan, Korea*

The main function of the breeder materials, such as Li_2TiO_3, Li_4SiO_4, and Li_2ZrO_3, in a solid type breeding blanket is to produce tritium, which is transferred to the fuel cycle system by helium purge gas and is used as a fuel for nuclear fusion reactions. Since the configuration of the breeder should be designed considering the purge gas flow, the form of pebble beds is mainly adopted, thereby leading to the smooth flow inside a blanket. The flow characteristics can be important design drivers because they also affect the design of the fuel cycle system. In order to increase a tritium production rate, a large amount of breeder should be placed in the same space and hence various sizes of pebbles instead of one size of pebbles can be applied to increase the packing fraction of the pebble beds. However, in this case, since the flow resistance may increase, the relationship between the pebble bed configuration and the purge gas flow should be carefully investigated. In this study, the characteristics of purge gas flow through binary-sized pebble beds are investigated in terms of the pressure drop and velocity variance in the breeding zone using discrete element method (DEM) and computational fluid dynamics (CFD).

Keywords: *Binary-sized pebble bed, Purge gas flow, Discrete element method (DEM), Computational fluid dynamics (CFD)*

Corresponding author: *dsohn@kmou.ac.kr*

Induced Jet Break-Up Fabrication of Advanced Ceramic Breeder Pebbles

P1-093

Oliver Leys [a]*, Patrick Waibel [b], Jörg Matthes [b], Regina Knitter [a]

[a] *Institute for Applied Materials (IAM), Karlsruhe Institute of Technology (KIT), Germany*
[b] *Institute for Automation and Applied Informatics (IAI), Karlsruhe Institute of Technology (KIT), Germany*

In order for future fusion reactors to be self-sufficient, it is imperative that the fuel component tritium is generated onsite. It is foreseen to feature lithium rich ceramic pebbles in the form of pebble beds in the wall of the reactor and upon irradiation, the lithium will decay into helium and tritium. The tritium is then to be extracted, processed and rerouted into the plasma to be used as a fuel for the fusion reaction. At the Karlsruhe Institute of Technology (KIT), a melt-based process is used for the fabrication of advanced ceramic tritium breeding pebbles. The synthesis powders are heated in a platinum alloy crucible up to approximately 1400 °C to form a melt. A pressure is then applied to the crucible to force the melt through a 300 μm nozzle on the underside of the crucible. As the melt leaves the nozzle, a laminar jet is formed which subsequently breaks up into droplets, which are solidified in a cooling tower to form the pebbles.

One of the most important steps of the process is the break-up of the jet into droplets. It is during this step that both the yield and the pebble size distribution of the product are predominantly determined. In general, random ambient disturbances cause instabilities to grow on the surface of the jet until the surface tension forces overcome the viscous forces and a droplet breaks off. Theory states that every laminar jet has an intrinsic optimum instability wavelength, which will grow fastest, thereby supressing the ambient disturbances and resulting in a monodisperse jet break-up. In order to produce the desired wavelengths on the jet, a frequency generator was developed which can apply desired frequencies to the process system. By maintaining a constant flowrate of the jet, the wavelengths were then controlled by applying predetermined frequencies. Optimum operating conditions were found for the process which led to an increased yield, as well as a higher degree of monodispersity.

Keywords: *ceramic breeders, lithium orthosilicate, lithium metatitanate, fabrication, jet break-up*

Presenting author: *oliver.leys@kit.edu*

Li_4SiO_4 Pebbles New Fabrication Process: Feasibility and Preliminary Experimental Data

P1-094

Rosa Lo Frano *, Monica Puccini *, Eleonora Stefanelli

University of Pisa DICI, Pisa, Italy

Open issue for fusion reactor technology is related to the breeder material to be used for nuclear fusion reaction. Despite in the last decades several studies were carried out, the selection, characterization and reliability of breeder material are still open issues.

In this study, the attention is focused on lithium orthosilicate (Li_4SiO_4), one of the most important candidate for TBM system. In particular, the development of a new Li_4SiO_4 fabrication method based on the drip casting at room temperature is presented and compared to the other studied/suggested methods.

To support the feasibility of this method, an experimental device was designed and constructed at Pisa University, Department of Civil and Industrial Engineering (DICI). The design constraints for such device were dictates by the requirements that a breeder material must satisfy, like the high lithium density, dimensions, stability in the blanket environment, etc. The pebbles were produced starting from a suspension of Li_4SiO_4 precursors prepared by sol-gel method, which was dripped through the experimental device and coagulated into solid spheres in a reacting solution. Li_4SiO_4 pebbles were finally obtained after sintering at high temperature.

The obtained pebbles have been analysed and the preliminary results such as density, microstructure and crystal form (by SEM and X-ray diffractometer), and morphology are presented. These preliminary results confirmed the feasibility of dripping at room temperature and provided further useful information for the optimization of the process.

Keywords: *Breeder blanket, Li4SiO4, Pebbles, Experimental, Drip Casting*

Corresponding author: *rosa.lofrano@ing.unipi.it; monica.puccini@unipi.it*

Benchmark of Serpent-2 with MCNP: Application to EU Fusion DEMO HCPB Breeding Blanket

P1-095

Yudong Lu [a,b], Guangming Zhou [b]*, Francisco A. Hernández [b], Pavel Pereslavtsev [b], Jaakko Leppänen [c], Minyou Ye [a]*

[a] *University of Science and Technology of China, Hefei, China*
[b] *Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany*
[c] *VTT Technical Research Centre of Finland, Espoo, Finland*

Fusion reactors such as the European fusion demonstration reactor (EU DEMO) have a very complex geometry, which makes the modelling of the nuclear analyses rather tedious and time consuming. The Monte Carlo neutron transport code Serpent-2 [1] developed by VTT Finland, able to directly import CAD geometry, is then considered as an attractive alternative to the Monte Carlo N-Particle Transport code (MCNP) for its application in fusion reactors.

As a first step, a comparison activity is carried out to benchmark the Serpent-2 with MCNP calculations. Both Serpent-2 and MCNP are used to perform the 3D Monte Carlo particle transport making use of the same geometry. The Helium Cooled Pebble Bed (HCPB) breeding blanket is a promising option for the EU DEMO driver blanket. The nuclear performance of EU DEMO HCPB such as tritium breeding ratio (TBR), neutron wall loading (NWL), nuclear heating are analyzed and the corresponding results are compared between Serpent-2 and MCNP. It was found that Serpent-2 is suitable for the application of nuclear analysis for the EU DEMO HCPB breeding blanket.

[1] Leppänen, J., et al. (2015) "The Serpent Monte Carlo code: Status, development and applications in 2013." Ann. Nucl. Energy, 82 (2015) 142-150.

Keywords: *Neutronics, Fusion, Monte Carlo, Serpent*

***Corresponding author:** guangming.zhou@kit.edu; yemy@ustc.edu.cn*

Update of Electromagnetic Loads on HCPB Breeding Blanked for DEMO 2017 Configuration

P1-096

Ivan Alessio Maione [a], Massimo Roccella [b], Francisco Hernández [a], Flavio Lucca [b]

[a] *Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany*
[b] *LT Calcoli srl, Merate, Italy*

The researches carried out over the past years inside the Breeding Blanket (BB) project have demonstrated the importance of considering electromagnetic (EM) loads for the structural assessment of BB segments and the design of some interface components (e.g. attachment system with VV). These loads arise from the interaction of the ferromagnetic/ conductive structure with the currents and the magnetic field generated inside the tokamak and act on the BB structure during both normal and off-normal operations. In particular, their distribution and magnitude strongly depends on the tokamak magnetic configuration, BB segment design and its electrical connections with the other conductive structures, as well as on the considered plasma scenario.

The last available EM analyses have been performed on the DEMO2015 baseline characterized, for what concern the EM point of view, by 18 sectors, a toroidal field B_T of 5.67 T at R_0 = 9.07 m, and an operational plasma current I_p of 19.6 MA. The definition of the current DEMO2017 baseline (with 16 sectors, B_T = 4.89 T at R_0 = 8.94 m, and I_p = 19.1 MA), coupled with a reduction of 30 cm in the BB outboard space reservation and the improvement on the BB segment design, has brought to the necessity to a new evaluation of the EM loads that, consistently, take into account these modifications. The work here presented reports the EM analysis for the Helium Cooled Pebble Bed (HCPB) BB. EM loads for the 2 inboard (IB) and 3 outboard (OB) segments have been calculated during a nominal plasma flat-top scenario (no induced currents in the structure) and a plasma vertical disruption (VDE) with a current quench time of 74 ms. The results are compared with the previous ones and the effect of the introduced modifications respect to the EM loads are discussed.

Keywords: *EU DEMO, Breeding Blanket, HCPB, electromagnetic loads, vertical plasma disruption*

***Corresponding author:** ivan.maione@kit.edu*

Three Dimensional Magneto Convective Flows in Geometries Relevant for DCLL Blankets

P1-097

Chiara Mistrangelo *, Leo Bühler, Viktor Klüber

Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany

The dual coolant lead lithium (DCLL) blanket is one of the liquid metal blanket concepts investigated within the framework of the EUROfusion Consortium. In the proposed design, the liquid metal serves as tritium breeder, neutron multiplier, and as coolant for the breeding zone. Reduced activation ferritic steel is used as structural material and helium is employed to cool first wall and blanket structure. Magnetohydrodynamic (MHD) effects deriving from the interaction of the moving liquid metal with the magnetic field that confines the fusion plasma are coupled with thermal phenomena. The induced electromagnetic forces lead to significant modifications of the hydrodynamic flow and affect in turn heat and mass transfer. The presence of a high-intensity neutron flux in the liquid metal yields volumetric heating and associated buoyant forces that drive complex flow patterns in the blanket. The resulting convective motions cause strong thermal mixing, and an increment in the effective heat transfer coefficient that may increase heat losses from the liquid metal into the helium coolant. This could deteriorate the thermal efficiency of the blanket. Buoyant MHD phenomena may also result in reversed flow and closed recirculations where tritium accumulates and temperature locally increases. The aim of the present investigation is to address by means of 3D numerical calculations the main features of magneto-convective flows in a geometry typical for DCLL blankets. The PbLi moves upwards in a poloidal duct that faces the first wall and, after making a 180° turn, flows downwards at the back of the blanket module. The liquid metal enters and leaves the model geometry by flowing radially in rectangular channels. A parametric study has been carried out to approach conditions relevant for fusion blanket operation.

Keywords: *Liquid metals, DCLL blanket, magnetohydrodynamics (MHD), magnetoconvection*

***Corresponding author:** chiara.mistrangelo@kit.edu*

Effect of Dragged Magnetic Field Lines into RAFM Steel Blanket Modules on First Wall Heat Load

P1-098

Yuya Miyoshi *, Ryoji Hiwatari, Youji Someya, Nobuyuki Asakura, Shinsuke Tokunaga, Yuki Hommma, Yoshiteru Sakamoto, Kenji Tobita, The Joint Special Design Team for Fusion DEMO

National Institutes for Quantum and Radiological Science and Technology, Aomori, Japan

Plasma heat flux in peripheral plasma reaches first wall (FW) along a magnetic field line, and it causes heat load. This heat flux is several MW/m^2 orders of magnitude, and most part of FW, magnetic field line incident angle θ is small ($sin\theta$ *is* 10^{-3} order of magnitude) and heat load value becomes small. At several regions such as blanket module leading edge, however, incident angle increases and heat load will become higher than module cooling performance (nearly $1MW/m^2$). In JA DEMO, blanket module is made of reduced-activation ferritic martensitic (RAFM) steel. This material is magnetic material and changes magnetic field. It is concerned that near RAFM steel, magnetic field line is dragged into each module and unexpected high heat load occurs. On the other hand, in out preliminary calculation, some magnetic field lines are dragged on module front surface before it reaches module leading edge and heat load is reduced. To design the blanket module shape, detailed assessment of the effect of such error field on heat load is necessary.

In our past research, the new analysis model APPLE model is introduced. This model traces magnetic field lines and divides peripheral plasma into Apple Peel Like Elements (APPLE). Calculating heat flux in each APPLE, and touched area between APPLE and FW, heat load profile on FW can be assessed.

In this conference, error field caused by RAFM steel is calculated considering the real arrangement of blanket modules and NBI ports. Using APPLE model, the plasma heat flux dragged on the blanket modules and heat load are calculated. From the calculation results, the design issues caused by such error field will be presented.

Keywords: *fusion reactor design, plasma heat load, JA DEMO*

***Corresponding author:** miyoshi.yuya@qst.go.jp*

Computational Search for Superionic Conductors for Efficient Lithium Isotope Separation with the Electrodialysis Technique

P1-099

Kenji Morita *, Tsuyoshi Hoshino

Rokkasyo Fushion Institute, National Institute for Quantum and Radiological Science and Technology (QST), Aomori, Japan

The tritium needed as a fuel for fusion reactors is produced by the neutron capture reaction of lithium-6 (^6Li), enrichment of ^6Li up to 90% is required for adequate tritium breeding in many fusion reactor concepts. Various methods such as amalgam, electromigration and chromatography were researched for lithium isotope separation in the past. Recently, electrodialysis of lithium-containing solutions using ionic conductors as a separation membrane has been proposed by the one of the authors (T.H.).

For an efficient separation with the new technology, the ionic conductors must fulfill several requirements such as high lithium ion conductivity which accelerates the concentration process and a conduction mechanism maximizing the difference between ^6Li and ^7Li. Because characteristic structural properties of such materials have not been identified yet, we explore existing materials on the basis of Inorganic Crystal Structure Database. We employ VASP code which is widely used first-principles calculation package to compute the ionic conductivity and the separation efficiency.

In this presentation, we report on the properties of several superionic lithium conductors and discuss the relevant conduction mechanism to the efficient isotope separation. Implications for possible new materials for the separation deduced from computational perspectives will also be discussed.

Keywords: *Lithium Isotope separation, Ionic conductors, First-principles calculations*

***Corresponding author:** morita.kenji@qst.go.jp*

Nuclear Analysis of the Water Cooled Lithium Lead DEMO Reactor

P1-100

Fabio Moro [a]*, A. Colangeli [a,b], A. Del Nevo [c], D. Flammini [a], G. Mariano [d], E. Martelli [a], R. Mozzillo [e], S. Noce [f], R. Villari [a]

[a] *ENEA, Department of Fusion and Nuclear Safety Technology, Frascati, Rome, Italy*
[b] *SBAI Department, Sapienza University of Rome, Rome, Italy*
[c] *ENEA, Department of Fusion and Nuclear Safety Technology, Camugnano, Italy*
[d] *DIAEE Department, Sapienza University of Rome, Roma, Italy*
[e] *CREATE, University of Naples Federico II, Naples, Italy*
[f] *University of Rome Tor Vergata, Industrial Engineering Department, Rome, Italy*

In the frame of the EUROfusion roadmap, the development of a conceptual design for the Demonstration Fusion Power Reactor (DEMO), beyond ITER, is a key issue. The DEMO reactor shall guarantee the tritium self-sufficiency, generate electricity and operate as a test facility for the fusion power plant relevant technologies, such as the breeding blanket (BB). The Water Cooled Lithium Lead (WCLL) concept has been chosen as a candidate for the DEMO BB: it relies on liquid Lithium Lead as breeder and neutron multiplier, Eurofer as structural material and pressurized water as coolant.

A detailed MCNP model of the latest WCLL BB layout has been generated and integrated in the DEMO MCNP generic model suitably designed for neutronic analyses. Three-dimensional neutron and gamma transport simulations have been carried out by means the MCNP Monte Carlo code and JEFF nuclear data libraries: the main topics investigated in the frame of this analysis is the evaluation of the WCLL-DEMO performances in terms of tritium self-sufficiency and shielding effectiveness to protect the vacuum vessel and the toroidal field coils. Moreover, the radial profiles of the neutron flux, nuclear heating, neutron damage and he-production are provided in the Inboard and Outboard equatorial planes.

The outcomes of the present study provide guidelines for the optimization of the WCLL DEMO reactor nuclear performances through the assessment of the loads on sensitive components and the estimation of its potential tritium generation capabilities.

Keywords: *DEMO, WCLL, Breeding Blanket, Neutronics, Nuclear, MCNP*

Corresponding author: *fabio.moro@enea.it*

Assessment of the Dose Rates Due to Water Activation on an Isolation Valve of the DEMO WCLL Breeding Blanket Primary Heat Transfer System

P1-101

P. Chiovaro [a], S. Ciattaglia [b], F. Cismondi [b], A. Del Nevo [c], P.A. Di Maio [a], G. Federici [b], Ivo Moscato [a]*, G.A. Spagnuolo [d], E. Vallone [a]

[a] *Dipartimento Ingegneria, Università di Palermo, Viale delle Scienze, Palermo, Italy*
[b] *EURO fusion Consortium, Garching, Germany*
[c] *ENEA Brasimone I, Camugnano, Italy*
[d] *Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany*

Within the framework of the activities foreseen by the EUROfusion action on the cooling water activation assessment for a DEMO reactor equipped with a Water Cooled Lithium Lead Breeding Blanket (WCLL BB), the University of Palermo is involved in the investigation of dose rates induced by the decay of nitrogen radioisotopes produced by water activation, nearby the main components (e.g. isolation valves) of both First Wall (FW) and Breeder Zone (BZ) cooling circuits.

The aim of this work is to assess the spatial distribution of these dose rates in the DEMO Upper Pipe Chase (UPC), focusing the attention on the space neighboring a typical isolation valve of the Primary Heat Transfer System (PHTS). To this end, a computational approach has been followed adopting

MCNP5 Monte Carlo code. In particular, a totally heterogeneous neutronic model of a portion of the UPC has been set up, including the valve and the main FW and BZ PHTS piping, and the spatial distribution of nitrogen isotopes concentrations, previously assessed, have been used to model the photonic and neutronic sources.

The results obtained, herewith presented and critically discussed, provided some information on the nuclear issues of the WCLL BB PHTS, to be considered as hints for the blanket design optimization.

Keywords: *DEMO, WCLL blanket, dose rate, neutronics*

Corresponding author: *ivo.moscato@unipa.it*

Neutronics Experiment with a Blanket Mock-Up Using a DD Fusion Neutron Source

P1-102

Keisuke Mukai [a]*, Yasuyuki Ogino [b], Juro Yagi [a], Satoshi Konishi [c]

[a] *Institute of Advanced Energy, Kyoto University, Gokasho, Uji, Kyoto, Japan*

[b] *Graduate School of Energy Science, Kyoto University, Kyoto, Japan*

A tritium breeding ratio (TBR) larger than unity is required for the self-sufficient fueling of a deuterium tritium fusion reactor. An experimental evaluation of tritium breeding performance of a blanket with spectral neutron fluxes is necessary because of uncertainty in the nuclear data and an unexpected neutron absorption by impurities in the blanket materials. In this work, a neutronics experiment was performed with a blanket mock-up composed of lithium carbonate solid breeder, polyethylene moderator, and carbon neutron reflector. By using a discharge-type fusion neutron source, DD neutrons with the energy of 2.45 MeV were generated and moderated within the mock-up. In the experiment, the neutron production rates of the neutron source were greater than 10^7 n/s by applying high voltages (V >70 kV). The neutron fluxes in different positions of the mock-up were measured by foil activation technique using a high purity germanium detector. In addition, a spatial distribution of thermal neutron fluence were measured by using a neutron imaging plate detector. After the irradiation, photostimulated luminescence (PSL) was read out from the neutron imaging plate and converted into neutron fluence Φ by using a linear relationship: $PSL \propto \Phi^{0.96}$. The measured data were compared with simulation results using the MCNP 5.0 code and FENDL 3.0 library.

Keywords: *neutron measurement, blanket, activation foil, tritium, neutron source*

Corresponding author: *k-mukai@iae.kyoto-u.ac.jp*

CHANGE!

P1-103

Moved to poster session 3. (P-3-135)

Measurement of Neutron Fluence with Removal of X-/γ-Ray Effect from Neutron Imaging Plate

P1-104

Yasuyuki Ogino [a]*, Makoto Kobayashi [b,c], Keisuke Mukai [d], Juro Yagi [d], Satoshi Konishi [d], Takeo Nishitani [b], Kunihiro Ogawa [b,c], Mitsutaka Isobe [b,c]

[a] *Graduate School of Energy Science, Kyoto University, Kyoto Japan*

[b] *National Institute for Fusion Science, National Institutes of Natural Sciences, Toki, Japan*

[c] *SOKENDAI (The Graduate University for Advanced Studies), Toki, Japan*

[d] *Institute of Advanced Energy, Kyoto University, Uji, Kyoto Japan*

This study proposes a new measuring technique of neutron fluence using imaging plate (IP) and neutron imaging plate (NIP). The NIP includes gadolinium as a convertor and is able to measure neutron fluence two-dimensionally by reading density of photo-stimulated luminescence (PSL). However, the NIP is also sensitive to X-/γ-rays. The IP has sensitivity to X-/γ-rays and intrinsically no sensitivity to neutrons. Neutron fluence on the NIP was obtained by substituting PSL density of the IP multiplied by sensitivity factor from that of the NIP. The IP was approximately two times more sensitive to the X-/γ-rays. Experiments were performed with the fission neutron source of ^{252}Cf and deuterium-deuterium fusion neutrons in the LHD. The ^{252}Cf spontaneous fission source emits γ-rays and fission neutrons which has broad energy spectrum at the same time. There are also neutrons and X-/γ-rays at the LHD. Though ^{252}Cf and the LHD are different fields about radiations, this method enables us to measure neutron fluence. Since there is different sensitivity to X-/γ-rays between the IP and the NIP, the calibration was performed with ^{137}Cs. The result showed that the IP was about twice as sensitive as the NIP to the X-/γ-rays. The results of this work show the feasibility of the measurement of neutron fluence using IP and NIP.

Keywords: *LHD, neutronics, imaging plate, neutron source*

Corresponding author: *ogino.yasuyuki.37w@st.kyoto-u.ac.jp*

Compatibility between Fusion Reactor Blanket Structure Material F82H and Solid Breeder Lithium Oxide

P1-105

Keiji Oishi *, Shinichi Inoue, Yumeng Rong, Kazuhiro Fudemae, Takayuki Terai

School of Engineering. The University of Tokyo, Tokyo, Japan

In Japan, the reduced activation ferritic steel F82H (Fe-8Cr-2W-0.2V-0.04Ta-0.1C) is planned to be used as a blanket structure material, and

Li_2TiO_3 has been considered as one of the most promising candidate materials among solid breeder materials. However, it has been pointed out that its Li_2O component vaporizes from the high temperature region of Li_2TiO_3 pebble bed to deposit on the low temperature piping wall. Nevertheless, there are few data on the compatibility between F82H and Li_2O at high temperature. Therefore, we investigated the compatibility between them for the safety evaluation of the solid breeder blanket.

Li_2O pellets (10 mm in diameter and 4 mm in thickness) were prepared by sintering under Ar gas atmosphere. The compatibility experiments were carried out by contacting F82H specimens (6 mm x 6 mm x 1 mm) with the Li_2O pellets directly in He + 1% H_2 sweep gas at 400 C, 600 C, and 800 C for 100 to 4000 hours. After each experiment, the O diffusion distance was actually measured from the SEM / EDX image, and the diffusion coefficient in F82H in the blanket condition was estimated to be

$0.9\sim1.9\times10^{-12}$, $1.5\sim2.6\times10^{-12}$ and $3.8\sim9.2\times10^{-11}$ cm^2 / s at 400 C, 600 C and 800 C, respectively. Based on the results, the corrosion depth of 75~110 μm is expected after the operation at 400 C for two years.

Keywords: *solid breeder blanket, F82H, Li_2O, corrosion, diffusion coefficient*

**Corresponding author:* *fusionkeiji@gmail.com*

Fundamental Analysis for Electrochemical Extraction and Monitoring of Coexisting Impurities in Lead Lithium Using Chloride Molten Salt

P1-106

Tomohiro Okada [a]*, Juro Yagi [b], Hiroyuki Miyagaki [a], Keisuke Mukai [b], Satoshi Konishi [b]

[a] *Graduate School of Energy Science, Kyoto University, Kyoto, Japan*
[b] *Institute of Advanced Energy, Kyoto University, Kyoto Japan*

LiPb liquid metal, a candidate of tritium breeding material in a fusion blanket, is easy to contain light element impurities such as oxygen, nitrogen and hydrogen. These impurities have bad influences on the compatibility with structural materials. However, an on-line measurement and extraction method for the impurities have not yet been established. In this research, electrochemical method is employed for the applications. The experimental setup was an electrochemical cell that has double liquid layer of LiPb and chloride molten salt (LiCl 58.5 at.%-KCl 41.5 at.%). LiPb was used as a counter electrode. The materials of working electrode were a glassy carbon rod, nickel wire and titanium rod in chloride molten salt. The reference electrode material was also LiPb and the potential of the reference electrode was standardized against the potential of Li+/Li. In the case of oxygen, lithium oxide was directly added to LiPb as oxygen impurities. Then the change of cyclic voltammograms (CV) caused by redox reaction at the working electrode in chloride molten salt was obtained. Furthermore, the concentration of oxygen ion in the molten salt was estimated to be 0.2mol% by comparing CV with various scan rate and this value was equal to 60% of the total input. This experimental result suggested the transfer of oxygen impurities and it was concluded that the amount of distribution of impurities could be estimated electrochemically. The results of the experiment using lithium hydride, lithium nitride and lithium oxide for the extraction and monitoring of coexisting impurities will be also presented in the conference.

Keywords: *LiPb, electrochemical, chloride molten salt*

**Corresponding author:* *okada.tomohiro.57e@st.kyoto-u.ac.jp*

Emissivity Measurement of PbLi Droplet in a Vacuum for the Heat Recovery by Radiation

P1-107

Fumito Okino [a]*, Junichi Miyazawa [b], Satoshi Konishi [c]

[a] *Institute of Advanced Energy, Kyoto University, Kyoto, Japan*
[b] *National Institute for Fusion Science, Toki, Japan*

The emissivity of liquid lithium-lead (PbLi) droplet while falling in a vacuum is theoretically estimated and verified by a dedicated experimental setup.

The authors proposed the heat and tritium simultaneous recovery from falling PbLi droplets in a vacuum. Heat is recovered by radiation and tritium is by dispersion. Permeation reduction can be expected by non-contact recovery.

Radiation depends on the emissivity of a PbLi droplet surface which is so far unknown. Theoretical estimation is performed by the electro-magnetic energy wave propagation theory at the boundary surface of finite conductivity media. Emissivity of 0.18 is obtained.

Experimental measurement is performed by the 1.1 mm diameter droplets at 425°C with the falling velocity of 3 m s^{-1}. Emissivity is calculated by the infrared temperature of droplet normal surface through a sapphire viewport, the direct droplet temperature measured by thermocouple, and the wall temperature.

Three different H_2 ambient pressure in a vacuum chamber are deployed to verify the simultaneous tritium recovery effect, which are the power of minus 3 Pascal, the power of zero, and the power of plus 2, respectively. Three conditions correspond the H_2 molecules incident rate on a droplet surface.

Obtained results are 0.35, 0.33, 0.36, respectively, which are higher than the theoretical estimation, and non-dependent to the ambient H_2 pressure. These results, still preliminary study, suggest that the simultaneous heat and tritium recovery concept is worth further consideration.

Keywords: *PbLi droplet, heat recovery, vacuum, emissivity, tritium recovery,*

**Corresponding author: fumito.okino@iae.kyoto-u.ac.jp*

Hydrogen Isotope Exchange at the Surface of C-W Mixed Material Layer on Tungsten by Gas Exposure

P1-108

Yasuhisa Oya [a]*, Akihiro Togari [a], Moeko Nakata [a], Ayaka Koike [a], Takuro Wada [a], Shota Yamazaki [a], Mingzhong Zhao [a], Fei Sun [b], Miyuki Yajima [c], Suguru Masuzaki [c]

[a] *Graduate School of Integrated Science and Technology, Shizuoka University, Shizuoka, Japan*

[b] *Faculty of Science, Shizuoka University, Shizuoka, Japan*

[c] *National Insitute for Fusion Science, Toki, Gifu, Japan*

Tungsten (W) is one of the leading candidates for plasma-facing materials (PFMs). Control of tritium retention in W is one of important issues for the enhancement of fuel efficiency and the establishment of radioactive safety in future fusion reactors. During the plasma operation, PFMs will be irradiated by energetic particles, namely hydrogen isotopes, neutron and impurities such as carbon (C). Therefore, the surface of PFMs will be dynamically changed. In especially, the formation of C-W mixed material layer on the surface may lead the changing of hydrogen isotope trapping characteristics. As a result, tritium retention in fusion reactor vessel will be significantly enhanced. As a possible method to remove the surface tritium, hydrogen isotope exchange with deuterium is suggested. To verify the effect of hydrogen isotope exchange, the hydrogen isotope retention behaviour in the W samples covered by the mixed material layer should be evaluated with various conditions.

In this study, hydrogen isotope exchange experiment by gas exposure for W samples was performed. Each sample has C-W mixed material layer, which is deposited by plasma-enhanced chemical vapor deposition (PECVD). The samples were exposed to hydrogen isotope gas at various temperatures, pressures and durations. In addition, sequential gas exposure of hydrogen and deuterium was performed at various conditions to confirm the effect of hydrogen isotope exchange reaction. Thereafter, thermal desorption spectroscopy (TDS) measurement was carried out from R.T. to 1173 K to evaluate the retention behaviour of all the hydrogen isotopes simultaneously. The desorption spectra of H_2, HD and D_2 were evaluated by quadrupole mass spectrometer and those of tritium were detected by proportional counters and liquid scintillation counters.

The experimental results show that carbon structure has a major effect for hydrogen isotope retention behaviour on W with mixture layer. In addition, hydrogen isotope exchange reaction was accelerated at the temperature of 573 K and it is efficient to remove retained tritium on the surface region of mixture layer.

Keywords: *Tungsten, deposition, retention*

**Corresponding author: oya.yasuhisa@shizuoka.ac.jp*

Construction of the Hot Helium Leak Test Facility for the ITER Blanket Shield Block

P1-109

Byungil Park [a]*, Sawoong Kim [a], Hee-Jin Shim [a], Yonghee Lee [a], Hyok Lee [b], Jong-Ha Park [b]

[a] *ITER Korea Domestic Agency, National Fusion Research Institute, Daejeon, Korea*

[b] *VITZROTECH Co., Ltd, Ansan-City, Kyunggi-Do, Korea*

The ITERBlanket Shield Block (SB) is one of a main in-vessel componentsto contribute in absorbing radiation and particle heat fluxes from plasma and providing neutronic shielding to the Vacuum Vessel (VV) and external vessel components. The SB is manufactured by using of 316L(N)-IG stainless steel with internal cooling channels made by deep drilling process.The water headers are adopt mainly plasma side of SB and closed by cover plates with welding process. All the in-vessel components shall be dealt with Vacuum Quality Class 1A (VQC-1A) which is most high classification level in the ITER systems, due to keep with anultra-high vacuum level in the VV. Therefore, each of the SBs have to be confirmed the soundness of welded parts and to reduce the risk of incorporating leaks during/after the manufacturing process. Vacuum leak tests shall be performed both at ambient temperature and at the maximum (or minimum) working temperatures of the component, with the pressure differential in the same direction as for operation of the component. Maximum leak rate of 1 x 10^{-10} Pa·m^3/s, minimum sensitivity level of 4 x 10^{-11} Pa·m^3/s at ambient temperature and 1 x 10^{-10} Pa·m^3/s at elevated temperature is specified for the SBs.

The ITER Korea Domestic Agency (KODA) had been launched to construct the Hot Helium Leak Test (HHLT) facility, which shall have sufficient working volume for the SBs including fixture system (max. 1.8 m in W x 1.3 m in T x 1.6 m in H, 5 tons). This paper provides the information on the design and construction of HHLT facility, and the results of the Factory Acceptance Test for the SB06 Full Scale Prototype including commissioning tests. The results also might be a good lessons

and learned for the design and construction of 2nd HHLT facility.

Keywords: *ITER Blanket Shield Block, Hot Helium Leak Test, Factory Acceptance Test*

Corresponding author: *bipark@nfri.re.kr*

Analyses of the Shielding Options for HCPB DEMO Blanket

P1-110

Pavel Pereslavtsev [a]*, Fabio Cismondi [b], Francisco A. Hernández [a]

[a] *Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany*
[b] *EUROfusion –Programme Management Unit, Garching, Germany*

The DEMO fusion facility will be built making use of a well proved and validated blanket technology providing a) tritium breeding sufficient for a self-sustaining fuel cycle, b) nuclear heat generation to be later converted into industrial electricity and c) reliable protection of the Vacuum Vessel (VV) from the intensive neutron and gamma irradiations. Within the Power Plant Physics and Technology (PPPT) programme of EUROfusion, development efforts are spent to the elaboration of two promising concepts: Helium Cooled Pebble Bed (HCPB) and Water Cooled Lithium-Lead (WCLL) breeder blankets. An arrangement of efficient materials mitigating accumulation of nuclear damage in the VV can be considered as a design option enhancing shielding performances of the DEMO blanket concept. To this end full-scale 3D completely heterogeneous geometry models of the both HCPB and WCLL DEMOs adopting the latest base line 2017 were developed for neutronic simulations with MCNP computer code. The optimization of the HCPB blanket shielding performances were performed making use of two different approaches. In the first, after a screening of possible, practical shielding materials for the HCPB, the shielding blocks of various materials such as B_4C, WC, ZrH_x and YH_x were arranged in the blanket manifold or behind blanket to achieve the same shielding performance like in the case of the WCLL. In the second, the radial thickness of the WCLL blanket was changed to achieve the maximum tritium breeding performance like in the case of the HCPB. The radial dimensions of the HCPB blanket were changed respectively, the gained additional space was used for the arrangement of the shielding materials and the damage accumulation in the VV were compared with the WCLL case. Technological and fabrication aspects, as well as an experience in the nuclear field with the most promising options will be discussed.

Keywords: *DEMO, HCPB, shielding materials, neutronics*

Corresponding author: *pavel.pereslavtsev@kit.edu*

Strength Analysis and Impact Test of Stub Key Pads of ITER Blanket

P1-111

Ivan Poddubnyi [a]*, V. Kolganov [a], M. Sviridenko [a], S. Timofeev [a], S. Khomiakov [a], S. Tomilov [a], D. Vlasov [a], A. Leshukov [a], Yu. Strebkov [a], I. Danilov [a], R. Tagiev [a], A. Putrik [b], K. Egorov [c], R. Raffray [c]

[a] *Joint-Stock Company "N.A. Dollezhall Research and Development Institute of Power Engineering", (JSC "NIKIET"), Moscow, Russia*
[b] *Institution "Project Center ITER", Moscow, Russia*
[c] *ITER Organization, Saint Paul lez Durance, France*

Insulated pads are used on ITER blanket system for transferring the loads induced in blanket modules during the plasma disruptions to vacuum vessel keys. To minimize or avoid current loops running between pads and the keys, an electrical insulation coating (Al_2O_3) is applied on the pads. There are two main types of insulated pads in the blanket module connections system: Intermodular and Stub Key (SK) pads, situated in inboard and outboard blanket, respectively. Previous analyses of electromagnetic forces induced in blanket modules during plasma disruptions and the resulting reactions on the pads have shown that the unsteady impact character of these reactions. The diagram of reactions due to electromagnetic forces has a half-sinusoidal form with amplitude up to 1.7 MN for the intermodular key pad and up to 1.2 MN for the SK pad. The characteristics duration of the pulse is 1.5 – 2.5 ms.

The integrity of electrical insulation and, consequently, the workability of insulation pads under electromagnetic forces need to be confirmed by impact test of full scale prototypes of the pads. These impact tests are carried out on a drop weight impact test jig, which has been designed and manufactured in JSC "NIKIET".

The description of the design of the SK pad, the results of strength analysis under impact test load fulfilled using the Johnson-Cook plasticity model, and the results of the impact test carried out on the SK pad prototype are presented in this paper.

Keywords: *Blanket system, Blanket module connectors, Insulated pads, impact tests*

Corresponding author: *poddubnyyii@nikiet.ru*

Turbulent Heat Transfer for Coolant Water Flow in Plasma Facing Component

P1-112

Junya Uchiyama [a], Yohji Seki [b], Tomoaki Kunugi [c], **Shin-ichi Satake** [a*]

[a] *Department of Applied Electronics, Tokyo University of Science, Tokyo, Japan*
[b] *National Institutes for Quantum and Radiological Science and Technology, Ibaraki-ken, Japan*
[c] *Kyoto University, Kyoto, Japan*

It is important to investigate characteristics of thermofluid for cooling blankets and divertors used in the high-temperature plasma of fusion reactors. Various coolant materials on the cooling problem such as a fusion reactor were widely considered from FLiBe to lithium. In the present study, we used the working fluid as a pressurized water for the blanket design. we perform a Direct numerical simulation (DNS) of a pipe flow by using the JFRS-1 supercomputer at the Computational Simulation Centre of the International Fusion Energy Research Centre (IFERC-CSC). Reynolds number is 2100 based on the pipe radius and friction velocity. The numbers of mesh points used for 4096 × 1024 × 1532 in the *z*−, r−, and □-directions, respectively. Prandtl number is set to be 0.87. The detailed turbulent quantities such as the mean flow, turbulent stresses, turbulent kinetic energy budget, and the temperature statistics were obtained.

Keywords: *DNS, Turbulent flow, coolant water flow, Plasma facing component*

Corresponding author: *satake@te.noad.tus.ac.jp*

Overview of Test Modules for Advanced Fusion Neutron Source A-FNS

P1-113

Satoshi Sato*, Makoto M. Nakamura, Saerom Kwon, Masayuki Ohta, ChangHo Park, Makoto Oyaidzu, Kentaro Ochiai, Atsushi Kasugai, Keishi Sakamoto, Shinichi Ishida

National Institutes for Quantum and Radiological Science and Technology, Aomori, Japan

Conceptual design of Advanced Fusion Neutron Source (A-FNS) has been carried out by QST. A-FNS generates highly intense neutrons by D-Li reaction. We newly design most of the irradiation test modules for A-FNS; Blanket Structural Material Test Module (BSMTM), Blanket Functional Material Test Module (BFMTM), Tritium Release Test Module (TRTM), Activated Corrosion Product Module (ACPM), Blanket Nuclear Property Test Module (BNPTM), Diagnostic Controlled Device Test Module (DCTM) and Creep Fatigue Test Module (CFTM). We acquire the irradiation data on the structural and functional (tritium breeder and neutron multiplier) materials of the DEMO blanket by using the BSMTM and BFMTM, respectively, under the condition of the fusion neutron spectrum. In the TRTM, we acquire the irradiation data on the tritium release and recovery properties from the small pebbles of tritium breeder and neutron multiplier. We flow the helium purge gas through the pebble packed capsules, and measure these data by on-line measurement. In the ACPM, we irradiate the cooling water pipe of F82H. High temperature and pressure water, which are 300 °C and 15 MPa, respectively, flows inside the F82H pipe, and these are same condition of the DEMO blanket. We measure the active corrosion production of the F82H pipe, and acquire the irradiation data for the safety evaluation of DEMO fusion reactor. In the BNPTM, we perform the nuclear property experiment by using the DEMO blanket mockup. We measure the detailed distribution of the tritium production rate and the neutron flux to evaluate the calculation accuracy for the neutronics design. We design these modules and associated shielding plug without any joints of cables and coolant pipes in the test cell. These designs enhance remote maintainability of these modules. We present the overview of these test modules, and the detail of the conceptual design on the BSMTM.

Keywords: *fusion neutron source, A-FNS, DEMO blanket, irradiation test module, F82H*

Corresponding author: *sato.satoshi@qst.go.jp*

Design and Experimental Study of Adsorption Bed for the Helium Coolant Purification System

P1-114

Chang Wook Shin [a], Eo Hwak Lee [a], Suk-Kwon Kim [a], Hyung Gon Jin [a], Dong Won Lee [a], Mu-Young Ahn [b]

[a] *Korea Atomic Energy Research Institute, Daejeon, Korea*
[b] *National Fusion Research Institute, Daejeon, Korea*

In the design of a fusion reactor, a helium cooling system (HCS) is used to extract heat from the blanket. To reduce the permeated tritium in the helium, coolant purification system (CPS) has been developed and experimental loop has been constructed. In the present paper, the main components of the CPS loop such as oxide and absorption beds were introduced focusing with design, fabrication and their performance tests.

Since the molecular size of tritium is too small to be captured, it is oxidized to Q_2O using a copper oxide bed. Then, an adsorption bed using molecular sieve captures Q_2O through physical adsorption. For the design of CPS, correlation of previous studies and general properties of molecular sieve were used. Since the concentration of Q_2 in this system is estimated as 0.3 Pa, which is very low compared to the helium pressure of 8 MPa, the used correlations for design and saturation characteristics of the absorption bed under low partial pressure of Q_2O should be validated. In the present study, component test was prepared to verify the designed absorption bed, which can simulate the helium flow condition with low concentration of Q_2. The test was performed under various conditions such as temperature, pressure, concentration of Q_2 and velocity. The results will be used for CPS loop design to be connected with helium supplying system (HeSS).

Keywords: *adsorption bed, molecular sieve, coolant purification system (CPS)*

Corresponding author: *cwshin@kaeri.re.kr*

Experimental Evaluation of the Dynamic Viscosity of Molten Salt Flinabe and Validation of the Polarizable Ion Model

P1-115

Hiroki Shishido [a]*, Hidetoshi Hashizume [a], Yoshiki Ishii [b], Norikazu Ohtori [c]

[a] *Department of Quantum Science and Energy Engineering, Graduate School of Engineering, Tohoku University, Miyagi, Japan*
[b] *Division of Chemical Engineering, Graduate School of Engineering Science, Osaka University, Osaka, Japan*
[c] *Faculty of Science, Niigata University, Niitgata, Japan*

Molten salt Flinabe - a mixture of LiF, NaF, and BeF_2 - has attracted attention recently as a tritium breeder candidate for a self-cooled liquid blanket system in a fusion reactor. Whereas the feasibility evaluation is vital for the Flinabe blanket system, the heat transfer properties have remained almost unclear. Our previous study has performed molecular dynamics simulations with a polarizable ion model to predict the heat transfer properties of Flinabe. In the results, Flinabe at a certain composition showed the considerably high viscosity at low temperature. The viscosity characteristics need to be clarified in order to confirm its applicability to a fusion blanket system.

The present study aims to reveal the viscous behavior of Flinabe at low temperature. First, molecular dynamics simulations are conduct to evaluate the dependence of viscosity on the temperature and composition of Flinabe. Calculation results have indicated that the several composition Flinabe has almost the same rate of viscosity increase for temperature; therefore, the lower melting point composition Flinabe has the higher viscosity at low temperature. Subsequent the viscosity of Flinabe is experimentally measured by using the rotating cylinder method. The measurement data is compared to the numerical simulation results in order to validate the polarizable ion model.

Keywords: *rotating cylinder method, physical properties, molecular dynamics, liquid blanket*

Corresponding author: *hiroki.shishido.a7@tohoku.ac.jp*

MHD Forced Convection Flow in Dielectric and Electro-Conductive Rectangular Annuli

P1-116

Simone Siriano [a]*, Alessandro Tassone [a], Gianfranco Caruso [a], Alessandro Del Nevo [b]

[a] *Department of Astronautical, Electrical and Energy Engineering, Sapienza University of Rome, Roma, Italy*
[b] *ENEA FSN-ING-PAN, ENEA CR Brasimone, Camugnano, Italy*

The Water-Cooled Lithium Lead (WCLL) is a blanket concept investigated for implementation in DEMO and the only one using an electrically conductive fluid as tritium breeder. The liquid metal is distributed to the elementary cells composing the component breeding zone (BZ) through a compact poloidal manifold which fulfills both the distribution and collection task. The manifold is composed by two co-axial rectangular channels of which the internal one, tasked with distributing the liquid metal, can be described as a rectangular annulus. In this paper, the MHD forced convection flow in this uncommon configuration is studied in order to gain insights about fluid dynamics and pressure losses. Numerical simulations are performed for a wide range of magnetic field intensity (H_a = 10 ÷ 1000) using the CFD code ANSYS CFX. First, the analysis is focused on an ideal case, in which both the bounding and internal annulus wall are perfectly insulating (c = 0), in order to characterize the basic flow features and phenomena. Successively, the study is extended to a more realistic case, such as it is foreseen in the WCLL current design, where both the bounding and internal annulus walls are characterized by finite electrical conductivity (c = 0.1) and high velocity jets appear close to walls parallel to the magnetic field direction.

A correct estimate of the MHD pressure drop is critical to design the WCLL PbLi in-vessel loop, therefore an accurate estimate for this configuration is required. The pressure gradient calculated by the CFD code for both dielectric and electro-conducting annuli (∇p_a) is compared with the analytical value characterizing the fully developed flow in a rectangular channel of equivalent cross-section (∇p_c) in order to define proper engineering correction factors ($\lambda = \nabla p_a/\nabla p_c$) to correlate these two configurations.

Keywords: *Magnetohydrodynamics (MHD), blanket engineering, WCLL, CFD, rectangular annulus*

**Corresponding author:* *simone.siriano@uniroma1.it*

Electromagnetic Coupling Phenomena in Co-Axial Rectangular Channels

P1-117

Simone Siriano [a]*, Alessandro Tassone [a], Gianfranco Caruso [a], Alessandro Del Nevo [b]

[a] *Department of Astronautical, Electrical and Energy Engineering, Sapienza University of Rome, Roma, Italy*
[b] *ENEA FSN-ING-PAN, ENEA CR Brasimone, Camugnano, Italy*

In the Water-Cooled Lithium Lead (WCLL) blanket, the eutectic alloy lithium lead (PbLi) is used as tritium breeder and carrier, neutron multiplier, and heat transfer medium. The liquid metal is distributed to and collected from the elementary cells composing the component breeding zone (BZ) through a compact poloidal manifold made of two co-axial rectangular channels. The external channel, tasked with flow distribution, and the internal one, to which the collection is demanded, are counter-flowing and share an electrically conductive wall ($c = 0.1$). Leakage currents are expected to significantly modify the flow features and pressure losses therein these channels.

In this paper, electromagnetic coupling phenomena are investigated with the aid of the CFD code ANSYS CFX in a wide range of magnetic field intensity (Ha = 10 ÷ 1000). First, the study is focused on a simpler configuration, in which both channels are co-flowing, to characterize the fluid dynamics compared with the uncoupled case, which is in turn composed by a rectangular electro-conductive annulus (external) and a square electro-conductive duct (internal). After that, the same configuration is investigated for counter-flowing channels in order to gain insights about the case currently envisioned in the WCLL blanket.

An accurate estimate of the MHD pressure drop is critical to design the WCLL PbLi in-vessel loop. Therefore, in order to define the proper engineering correction factors (ψ) to correlate the coupled case pressure drop (∇p_{ec}) with the uncoupled one (∇p_{uc}), the pressure gradient calculated by the CFD code for both the cases is compared ($\psi = \nabla p_{ec}/\nabla p_{uc}$).

Keywords: *Magnetohydrodynamics (MHD), blanket engineering, WCLL, CFD, electromagnetic coupling*

**Corresponding author:* *simone.siriano@uniroma1.it*

Development of Load Specifications for the Design of the Breeding Blanket System

P1-118

Gandolfo Alessandro Spagnuolo [a]*, Lorenzo Virgilio Boccaccini [a], Gaetano Bongiovì [a], Fabio Cismondi [b], Ivan Alessio Maione [a]

[a] *Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany*
[b] *EUROfusion Consortium, Garching, Germany*

The Breeding Blanket (BB) system with a volume of about 1700 m^3 and a plasma front surface of about 1400 m^2 represents the largest In-Vessel component of DEMO reactor. For its position and for its key functions (e.g. tritium production, power removal and shielding performance), the BB is also one of the most critical component. The single loads acting on the BB can be of different nature (inertial, pressure, thermal and electromagnetic loads, for instance) and their combination may produce high stresses jeopardizing the BB structural integrity if not carefully taken into account during the design. For these reasons, within the EUROfusion consortium, the development of BB system load specifications has been pursued since the early design stage.

The main goals of this work are: (i) to list of all relevant single loads and load combinations to be considered to verify the BB structural integrity, and the categorization of these relevant load combinations, (ii) to identify the short list of load combinations relevant to the pre-conceptual design review phase in sight of 2020 Gate Review. Particular emphasis is also given to the most representative postulated initiating events, which drive the design of the BB, providing their respective load combination.

Keywords: *DEMO, Breeding Blanket, load specifications, postulated initiating events*

**Corresponding author:* *alessandro.spagnuolo@kit.edu*

Analysis of Flow Channel Insert Deformations Influence on the Liquid Metal Flow in DCLL Blanket Channels

P1-119

Dionisio Di Giulio [a], Daniel Suarez i Cambra [b]*, Lluis Batet [b], Elisabet Mas de les Valls [b], Laura Savoldi [a]

[a] *Dipartimento di Energia, Politecnico di Torino, Torino, Italy*
[b] *Departament de Fisica, Universitat Politécnica de Catalunya, Barcelona, Spain*

The dual coolant lithium lead (DCLL) is a candidate to be an effective breeding blanket (BB) concept for nuclear fusion technologies. One critical point of this design is the magnetohydrodynamic (MHD) effects involving Lorentz damping force which produces relevant pressure drop in the eutectic flow. In the framework of the European DEMO, the application of sandwich-like steel-alumina-steel Flow Channel Insert (FCI) seems to be the best solution to reduce the pressure drop by electrically decoupling the liquid *PbLi* from the EUROFER walls. The impact of the FCI on the *PbLi* velocity profile is analyzed here with a CFD solver implemented on OpenFOAM. Under the assumption of non-buoyant fully developed channel flow, also the temperature map in the channel is computed and, based on that, the induced deformation is evaluated. The effects of the FCI deformation and possible rupture on the velocity profile and on the corresponding pressure drop are then parametrically investigated.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No. 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *MHD, FCI, BB, DCLL, EU-DEMO, CFD*

****Corresponding author:*** *daniel.suarez.cambra@upc.edu*

Development of Secondary Charged Particle Activation Based Method for Tritium Production Rate Measurement in Fusion Blankets

P1-120

Gergő I. Pokol [c], Máté Szieberth [a]*, Gergely Klujber [a], Árpád Bihari [b], László Palcsu [b]

[a] *Budapest University of Technology and Economics (BME), Institute of Nuclear Techniques (NTI), Budapest, Hungary*
[b] *Institute for Nuclear Research of the Hungarian Academy of Sciences (MTA ATOMKI), Debrecen, Hungary*
[c] *Wigner RCP, Budapest, Hungary*

In the recent years a secondary charged particle activation (SCPA) based technique was developed and tested for tritium production rate (TPR) measurement at BME NTI. In samples containing Li mixed with properly chosen indicator material the accelerated tritium particle produced from Li by neutron induced reaction can produce a secondary nuclear reaction in the indicator nuclei. Its reaction rate will be proportional to the TPR and can be measured based on the gamma radiation of the reaction product (monitor). The advantage of the method is that it can provide and accurate estimation of the TPR on site, in a short time after gamma spectrometry of the irradiated sample, without the complications of the standard tritium measurement techniques (e.g.: Liquid SCintillation). The method is suitable to be applied in ITER test blanket modules (TBM) and in the breeding blankets of future fusion reactors. The method was verified with several indicator materials by irradiations at the 100 kW power Training Reactor of BME NTI (BME TR) and it is planned to be tested in a fusion relevant neutron spectrum during JET DTE2 campaign as part of the EUROfusion WPJET3 project. In order to cope with the low neutron fluent in JET pulses the $^{16}O(t,n)^{18}F$ reaction was chosen having the highest tritium induced cross-section, and a larger sample composed of Li_2CO_3 was developed and tested at BME TR. The tritium content of the samples was also measured by LSC technique at MTA ATOMKI. The comparison of the results confirms linear relation between the amount of produced tritium and ^{18}F in the pellets.

The paper presents the principles of the SCPA method, the design of the planned measurements at JET, the testing of the method performed at BME and the comparison with standard tritium measurements.

Keywords: *tritium production, breeding blanket, SCPA, LSC, JET*

****Corresponding author:*** *szieberth@reak.bme.hu*

Applicability Study of Blanket Systems with Liquid Tritium Breeder/Coolant and Liquid Neutron Multiplier in Helical Reactor FFHR Designs

P1-121

Teruya Tanaka [a,b]*, Junichi Miyazawa [a,b], Masatoshi Kondo [c], Juro Yagi [d], Takuya Goto [a,b], Akio Sagara [a,b], Hitoshi Tamura [a,b], Nagato Yanagi [a,b] and the FFHR Design Group [a]

[a] *National Institute for Fusion Science, Gifu, Japan*
[b] *The Graduate University for Advanced Studies, Gifu, Japan*
[c] *Tokyo Institute of Technology, Tokyo, Japan*
[d] *Kyoto University, Kyoto, Japan*

In the designs of the helical reactor FFHR series, self-cooled FLiBe+Be (liquid FLiBe tritium breeder /coolant and solid beryllium neutron multiplier) and FLiNaBe+Be blanket systems have been adopted as the first candidates due to the high tritium breeding performance, significantly low MHD pressure drop under the magnetic field and high safety [1,2]. In the present study, applicability of self-cooled liquid blanket systems without using solid beryllium multiplier, i.e., FLiBe+Pb, FLiNaBe+Pb, FLiNaK+Pb and LiPb blanket systems (^{6}Li: 90 % for all), has been studied to explore attractive optional concepts for the present FFHR design. Tritium breeding ratios (TBRs) of the three molten salt blanket systems calculated by the MCNP5 code for the FFHR-d1 Cardistry blanket model [3] and the values of > 1.07 were obtained. Further improvement of the TBRs could be expected by the position optimization of Pb multiplier. Regarding the LiPb blanket system, the TBR obtained for the composition of 17Li-83Pb was 1.05. Since the basic configuration of the

reactor components of FFHRd1 is determined by enlarging that of the present plasma experimental machine LHD (Large Helical Device) for aiming at the earliest realization of a fusion reactor, the blanket thickness at the inboard side is thin compared with other reactor designs and this was the reason for the relatively low TBR. While the melting point increases from 235 °C to ~300°C, the TBR was improved to 1.14 with the composition of 25Li-75Pb. The results of the TBR calculations indicate that all the four blanket systems could be options in the FFHR designs from the aspect of tritium breeding. Although these four blanket concepts could eliminate necessity for development of Be pebbles used in a coolant circulation, other issues such as material compatibilities, requirement of a new data base, etc. are considered to be appeared. Impacts on the blanket module designs are also being discussed in the study.

[1] A. Sagara et al., Fusion Eng. Des. 29(1995)51-56.

[2] A. Sagara et al., Fusion Eng. Des. 89(2014)2114-2120.

[3] J. Miyazawa et al., Plasma and Fusion Research 12 (2017)1405017.

Keywords: *FLiNaBe, FLiBe, Pb multiplier, LiPb, liquid blanket*

Corresponding author: *teru@nifs.ac.jp*

MHD Pressure Drop Estimate for the WCLL In-Vessel PbLi Loop

P1-122

Alessandro Tassone [a]*, Simone Siriano [a], Gianfranco Caruso [a], Marco Utili [b], Alessandro Del Nevo [b]

[a] *Department of Astronautical, Electrical and Energy Engineering, Sapienza University of Rome, Roma, Italy*

[b] *ENEA FSN-ING, CR Brasimone, Camugnano, Italy*

In the Water-Cooled Lithium Lead (WCLL) blanket, the eutectic alloy lithium lead (PbLi) is used as tritium breeder and carrier, neutron multiplier, and heat transfer medium. The PbLi hydraulic loop section within the vacuum vessel, which includes the breeding blanket, is affected by intense magnetic fields which cause the liquid metal transition to the MHD regime. Intense Lorentz forces oppose the fluid motion and cause pressure losses several order of magnitude higher than for the ordinary hydrodynamic regime. An accurate estimate of the MHD pressure drop is mandatory to properly design the PbLi hydraulic loop and to optimize the flow path in the breeding blanket.

In this paper, the in-vessel section of the PbLi loop for the WCLL blanket is divided into three main hydraulic regions (feeding and draining pipe, manifold, and breeding zone) which are further discretized into basic hydraulic elements (straight duct, bends, sudden cross-section variation, etc.). Analytical correlations and numerical results available in the literature are then used to calculate the two-dimensional and three-dimensional MHD pressure drop terms for each element, thus allowing to estimate the overall pressure drop in both the inboard and outboard in-vessel section of the Pb-Li loop. The study highlights that the highest contributions to the pressure losses are localized within the blanket spinal manifold and the connection pipes with the ex-vessel section of the PbLi loop. Sensitivity analyses to assess the influence of geometrical parameters on MHD pressure drop in these elements are presented and optimization strategies to minimize the pressure losses are suggested. Corrosion rates are estimated in the blanket manifold and connection pipes from the calculated local MHD velocity.

Keywords: *Magnetohydrodynamics (MHD), blanket engineering, WCLL, PbLi, pressure drop*

Corresponding author: *alessandro.tassone@uniroma1.it*

Magnetohydrodynamics Effect on Tritium Transport at Breeder Unit Level for the WCLL Breeding Blanket of DEMO

P1-123

Ciro Alberghi [a], Luigi Candido [a], Raffaella Testoni [a]*, Marco Utili [b], Massimo Zucchetti [a]

[a] *Dipartimento Energia "Galileo Ferraris" - Politecnico di Torino, Torino, Italy*

[b] *ENEA FSN-ING- QMN- C.R. Brasimone, Camugnano, Italy*

The Water-Cooled Lithium-Lead (WCLL) is one of the four breeding blanket concepts proposed by Europe in view of DEMO reactor. The velocity field of the electrically conducting lead-lithium eutectic alloy inside the blanket is strongly influenced by the external magnetic field used for plasma confinement combined with buoyancy effect. The strength of the magneto-hydro-dynamics (MHD) effect depends on the intensity of the magnetic field and its orientation with respect to the direction of lead-lithium motion. This phenomenon significantly influences the resulting temperature and velocity fields, and therefore the tritium transport inside the breeder blanket. A multi-physics approach of a 3D tritium transport model is presented for a simplified geometry of WCLL breeding blanket. In particular, a single water tube surrounded by lead-lithium has been considered. The tritium transport has been coupled with the MHD and heat transfer aspects. In particular, the advection-diffusion of tritium into the lead-lithium eutectic alloy, transfer of tritium from the liquid interface towards the steel (adsorption/desorption), diffusion of tritium inside the steel, transfer of tritium from the steel towards the coolant (recombination/dissociation), and advection-diffusion of diatomic tritium into the coolant, temperature field, velocity fields of both lead-lithium and water, and MHD effect have been included in this study. With this model, the tritium concentration and the inventories inside the lead-lithium, in the Eurofer pipes, and in the water coolant have been evaluated. Results deriving from the developed model, with the above specified phenomena, input and boundary conditions are illustrated in detail within the paper.

Keywords: *WCLL, breeding blanket, DEMO, tritium transport, MHD*

**Corresponding author:* *raffaella.testoni@polito.it*

Megawatt Power Generation of the Dual-Frequency Gyrotron for TCV at 84 and 126 GHz, in Long Pulses

P1-124

S. Alberti [a], J.-P. Hogge [a], K.A. Avramidis [b], A. Bruschi [d], W. Bin [d], F. Cau [f], F. Cismondi [g], J. Dubray [a], D. Fasel [a], G. Gantenbein [b], S. Garavaglia [d], J. Genoud [a], S. Illy [b], J. Jin [b], F. Legrand [c], R. Marchesin [c], B. Marlétaz [a], J. Masur [a], A. Moro [d], C. Moura [a], I.Gr. Pagonakis [b], E. Périal [c], L. Savoldi [e], T. Scherer [h], U. Siravo [a], M. Thumm [b], M. Toussaint [a], **Minh-Quang Tran** [a*]

[a] *EPFL-Swiss Plasma Center (SPC), Lausanne, Switzerland*
[b] *IHM, Karlsruhe Institute of Technology, Karlsruhe, Germany*
[c] *Microwave Imaging Solution, THALES, Vélizy-Villacoublay, France*
[d] *Institute of Plasma Physics "P. Caldirola", National Research Council of Italy, Milano, Italy*
[e] *NEMO Group, Dipartimento Energia, Politecnico Di Torino (POLITO), Torino, Italy*
[f] *Fusion for Energy, Barcelona, Spain*
[g] *Euro-Fusion, Garching, Germany*
[h] *Institut für Angewandte Materialien, KIT, Karlsruhe, Germany*

For future fusion devices (e.g. the EU DEMO), to increase the flexibility of the ECW systems, gyrotron operating at two frequencies are being developed. This paper presents the experimental results of such gyrotron developed in the frame of the TCV Tokamak upgrade, for which two 84/126 GHz/2 s dual frequency Gyrotrons designed by SPC and KIT and manufactured by THALES will be added to the existing EC-System. The first unit has been delivered to EPFL-SPC and tested. Owing to the flexible triode gun design giving the possibility to adjust the pitch angle parameter, the specifications were met at both frequencies. At 84 GHz ($TE_{17,5}$ mode), a power of 0.930 MW was measured in the calorimeter, with a pulse duration of 1.1 s. At the high frequency (126 GHz, $TE_{26,7}$ mode), a power of 0.98 MW was reached for a pulse length of 1.2 s. Accounting for the load reflection and the ohmic losses in the various subcomponents of the transmission line and the tube, it is estimated that the output power at the gyrotron window is in excess of 1 MW at both frequencies, with an electronic efficiency of 32% and 34% at 84 GHz and 126 GHz respectively. The gyrotron behavior is remarkably robust and reproducible, and the pulse length is limited by external systems that will be improved shortly.

Keywords: *ECRH, Gyrotrons, High Power microwave sources.*

**Corresponding author:* *minquang.tran@epfl.ch*

Experimental Qualification of New Instrumentation for Lead-Lithium Eutectic in IELLLO Facility

P1-125

Alessandro Venturini [a*], Marco Utili [b], Nicola Forgione [a]

[a] *Dipartimento di Ingegneria Civile e Industriale, University of Pisa, Pisa, Italy*
[b] *ENEA Brasimone, Camugnano, Bologna, Italy*

The experimental facility IELLLO (Integrated European Lead Lithium LOop) was installed in ENEA Brasimone R.C. in 2007, aiming to support the design of liquid Test Blanket Modules that will be installed in ITER and to contribute to the development of Lead-Lithium Eutectic (LLE) technologies. A first experimental campaign was carried out in 2015 to assess the performances of the main components of the loop and of the installed instruments. IELLLO was then upgraded by installing new instrumentation that was identified as relevant for ITER application and that have to be qualified in flowing LLE. Five differential pressure transducers, a Coriolis mass flow meter and a thermal mass flow meter were installed in the facility. A new experimental campaign was planned, setting two main objectives. The first objective was to qualify new instrumentation for flowing LLE, varying the flow rate and the temperature of the alloy. The installation of a differential pressure transducer across each flow meter made also possible to better characterize these instruments by measuring their pressure drops. The second objective of this activity was to improve the results of the 2015 campaign by analyzing the performances of the main components of the loop at lower mass flow rates (namely 0.1-1.2 kg/s) and by quantifying the pressure head of the permanent magnet pump and the pressure drops across the air cooler and the economizer. The investigated flow rates were chosen to be relevant for the LLE loop of the WCLL TBS (Water Cooled Lead-Lithium Test Blanket System).

This work presents the results of the new experimental campaign, paying particular attention to underline the lessons learned on how to correctly operate a LLE loop.

Keywords: *Lead-Lithium Eutectic, Instrumentation, Pressure Transducer, Flow Meter, IELLLO*

**Corresponding author:* *alessandro.venturini@for.unipi.it*

Recent Progress in Manufacture Technology of CFETR WCCB at ASIPP

P1-126

Wanjing Wang [a]*, Jichao Wang [a], Qingran Gao [a], Chunyi Xie [a], Qiang Li [a], Sigui Qin [b], Yingli Shi [b], Guohui Liu [b], Songlin Liu [a] Guang-Nan Luo [a]

[a] *Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP), Hefei, China*

[b] *Advanced Technology and Materials Company (AT&M), Beijing, China*

As one candidate of blankets for China Fusion Engineering Test Reactor (CFETR), the Water Cooled Ceramic Breeder (WCCB) blanket has been proposed and designed in Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). The structural material uses Reduced Activation Ferritic/Martensitic (RAFM) steel, while the first wall (FW) material is tungsten armor to protect it from plasma erosion and corrosion. To get a better heat transfer capability, these two materials must be joined by metallurgical bonding method. In addition, the sub-components of the blanket, such as U-shape first wall, the cooling and stiffening gird plates, with a series of long square tube, also need to develop the appropriate welding technology to achieve the bonding of Steel/Steel in its fabrication. Except for these fundamental science issues, we also face some engineering technical problems in the manufacture of small-size mockup and full-size component of WCCB blanket, including the practice of former bonding technical on enlarged mockup and the assembling of sub-components.

In the past several years, I have led a team to research the bonding issues and develop the manufacture technology of CFETR-WCCB blanket. We have proposed an R&D roadmap of 5 years. In this work, the bonding technology for W/Steel and Steel/Steel developed at ASIPP has been summarized. In the W/Steel bonding, the effects of interlayer and intermetallic on the joints failure have been discussed. In the Steel/Steel bonding, the deformation issue and the heat treatment of RAFM Steel have been presented, and their effects on the bonding quality also have been shown. Then some recent progress on the manufacture of WCCB blanket mockup has also been presented, including the development of high-heat flux testing requirement for first-wall. It is considered that these researches construct a foundation to manufacture the WCCB component.

Keywords: *Blanket/First wall, Manufacture technology, W/Steel bonding*

Corresponding author: *wjwang@ipp.ac.cn*

Study of Hydrodynamics and Thermal Transfer in Duct with 180° Sharp Bend in DCLL Blankets

P1-127

He Wang [a]*, Nian-Mei Zhang [b]

[a] *School of Engineering Sciences, University of Chinese Academy of Sciences, Beijing, China*

[b] *School of Engineering Sciences, University of Chinese Academy of Sciences, Beijing, China*

The metal fluid flow in DCLL Blankets is subjected to the strong external magnetic field and big heat source produced by neutron reaction. The flow, heat transfer and thermal stress of duct interactively make up the magneto-thermo-fluid-structure multi-physical coupled field. In this work, a numerical simulation of liquid metal flow in multi-physical coupled field with 180° sharp bend is investigated, where the Reynolds number Re and the Hartmann number Ha vary in the respective ranges [0-30000] and [0-3000]. The N-S equation considering Lorentz force is applied to describe the fluid flow and Ohm's Law is adopted to calculate induced electrical current. The energy equation is employed to analyze temperature field. And force equilibrium equation, geometric equation and constitutive relation are used to investigate deformation and stress of structure. The simulation platform is developed based on the finite volume method for fluid solution and finite element method for structure solution. Consistent and conservative scheme is used to guarantee the charge conservation. The effect of magnetic field on flow features such as the length of recirculation bubbles and the formation of vortex around the sharp bend is identified. Both the pattern of flow streamlines in symmetry plane and the secondary flow on the downstream cross-section are affected by magnetic field. The simulation results reveal that the magnetic field has the non-monotonic effect on the flow. For the Interaction number $N>2.5$ cases, the magnetic field is able to stable metal fluid flow, restrain the vortex and decrease the recirculation bubble, which agreed with the conventional understanding about the inhibiting effect of magnetic field. But under the threshold Hartmann number, compared to the corresponding non-MHD case, the magnetic field has the motivation effect on the flow. Besides, physical mechanism of the interaction of multi-physical coupling fields is also analyzed. The flow separation, due to the abrupt change of the sharp bend, has the potential to enhance the heat transfer process. In present work, the parametric regimes where the magnetic field has the positive contribution to the flow are confirmed. Moreover, the temperature distribution in both fluid field and duct structure are illustrated. The functional relationships between the magnetic field magnitude and thermal stress are given. This study

provides some useful information for design of the DCLL blanket and prediction of structure safety.

Keywords: *Multi-physics fields, magnetohydrodynamics, 180° sharp bend*

Corresponding author: *nmzhang@ucas.ac.cn*

Electrochemical Measurements of Blanket Structural Materials' Corrosion in HF-Containing Molten FLiNaK Salt

P1-128

Gaku Yamazaki [a]*, Takuya Nagasaka [a,b], Teruya Tanaka [a,b], Juro Yagi [c], Takashi Watanabe [d], Takuya Goto [e], Akio Sagara [a,b]

[a] *SOKENDAI (The Graduate University for Advanced Studies), Gifu, Japan*
[b] *National Institute for Fusion Science, Gifu, Japan*
[c] *Kyoto University, Gokasho, Uji, Kyoto, Japan*
[d] *Thorium Tech Solution Inc., Tokyo, Japan*
[e] *Doshisha University, Kyoto, Japan*

In order to establish a liquid blanket with a lithium-containing molten fluoride salt, it has been necessary to investigate and control the corrosion of structural materials due to tritium ions, an oxidizing agent bred by neutron irradiation. This study focused on an electrochemical measurement which identified the corrosion mechanism by analyzing current-potential relationship.

A pure iron sample was immersed at 500°C into molten FLiNaK (LiF-NaF-KF: 46.5-11.542.0mol%) salt contained in a graphite crucible. To simulate the tritium breeding, 7×10^{-3} mol/m^{3}s of hydrogen fluoride (HF) (10 times larger than that of the blanket environment) with an argon carrier gas was injected in the FLiNaK salt. A current-potential curve of the sample was obtained by stepwise overpotential scans between -0.4 and +0.4 V vs. a platinum quasi-reference electrode.

Fitting the curve into a Butler-Volmer-type (BV) equation gave the corrosion current density of 14±6 A/m^2. The value was 10 times larger than that without HF gas injection, implying that presence of HF accelerates the corrosion. The current density was constant from -0.4 to -0.2 V vs. the Pt electrode. Since such a constant value is explained by diffusion terms in the BV equation, one of the rate-limiting mechanisms for the corrosion is the diffusion of hydrogen ions in molten FLiNaK salt.

Electrochemical measurements for a main candidate structural material, JLF-1, using a gas with smaller HF concentration, will be performed in order to predict closer corrosion mechanisms to the actual blanket.

Keywords: *corrosion, molten fluoride salts, structural materials, electrochemistry, liquid blankets*

Corresponding author: *yamazaki.gaku@lhd.nifs.ac.jp*

Fabrication of Small Mock-Up and Performance Verification Tests to Verify E-beam Applicability to HCCR Blanket

P1-129

Jae Sung Yoon [a]*, Suk-Kwon Kim [a], Dong Won Lee [a], Eo Hwak Lee [a], Seungyon Cho [b]

[a] *Korea Atomic Energy Research Institute, Daejeon, Korea*
[b] *National Fusion Research Institute, Daejeon, Korea*

Small mock-ups were fabricated using an E-beam weld to verify the manufacturing procedure the HCCR blanket. To establish and optimize welding procedures in E-beam welding using ARAA material, distortion and radiographic tests were carried out from E-beam welding results. Radiographic and pressure tests were performed based on the ASME standards. After an E-beam weld was generated for the small mock-ups, gamma ray radiographic tests were carried out to investigate the joint integrity. The amount of deformation was also investigated using a distortion test dial gauge to confirm deformation after E-beam welding. It should be noted that a small amount of distortion occurred, but the values were small enough to neglect for the fabrication. In addition, a helium leak test and a water pressure test were conducted for verification of the fabricated small mock-ups.

Corresponding author: *jsyoon2@kaeri.re.kr*

Radiation-Induced Defects and Radiolysis Products in 5 MeV Electron-Irradiated Lithium Orthosilicate Pebbles with Various Contents of Lithium Metatitanate

P1-130

Arturs Zarins [a,b]*, Janis Cipa [a,c], Gunta Kizane [a], Arnis Supe [a], Larisa Baumane [a,d], Laima Trinkler [c], Oliver Leys [e], Regina Knitter [e]

[a] *University of Latvia, Institute of Chemical Physics, Riga, Latvia*
[b] *Daugavpils University, Faculty of Natural Sciences and Mathematics, Daugavpils, Latvia*
[c] *University of Latvia, Institute of Solid State Physics, Riga, Latvia*
[d] *Latvian Institute of Organic Synthesis, Riga, Latvia*
[e] *Karlsruhe Institute of Technology, Institute for Applied Materials (IAM), Karlsruhe, Germany*

Advanced two-phase ceramic pebbles, 70 mol% lithium orthosilicate (Li_4SiO_4) and 30 mol% lithium metatitanate (Li_2TiO_3), were recently proposed as tritium breeding material for nuclear fusion reactors. In this research, radiation-induced defects (RD) and radiolysis products (RP) in 5 MeV electron-irradiated Li_4SiO_4 pebbles with various contents of Li_2TiO_3 were analysed by using electron spin resonance, thermally stimulated luminescence and absorption spectrometry. On the basis of the obtained results, it is concluded that the generation mechanism and the structure of RD and RP (except Ti^{3+} centres) in the advanced Li_4SiO_4-Li_2TiO_3 pebbles under action of accelerated electrons are similar to the single phases. Several species of electron and hole type RD and RP were detected, such as E' centres (SiO_3^{3-} and TiO_3^{3-}), HC_1 and HC_2 centres (SiO_4^{3-} and TiO_3^{-}), Ti^{3+} centres, colloidal lithium (Li_n) particles etc. The generation of RD and RP takes place through two stages. In the first stage (fast process), the generated electrons and holes can be self-trapped on intrinsic and extrinsic defects (up to 6 MGy absorbed dose), while in the second stage (slow process), radiolysis take place and atomic rearrangements occur in the crystalline lattice of both phases. The additions of Li_2TiO_3 as a secondary phase in the advanced Li_4SiO_4 pebbles slightly increase the total concentration of the accumulated RD and RP in comparison to the EU reference Li_4SiO_4 pebbles (without additions of Li_2TiO_3). Nevertheless, the advanced Li_4SiO_4-Li_2TiO_3 pebbles have a good radiation stability, and the radiation chemical yield (G) of paramagnetic RD and RP is below 0.8 defects/products per 100 eV. The G value of paramagnetic RD and RP in the reference Li_4SiO_4 pebbles is approx. 0.15 defects/products per 100 eV.

Keywords: *tritium breeding ceramics, lithium orthosilicate, lithium metatitanate, radiolysis*

Corresponding author: *arturs.zarins@lu.lv*

Thermal Hydraulics Activities for the Consolidated HCPB Breeding Blanket of EU DEMO

P1-131

Guangming Zhou *, Qinlan Kang, Francisco A. Hernández

Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany

One of the main objectives of the EU DEMO is to deliver net electricity, which is closely coupled with the thermal hydraulics of breeding blanket. Within EUROfusion Breeding Blanket (BB) Project, Karlsruhe Institute of Technology (KIT) is leading the Helium Cooled Pebble Bed (HCPB) breeding blanket (BB), which is one of the two driver blanket candidates selected for the EU DEMO. In this work, the thermal hydraulics activities supporting the design consolidation of HCPB BB are performed. The thermal hydraulics analyses and optimization based on two typical outboard and inboard blanket unit slices are conducted. Furthermore, thermal hydraulics assessment of full blanket segment taking into account the spatially variable heat fluxes coming from plasma to investigate and optimize the FW design and provide inputs to the structural assessment of full blanket segment.

The results confirmed the soundness of the design from thermal hydraulics point of view and provided inputs for the structural assessment.

Keywords: *EU DEMO, Helium Cooled Pebble Bed, Breeding Blanket, Thermal Hydraulics*

Corresponding author: *guangming.zhou@kit.edu*

Role of Silicate Sorption by Ankerite in Silicic Acid Species Solution

P2-001

Ali AL Dabbas *, Katalin Kopecskó

Department of Engineering Geology and Geotechnics, Faculty of Civil Engineering, Budapest University of Technology and Economics, Budapest, Hungary

The present experimental work is representative to the potential impact of geological deposition conditions on durability of glass used for radioactive waste disposal which could be influenced by iron mineral compositions or other parameters.

Natural ankerite ($CaFe(CO_3)_2$) and master silicate solution contain partially dissolute silica gel are added to deionised water. Temperature and pH values are adjusted at room temperature and 7.0 respectively to achieve dissolved silicate saturation and optimum (SiO_2) concentration in the solution. Sorption of H_4SiO_4 on the carbonatite mineral (ankerite) surfaces is studied. The ankerite is crushed to increase the alteration surfaces for much clear characterization by Scanning Electron Microscopy (SEM) and Energy Dispersive X-ray Spectroscopy (EDS). Aqueous processing is performed by Mass Spectrometry (ICP-MS) and X-Ray Fluorescence (XRF) to determine the cultivation consequences on silicates concentration, and pH meter to measure the acidity of the solution and control silicate precipitation. Aqueous processing is performed after 7, 14 and 30 days of cultivation to study the precipitation principle/role on the ankerite surfaces.

The experimental work is currently running and processing. All experimental results and new approximations will be included in the manuscript.

Keywords: *Ankerite, master silicate solution, silicate sorption*

Corresponding author: *ali.al.dabbas@epito.bme.hu*

Pre-Experiment Analysis for Permeation of Multi-Component Hydrogen Isotopes through Metals in Non-Steady-State

P2-002

Nicolae Bidica *, Anisia Bornea, Ciprian Bucur, Nicolae Sofilca, Marian Curuia, Amalia Soare

National R&D Institute for Cryogenics and Isotopic Technologies - ICSI, Ramnicu Valcea, Romania

The permeation of hydrogen isotopes (especially tritium) through the structural materials of the fusion reactors is an important issue. The isotope effects have to be taken into account in the permeation assessments because both deuterium and tritium are fuel constituents. However, the mutual influence among hydrogen isotopes, when multiple isotopes permeate simultaneously, is still an unresolved issue. We have recently obtained some interesting results showing that the permeation flux of heavier isotopic species is not reduced by the presence of a lighter isotope, but on the contrary it is increased. We have used a non-steady-state model based on classical kinetic theory, for multi-isotope permeation in surface limited regime. The effect of mutual influence was better emphasized over the transition region to steady-state. At steady-state, this effect could be emphasized only if the hetero-isotopic molecular species is present on the high pressure side of the membrane. To experimentally verify these predictions, we propose an installation which includes gas purging facilities, such that a controlled and quantified purge gas will continuously flow on both sides of the membrane. Thus, the changing in the initial isotopic composition of gas at the membrane surfaces could be avoided, and also the boundary conditions could be kept as close to those assumed in the model. This paper describes the experimental installation features and the pre-experiment analysis we performed for several experimental testing scenarios.

Keywords: *Hydrogen, Deuterium, Tritium, Permeation, Multi-Isotope, Isotope Effect*

Corresponding author: *nicolae.bidica@icsi.ro*

Mass Transfer Performance Test of Structured Packings for Tritiated Water Distillation Detritiation

P2-003

Chao Chen *, Jingwei Hou, Jiamao Li, Xiaojun Chen, Chengjian Xiao, Qiang Wang, Yu Gong, Lei Yue, Linjie Zhao, Guangming Ran, Xiaolong Fu, Xiulong Xia, Heyi Wang

Institute of Nuclear Physics and Chemistry, China Academy of Engineering Physics, Mianyang, China

With putting the construction of inland nuclear plants on the agenda, the tritiated water problem becomes increasingly acute. Because the dilution capacity of inland rivers or lakes is weaker than that of ocean. The tritiated effluent from inland nucear plants must be detritiated before discharged into the rivers or lakes. The throughput required by the detritiation facility is more than 500 kg/h. On the other hand, the throughput required by recovering tritium from the coolant of the water-cooling solid breeder (WCSB) banket fusion reactor is assessed to be more than 100kg/h to maintain the coolant tritium level at less than 10 Ci/kg. Due to the throghput is limited, the combined electrolysis catalytic exchange (CECE) process is hardly used to recover tritium from the collant without a pre-concentration facility. With advantages of simple configuration, enormous throughput, simplicity of the design and operation, the absence of explosive, high corrosive and toxic substances, water distillation (WD) offers highly attractive possibility for water detritiation from inland nuclear plants effluent and for the preconcentration process at the front of the CECE process to recover tritium from the coolant of WCSB blanket of fusion reactor.

A tritiated water distillation detritiation experimental facility with 1kg/h capacity is established in Institute of Nuclear Physics and Chemistry, China Academy of Engineering Physics. Height of 12 m bilayer gauze corrugated structured packing is installed in the column with 261 mm inner diameter. Distillation in a total reflux mode for H2O-HTO with tritium activity of 13.5 MBq/kg has been performed for mass transfer performance testing. The height equivalent of theoretical plate (HETP) value is 160 mm and the pressure drop is 167 Pa/m. The data obtained can be used for the design, construction and operation of industrial scale WD detritiation facility.

Corresponding author: *chaochen@caep.cn*

Preliminary Thermal-Hydraulic Analysis of the EU-DEMO Helium-Cooled Pebble Bed Fusion Reactor by Using the RELAP5-3D System Code

P2-004

Salvatore D'Amico [a]*, Xue Zhou Jin [a], Francisco A. Hernández [a], Ivo Moscato [b], Guangming Zhou [a]

[a] *Karlsruhe Institute of Technology (KIT), Institute for Neutron Physics and Reactor Technology (INR), Eggenstein-Leopoldshafen, Germany*

[b] *Dipartimento di Ingegneria, Università di Palermo, Viale delle Scienze, Palermo, Italy*

In view of the development of the EU-DEMO fusion reactor, to fulfil its potential features in terms of low accident hazard and good operational safety, it has to be considered as pivotal to incorporate the needed provisions to improve the overall plant safety and reliability performances as well as to analyse possible mitigation action. To this purpose, within the framework of EUROfusion Safety and Environment actions, an intense research campaign has been launched in order to develop a model, at thermal-hydraulic system code level, for the EU-DEMO Helium-Cooled Pebble Bed (HCPB) Breeding Blanket (BB) concept, aimed at characterizing its response both under normal operational conditions and during accidental scenarios.

The research activity has been focused on the representative and safety relevant cooling loop of the HCPB BB Primary Heat Transfer System, purposely selected by the safety team, in order to assess its thermal-hydraulic behaviour during normal operational conditions (ramp-up/down and steady state). Thereafter the model has been extended in order to investigate the thermal-hydraulic consequences of both an in-vessel and ex-vessel LOCA accidental scenarios and to figure out the capabilities of the mitigation systems intended to withstand such events. Furthermore, the model has proved to be particularly well suited to a fast-track approach for the optimization of certain performance figure of the blanket design allowing a fast estimation during the iterative process.

The research activity has been carried out following a theoretical-computational approach based on the finite volume method adopting the RELAP5-3D system code along with a computational fluid dynamic code, which were properly integrated to achieve a more detailed and realistic simulation of the EU-DEMO reactor thermal-hydraulics.

Models, assumptions and outcomes of this preliminary study are herein presented and critically discussed.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014 – 2018 and 2019 – 2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *EU-DEMO, HCPB, RELAP5-3D, Thermal-hydraulics, Safety and Environment*

Corresponding author: *salvatore.d-amico@kit.edu*

Generation and Transport of Activated Corrosion Products and ^{7}Be in the Lithium Loop of IFMIF-DONES

P2-005

Tamás Dézsi [a]*, S. Hegedűs [b], F.S. Nitti [c], Y. Qiu [d], F. Arbeiter [d], M. Tóth [b], A. Ibarra [e]

[a] *HAS, Wigner Research Centre for Physics / C3D Engineering Ltd., Budapest, Hungary*

[b] *HAS, Wigner Research Centre for Physics, Budapest, Hungary*

[c] *ENEA Brasimone, Camugnano, Italy*

[d] *Karlsruhe Institute of Technology, Karlsruhe (KIT), Germany*

[e] *Centro de Investigaciones Energeticas Medioambientales y Tecnologicas (CIEMAT), Madrid, Spain*

Activated Corrosion Products (ACPs) and ^{7}Be belong to the dominant sources of radiation hazard in IFMIF-DONES Lithium loop, and they determine the occupational radiation exposure during maintenance. A numerical model has been being developed to study the generation and transport of these impurities along the Lithium loop, including the release and deposition from/to the wall of the different components. The model is based on an object-oriented approach, built in Modelica language and using components from Modelica Standard Library. That approach makes it possible to facilitate the setup of complex circuits and study different operational regimes (e.g. loop start-up, shut-down), and complex sub-system behaviors, e.g. different impurity control techniques. The paper describes the structure of the Modelica library being developed, as well as the underlying approach which is used to model the generation and mass transport of the ACPs and ^{7}Be. Furthermore, it summarizes the simulation setup of the DONES lithium loop and the first results obtained.

Acknowledgments: This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and

2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *IFMIF-DONES, Irradiation, Lithium loop, Activated Corrosion Products, Modelica*

**Corresponding author:* *Tamas.Dezsi@c3d.eu*

Exploratory Fire Analysis in DONES Lithium System

P2-006

Danilo Nicola Dongiovanni *, MariaTeresa Porfiri

ENEA, FSN Department, C.R. Frascati, Frascati, Italy

The exploitation of Nuclear Fusion energy in power plants will require the development and qualification of materials able to withstand outstanding neutronic loads for which no operating experience is currently available. Therefore, a Demo-Oriented early NEutron Source (DONES) facility for material irradiation is currently being designed within EUROfusion programme aimed at the production of neutrons with fusion relevant spectrum and fluence by means of D – Li stripping reactions occurring between a deuteron beam impacting a stable liquid lithium flowing film implementing the target. The Lithium System (LS) in DONES shall provide circulation of liquid lithium with suitable thermal-hydraulic characteristics and assure impurity control and heat removal. Given the hazard constituted by liquid lithium inventory and the potential risk of reactions with air eventually resulting into fire events, a preliminary evaluation of the modality of occurrence and evolution of such abnormal events in LS has been performed. In particular, two events have been selected for fire analysis. The first event considers a failure of off-line sampler equipment with water getting in contact with sampled lithium inventory. The initiating event is a failure in the glove box containing the off-line sampler. An air ingress occurs in the glovebox and a break of the Li flask is hypothesized at the same time. The second event assumes a leak at the outlet of the electromagnetic pump with loss of Lithium from the Li pipe in Heat Rejection System in LS room. Simultaneous air ingress is hypothesized as well in LS room (normally filled with inert argon). Fire loads were initially identified for the selected events and the room models developed considering dimensions, lithium inventory and fire compartment assumed to coincide with room limits. The ignition of lithium in contact with air was assumed to occur at liquid lithium operating temperature as reported in most conservative observations in literature and lithium fires were simulated as heat flux associated to lithium – air reactions rates observed in literature. A model for both events was implemented in Fire Dynamic Simulator (FDS) code to evaluate fire dynamics and a sensitivity analysis was performed on relevant inventories in the lithium loop area to investigate possible consequences.

Keywords: *DONES, Lithium, Fire, FDS*

**Corresponding author:* *danilo.dongiovanni@enea.it*

Sensitivity Analysis for the Hydrogen Production during an Ex-Vessel LOCA without Plasma Shutdown for the EU DEMO WCLL Blanket Concept

P2-007

Matteo D'Onorio [a]*, Gianfranco Caruso [a], Maria Teresa Porfiri [b]

[a] *Department of Astronautical Electrical and Energy Engineering (DIAEE), Sapienza University of Rome, Rome, Italy*
[b] *UTFUS-TECN, ENEA CR Frascati, Frascati, Italy*

As for fission power plants, the main environmental and safety issue for the future DEMO fusion reactor is the confinement of radioactive products into the reactor buildings during both normal operation and accidental conditions. Therefore, the possibility of a tungsten-steam reaction during severe accidents is a safety concern because the hydrogen produced from this reaction could pose a flammability or detonation hazard.

In this early development phase of the DEMO design, a limit for the hydrogen inventory has not yet been established, because it will be an outcome of the accident analyses to detect the critical conditions for the vacuum vessel and the contiguous containment volumes. However, several assumptions related to governing parameters, physical phenomena and accident sequence should be made in order to perform detailed accident simulations. Sensitivity analyses have to be performed to predict a wide range of possible code outcomes and to quantify the dependence of hydrogen produced and perturbed input parameters.

The Beyond Design Basis Accident (BDBA) analysis of an ex-vessel LOCA for WCLL blanket concept has been simulated with the fusion version of MELCOR code. The postulated initiating event (PIE) is a double ended break in the FW primary system distributor ring, with simultaneous failure of the plasma control and monitoring system. An in-vessel breach of the coolant system occurs because of first-wall melt-through, with consequent unmitigated plasma shutdown transient. Sensitivity analysis is performed through the RAVEN software tool by varying the decay heat, first-wall failure temperature, tungsten surface available for the chemical reaction, in-vessel and ex-vessel breach size and full closure duration time of isolation valves.

Keywords: *Ex-vessel LOCA, Hydrogen, Sensitivity, RAVEN, MELCOR*

**Corresponding author:* *matteo.donorio@uniroma1.it*

Preliminary Safety Analysis of an In-Vessel LOCA for the EU DEMO WCLL Blanket Concept

P2-008

Matteo D'Onorio [a]*, Gianfranco Caruso [a], Maria Teresa Porfiri [b]

[a] *Department of Astronautical Electrical and Energy Engineering (DIAEE), Sapienza University of Rome, Rome, Italy*
[b] *UTFUS-TECN, ENEA CR Frascati, Frascati, Italy*

In-vessel loss of coolant accident (LOCA) is one of the Design Basis Accident to be considered to support the future DEMOnstration power plant safety assessment.

The water-cooled lithium-lead (WCLL) breeding blanket concept relies on Lithium Lead as breeder, neutron multiplier and tritium carrier. The breeding modules are cooled by two independent pressurized water systems: the fist-wall (FW) and the breeding zone (BZ) coolant systems.

The postulated initiating event (PIE) considered in this analysis is a double ended pipe rupture of the blanket module first wall channels. This event causes the inlet of coolant into the plasma chamber volume triggering an unmitigated plasma disruption and the pressurization of the vacuum vessel volume. The high heat load to FW via conduction/convection by charged thermal particles causes a FW tubes break in the zone affected by the disruption, allowing the entry of an additional steam in the plasma chamber. This accident is analyzed assuming a 72 hours loss of offsite power during which Class III emergency power are not available. Decay Heat Removal (DHR) system, powered by emergency AC generators, feeds the W-PHTS cooling loops for the whole duration of the accident.

The present analysis is performed with the fusion version of MELCOR code (ver. 1.8.6). Two different simulations are performed with the presence and absence of the downstream isolation valves, respectively.

Pressure and temperature transient behavior in the tokamak volumes demonstrate that safety margins are respected during the accidental sequence, even though no external safety system is foreseen or actuated. The chemical reaction between the coolant and the first wall tungsten layer inside the vacuum vessel has also been considered. In order to evaluate radioactive releases towards the environment, the transport model of radioactive products is used in MELCOR. The mobilized radioactive materials are: activated tungsten dust, activated corrosion products (ACP) and tritium, treated in form of tritiated water (HTO).

Keywords: *In-vessel LOCA, DEMO, safety analysis, source term, WCLL*

Corresponding author: *matteo.donorio@uniroma1.it*

Atmospheric Activation in the IFMIF-DONES Accelerator Systems

P2-009

Laura Estévez-Núñez [a]*, Francisco Ogando [a], Francisco Martín-Fuertes [b], M. Martínez [c]

[a] *Department Ingeniería Energética, UNED, Madrid, Spain*
[b] *CIEMAT, Madrid, Spain*
[c] *Empresarios Agrupados, Madrid, Spain*

The International Fusion Materials Irradiation Facility - Demo Oriented NEutron Source (IFMIFDONES) impinges a beam of 40 MeV deuterons into a liquid lithium target producing highly energetic neutrons to test and predict nuclear damage in structural material candidates for the DEMO fusion reactor. Unavoidable streaming from the target, and interactions with scrapers and collimator located along the line of acceleration produce high enough neutron fluxes to activate the atmosphere of the rooms within the Accelerator Systems (AS). An activation study is therefore needed to evaluate the expected doses to workers and members of the public who can be exposed to radiation via inhalation and submersion due to effluents, both in normal and accidental conditions.

This work shows the activity levels derived from the radioisotopes produced by the activation of the atmosphere in the main rooms of the AS. Derived from the activity estimations, effective dose rates are calculated for the members of the public, concluding that the facility complies with the maximum dose limits from liquid and gaseous effluents. In order to satisfy the ALARA principle, alternative filling gases and other mitigation measures are also proposed to minimize the estimated doses.

Keywords: *IFMIF-DONES, Neutronics, Activation, Shielding, Accelerators*

Corresponding author: *lestevez@bec.uned.es*

Numerical Quantification of the Beneficial Effects of Primary Heat Transfer System Isolation Valves in case of In-Vessel Loss-Of-Coolant Accidents for the EU DEMO

P2-010

Antonio Froio [a]*, Luciana Barucca [b], Sergio Ciattaglia [c], Fabio Cismondi [c], Laura Savoldi [a], Roberto Zanino [a]

[a] *NEMO group, Dipartimento Energia, Politecnico di Torino, Torino, Italy*

[b] *Ansaldo Nucleare, Genova, Italy*

[c] *PPPT Department, EUROfusion Consortium, Garching bei München, Germany*

As the first European device to produce electricity from fusion, the EU DEMO Primary Heat Transfer Systems (PHTS) will be consistently larger if compared to present or next-generation tokamaks such as ITER. The consequences of in-vessel Loss-Of-Coolant Accidents (LOCA) would then be more important, and within the EUROfusion Consortium different possible mitigation measures are being investigated. Among these, the introduction of Isolation Valves (IVs) on the main coolant loops of the Breeding Blanket is considered, in view of the many benefits they would introduce, not only in case of accidents, but also e.g. during the maintenance of the in-vessel components. Fast-closing IVs on the PHTS would help not only in relaxing the requirements of the VV pressure suppression system design, but also those related to the expansion volumes that shall accommodate the contaminated coolant discharged from the PHTS after a LOCA

In the present work, the GETTHEM code, a system-level thermal-hydraulic model developed at Politecnico di Torino, is used to quantitatively assess the beneficial effects of the introduction of the IVs. The closing time of the isolation valves and their location are parametrically investigated, keeping into account feasibility constraints and considering both water and helium as PHTS coolants, to evaluate the reduction of the in-vessel average pressure and of the suppression system size. The quantitative results are then critically discussed.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No. 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *DEMO, Vacuum Vessel Pressure Suppression System, in-vessel LOCA, safety, isolation valves*

***Corresponding author:** *antonio.froio@polito.it*

Shut-Down Dose Rate Analysis for the Activated Target Assembly in IFMIF-DONES during Maintenance

P2-011

Stankunas Gediminas [a]*, Yuefeng Qiu [b], Andrius Tidikas [a], Ulrich Fischer [b]

[a] *Lithuanian Energy Institute, Laboratory of Nuclear Installation Safety, Kaunas, Lithuania*

[b] *Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany*

IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO-Oriented NEutron Source) is an accelerator based on d-Li neutron source which aims the qualification of materials at the irradiation conditions of the DEMO fusion power reactor, which is being developed in the frame of EUROfusion Power Plant and Technology (PPPT) programme. The high intense neutron radiation produced in the liquid lithium target results strong activations of the inner Test Cell (TC) components, including the Target Assembly (TA) consisted of the end section of the beam line, the target back-plate and the lithium loop. After shutdown, the TA has to be transfer from the TC to the Access Cell (AC) above within one day. The strong decay gamma dose is important and needs to be estimated for the safety concern of workers and public, as well as the protection of electronic systems.

This paper presents the activation and shutdown dose analyses of the DONES TA. To this end, a series of coupled transport and activation calculations are performed using McDeLicious code, which is based on the MCNP Monte Carlo code for simulating the d-Li reactions, and the FISPACT inventory code. The latest DONES TC model has been employed and FENDL-3.1 neutron cross-section data library has been used. Shut-down dose rate calculations were performed using the R2Smesh code system developed at KIT. The obtained dose rate maps in the AC as well as the shielding assessments will be presented in the paper.

Acknowledgments: This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *neutronics, shut-down dose rate, IFMIF-DONES*

***Corresponding author:** *gediminas.stankunas@lei.lt*

Diamond Fast-Neutron Detector Applied to the KSTAR Tokamak

P2-012

Youngseok Lee *, Jong-Gu Kwak, Yong-Un Nam, Si-Woo Yoon, Hee-Soo Kim, Gap-Rai Park

KSTAR Research Center, National Fusion Research Institute, Daejeon, Korea

Neutron spectrometer based on a diamond fast-neutron has applied to detect and determine the DD neutrons energy and flux detector for neutron diagnostics at the KSTAR tokamak. For the work, the diamond fast-neutron detector based spectrometer has placed on a viewport where was located in the midplane of port P outside the vacuum vessel of KSTAR. The energy and flux measurements of deuterium-deuterium (D-D) fusion neutrons during KSTAR D-D plasmas operations have performed with the diamond based neutron spectrometer.

The resent results show not only satisfactory performance of a diamond detector in a harsh environment caused by mixed a neutron-gamma high radiation field, and a strong electromagnetic field, but also applicability as a neutron spectrometer for fusion reactors.

Keywords: KSTAR, diamond fast-neutron detector, neutron spectrometer, DeuteriumDeuterium(D-D)/Deuterium-Tritium (D-T) fusion neutron energy, neutron flux, neutron diagnostics

**Corresponding author:* *yslee@nfri.re.kr*

Numerical Analysis of Steam Condensation at Sub-Atmospheric Pressure in Water Suppression Tank

P2-013

Rosa Lo Frano [a]*, Alessio Pesetti[a], F. Venturi[b], D. Aquaro[a]

[a] *University of Pisa, lg L. Lazzarino, Pisa, Italy*
[b] *ITER Org., St Paul Lez Durance Cedex, France*

Particular attention in fusion reactor safety is given to the Vacuum Vessel Pressure Suppression System (VVPSS) that is a key component, operating at sub-atmospheric pressure, to protect the Vacuum Vessel (VV) from over-pressuring events like, e.g., the Loss Of Coolant Accident (LOCA), namely Ingress of Coolant Event (ICE).

The aim of the paper is to assess and compare the main thermal-hydraulic parameters, monitored in the small-scale test facility built at the DICI- University of Pisa in order to investigate steam condensation at the sub-atmospheric condition. The experimental data obtained from the broad experimental performed are compared with the numerical results of RELAP/SCDAP code: such a type of investigation represents a novelty for the scientific literature (first study in the matter).

In particular, the capability to describe the correct steam flow, logic system controls and the direct contact condensation (DCC) inside Condensation Tank (CT) are investigated. Being peculiar the physical conditions foreseen in a fusion reactor during the Loss Of Coolant Accident (LOCA), namely Ingress of Coolant Event (ICE), and/or Loss Of Vacuum Accident (LOVA), a proper nodalization is prepared and suitable simulation techniques are implemented to represent the axial temperatures distribution inside the CT during the transient behaviour. To the purpose, 50 g/s steam is injected at about 150°C through the sparger holes; the upstream pressure is initially fixed based on the tank water level.

Numerical results, in terms of average temperatures, mass flow rates and pressures are provided for different steam condensation regimes (chugging, condensation oscillation, etc.). They could be in turn used to determine the strength capacity of the VVPSS.

Keywords: ITER; VVPSS; DCC; sub-atmospheric pressure; RELAP/SCDAP

**Corresponding author:* *rosa.lofrano@ing.unipi.it*

Progress in Development of Advanced D1S Dynamic for Three-Dimensional Shutdown Dose Rate Calculations

P2-014

Giovanni Mariano [a,b]*, Davide Flammini [a], Nicola Fonnesu [a], Fabio Moro [a], Rosaria Villari [a]

[a] *ENEA, FSN Department, Frascati, Rome, Italy*
[b] *DIAEE, Sapienza University of Rome, Rome, Italy*

An accurate evaluation of the shutdown dose rate is fundamental for shielding design, materials' requirements, licensing and planning of maintenance operations in high performances fusion devices. The calculation of the shutdown dose rate requires to combine radiation transport and inventory codes. The Advanced Direct 1-Step (AD1S) is one of the most validated in complex fusion tokamak machines. Developed by ENEA and based on MCNP5 Monte Carlo and FISPACT inventory codes, it has unique computation capabilities, such as the mesh tally maps and continuous time evolution assessment of the dose rate in a single run. The Direct 1-Step approach foreseen a single transport simulation both of neutrons and decay gammas; photons are treated as promptly emitted and weighted by correction factors to account for build-up and decay of the considered radionuclides. Advanced D1S has been validated through measurements at JET tokamak and comparison with Rigorous 2-Step codes and it has been also extensively used for ITER and DEMO calculations. The Advanced D1S Dynamic is an improved version of Advanced D1S with new features aimed at extending

applications, at overcoming present limitations and at improving its versatility. The most relevant improvements are the possibility to change the machine configuration during irradiation and at the shutdown (i.e. variation of the machine geometry during operations and at the shutdown, management of multiple lifetimes and decay gamma source portability), handling of libraries in different formats and treatment of multi-step reactions. New Python utilities have also been developed to optimize the MCNP-FISPACT interface and data analyses. The present work describes the developments of Advanced D1S Dynamic and its first applications to tokamak calculations.

Keywords: *Shutdown dose rate, D1S, neutronics, activation, code development*

Corresponding author: *giovanni.mariano@uniroma1.it*

Use of Phenomena Identification and Ranking Tables Technique in the Fusion Safety Assessment

P2-015

Dobromir Panayotov *

Fusion for Energy (F4E), Barcelona, Spain

Phenomena identification and ranking tables (PIRT) is a systematic way of gathering information on a specific subject, and ranking the importance of the information, in order to meet some decisionmaking objective, e.g., determining the highest priority. Originally formulated to support the US NRC best estimate licensing option in late 80s it demonstrated to be powerful and robust tool to establish safety analysis computer code phenomenological requirements. PIRT is practical and flexible technique that offers a holistic approach to the safety assessment. In recent years the value of the PIRT process has been recognized outside the nuclear safety community as an important component of any validation process. PIRT technique could offer significant advantages in the fusion safety for example to: identify, categorize, and characterize the phenomena and issues relevant to the risk and safety; prioritize research activities; inform decisions regarding the development analytical tools; define the course of accident sequences and safety system success criteria; establish technical basis and cost effective organization for new programs; and provide insights for the review of safety analysis and supporting data bases.

The PIRT applications to the following fields related to the accidental analyses: i) experimental programs; ii) codes' development; iii) code assessment and; and iv) uncertainty evaluation will be overviewed. The application of PIRTs for the definition of the fusion accident analysis specifications, assessment and selection of analysis code(s), development and qualification of the codes models and uncertainty evaluation will be illustrated on the example of EU Test Blanket Systems. Based on the lessons learnt from the adaptation of fission methods the presentation will draw on availability of tools for accidents analyses, use of PIRT, the verification and validation by means of separate and integral effect tests and establishing benchmark problems as well on code assessment and development of multi-physics, multi-fluids integrated code systems.

The work reported in this paper has been performed by F4E. The views expressed in this publication are the sole responsibility of the author and do not necessarily reflect the views of the Fusion for Energy and the ITER Organization as nuclear operator**.** Neither Fusion for Energy nor any person acting on behalf of Fusion for Energy is responsible for the use, which might be made of the information in this publication. This paper does not commit the nuclear operator – ITER IO.

Keywords: *Safety, Accident analyses, Fusion safety, DEMO, Fusion power plant*

Corresponding author: *dobromir.panayotov@f4e.europa.eu*

Experimental and Numerical Analysis of Sub-Atmospheric Steam Condensation in Suppression Tank with SIMMER IV Code

P2-016

Alessio Pesetti *, Rosa Lo Frano *, Donato Aquaro, Francesco Venturi

University of Pisa DICI, Pisa, Italy

One of the key safety components for nuclear fusion plants is the suppression tank, which is designed to protect the Vacuum Vessel (VV) against accidental pressurization events, e.g. the Loss Of Coolant Accident (LOCA).

In this framework the attention is focused on the Vacuum Vessel Pressure Suppression System (VVPSS), made of water tanks in which the pure steam, or eventually mixed with incondensable gases, is injected; consequently the overpressure is dumped profiting of Direct Contact Condensation (DCC). The design constraints of fusion reactor dictates that the pressure resulting (long-term) from any accidental or baking condition should be always kept lower than the atmospheric pressure.

The study of the phenomena evolving during DCC in LOCA conditions is the major novelty, especially in consideration of the lack of similar studies in the available literature.

In this context, a wide series of experimental tests was carried out at Pisa University, Department of Civil and Industrial Engineering (DICI), in a Small Scale Test Facility (SSTF), designed and instrumented for investigating DCC at sub-atmospheric pressure, by varying water pool temperature, pressure and steam mass flow rate.

The adoption and assessment of suitable numerical codes to reliably simulate, such a cutting-edge multiphase multicomponent scenario, have a crucial role for contributing to the phenomena understanding and for possible safety analysis of full-scale components. On this basis, a post-test analysis was carried out with three-dimensional SIMMER IV code. The comparison of numerical results and experimental data highlighted the suitability of the adopted code in simulating DCC (i.e. pressure and temperature time trends in the condensation tank) at sub-atmospheric conditions.

Moreover, SIMMER IV code simulated the steam plume dynamics consistently with video recordings.

Keywords: *sub-atmospheric pressure, DCC, Vacuum Vessel, multiphase flow, SIMMER code*

Corresponding author: *alessio.pesetti@for.unipi.it, rosa.lofrano@ing.unipi.it*

Safety Assessment for European DEMO – Achievements and Open Issues in View of a Generic Site Safety Report

P2-017

Maria Teresa Porfiri [a]*, Neill Taylor [b], Sergio Ciattaglia [c], Xue Zhou Jin [d], Jane Johnston [b], Bethany Colling [b], Tim Eade [b], Dario Carloni [b], Tonio Pinna [a], Egidijus Urbonavicius [e], Robert Vale [b], Andrija Volkanovski [f], Gianfranco Caruso [g], Wolfgang Hering [d], Robert Stieglitz [d]

[a] *ENEA UTFUS-TECN, Frascati (Roma), Italy*
[b] *Culham Centre for Fusion Energy, Culham Science Centre, Abingdon, UK*
[c] *EUROfusion Consortium, Garching, Germany*
[d] *Karlsruhe Institute of Technology (KIT), Institute of Neutron Physics and Reactor Technology (INR), Eggenstein-Leopoldshafen, Germany*
[e] *Lithuanian Energy Institute, Lithuania*
[f] *Jožef Stefan Institute, Ljubljana, Slovenia*
[g] *Department of Astronautical Electrical and Energy Engineering, Sapienza University of Rome, Rome, Italy*

The way to arrive at a licensing phase for a nuclear fusion installation is not straightforward mainly because of the lack of operating experience and of dedicated nuclear regulations for this type of technology. In fact only small/medium fusion experimental facilities exist with limited licensing processes and only one large experiment, ITER, has obtained a construction license. Therefore the safety assessment and the preparation of the preliminary safety report are mandatory for a first kind of DEMOnstration power plant. (DEMO). A Generic Site Safety Report (GSSR) has begun taking advantage of experience of fission power plants, and considering lessons learned from the ITER safety studies. Preparation of a Generic Site Safety Report (GSSR) will require some years to be completed. However currently, at the starting point, the structure and content of GSSR is clear and well defined. This paper will deal with major safety issues that are addressed in the GSSR because, in the frame of the EUROfusion Work Programme for DEMO from 2014 up to 2018, they have been identified and should be finalized in the future. These for example include the safety requirements for the plant and the systems, the tools to be used for the safety assessment and the procedures for the selection of the reference accidents. While resolution of many of these issues is well developed, some require further effort to meet the anticipated licensing requirements that may be applied to DEMO. A complete spectrum of Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs) that can determine the risk of releases from the main systems of the plant is still in a working phase. The safety classification of most of the Structures, Systems and Components (SSCs), the feasibility and analyses of some accident mitigation systems are going on.

The GSSR will also address the issue of radioactive materials arising from the operation of a future DEMO fusion power plant. A strategy for managing the materials during the lifetime of the plant and after decommissioning will be covered in the GSSR. The major issues involved will be outlined together with the justification for the proposed treatment methods.

This paper presents the results of the quantification, when possible, of the gap between the results achieved and the goals established in the plant guidelines.

The assessed effort required in terms of studies, experiments and human resources to reach a good stage for a successful DEMO licensing will be presented.

Keywords: *DEMO, Safety, Generic Site Safety Report, Requirement, Accident Analysis*

Corresponding author: *mariateresa.porfiri@enea.it*

Shutdown Dose Rate Calculations for the IFMIF-DONES Lithium Loop Cell Using Variance Reduction Techniques

P2-018

Yuefeng Qiu *, Ulrich Fischer, Frederik Arbeiter

Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany

IFMIF-DONES (International Fusion Material Irradiation Facility-DEMO Oriented NEutron Source) is an accelerator-based neutron irradiation facility which aims at providing the irradiation data required for the construction of a DEMO fusion power plant. High energy neutrons are produced in DONES by a 125 mA beam of deuterons accelerated to

40 MeV impinging on a liquid lithium target. The lithium loop also provides a high-speed lithium flow for removing the beam power from the target assembly placed in the Test Cell (TC). The lithium loop cell (LLC) located below the TC is foreseen to be maintained one day after shutdown. Therefore, it is required to estimate the shutdown dose rate at this location for radiation safety considerations.

Accordingly, this work is devoted to the shutdown dose rate analysis of the LLC based on the use of the R2Smesh code system developed at KIT. During the maintenance, the lithium is drained out of the loop, thus the inlet and outlet pipe penetrating to the TC are empty. The strong gamma ray streaming from the inner TC to the LLC is very challenging for the shutdown dose calculation. A variance reduction technique based on the ADVANTG weight window approach has been employed for the decay gamma transport calculation, which enabled a reliable estimation of the shutdown dose in the LLC. Recently, a new version of the R2Smesh code was developed to provide more advanced capabilities. Its verification has been performed through the comparison with the current version of R2Smesh on the LLC calculation, and reasonable agreements of the decay gamma sources have been obtained.

Acknowledgments: This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *IFMIF-DONES, shutdown dose, MCNP, variance reduction, lithium loop cell*

Corresponding author: *yuefeng.qiu@kit.edu*

Ionizing Radiation Monitoring Requirements at the Divertor Tokamak Test Facility

P2-019

Sandro Sandri [a]*, Maurizio Angelone [b], Gian Marco Contessa [a], Manuela Guardati [a], Rosaria Villari [b]

[a] *ENEA, Institute of RadiationProtection, Frascati, Rome, Italy*
[b] *FSN Department, ENEA, Frascati, Rome, Italy*

The Divertor Tokamak Test facility (DTT) is an experimental nuclear fusion machine, based on a tokamak configuration, now in advanced design phase. DTT will operate with D-D plasmas producing pulses lasting about 60 seconds and producing a neutron yield up to more than 10^{17} n/s. Due to the high neutron yield, activation of the structures and residual radioactivity after shut-down will be present. As a consequence DTT will need a robust radiation protection system to prevent undue worker and population exposure. The main objectives of this system are:

- to keep the radiation doses as low as practically possible and below the relevant dose limits;
- to ensure acceptably safe and satisfactory radiological conditions in the workplaces;
- to keep records of monitoring for the purposes of regulation and good practice.

All of the above points must be met during normal operation, maintenance, decommissioning and in emergency situations, by maintaining a number of safety features, among which the radiation monitoring system is probably the most crucial. This paper presents the general requirements of this monitoring system, giving some details about the preferred characteristics for the involved devices and monitors. The monitoring system shall be composed by portable instruments, passive ambient dosimeters and a limited number of fixed active measuring stations, suitable to detect and record the appropriate dosimetric quantities both for neutrons and photons. The active measuring stations and the passive dosimeters will form two monitoring networks designed in such a way to cover the area of interest to radiological survey around DTT. In this paper the requirements for the two networks and for the portable measuring devices will be described.

Keywords: *Divertor Tokamak Test facility, Radiation Measurements, Radiation Monitoring*

Corresponding author: *sandro.sandri@enea.it*

Consideration of the Low Contamination Device for the Measurement of Tritium Release Rate

P2-020

Toshiharu Takeishi [a]*, Kazunari Katayama [b], Yoshiya Kawabata [a]

[a] *Department of Applied Quantum and Nuclear Engineering, Graduate School of Engineering, Kyushu University, Fukuoka, Japan*
[b] *Interdisciplinary Graduate School of Engineering Science, Kyushu University, Kasuga, Japan*

In the evaluation of the tritium dynamic characteristics and safety for the nuclear fusion fuel cycle, tritium trapping amount or releasing rate for the component material of the fusion reactor are important values. When the tritium release rate is experimentally obtained, a preliminary blank test is required because of tritium trapping and releasing arise not only on the specimen but on the inner surface of piping and apparatus. The authors reported that tritium contamination on the inner surface of the tritium handling device can be effectively decreased by adding water vapor to decrease the tritium abundance ratio at the surface. As an application of this result, the authors devised the tritium release rate measurement apparatus for the fusion materials equips a tritium releasing chambers and a trapping chamber. Since water vapor is added to the releasing chamber, tritium released from the surface of the material is hardly trapped on the inner wall of piping and apparatus.

The present report shows an outline of the apparatus and preliminary experimental results conducted to obtain the amount of tritium on the inner surface under various conditions. In the preliminary measurement, the amount of tritium remaining on the inner surface of the stainless steel pipe was compared after flowing dry HTO and HTO with water vapor, respectively. When water vapor was added to flowing HTO, the amount of residual tritium on the piping inner surface was less than a few % of the case where water vapor was not added. It can be expected that the measurement of releasing rate with the present apparatus is not affected by tritium trapping at the inner surface of the apparatus and does not require a preliminary blank test.

Keywords: *tritium, contamination, releasing rate*

Corresponding author: *take@nucl.kyushu-u.ac.jp*

Activation Analysis of the European DEMO Divertor with Respect to the Different Breeding Blanket Segmentation

P2-021

Andrius Tidikas *, Gediminas Stankūnas

Lithuanian Energy Institute, Kaunas, Lithuania

Neutron activation is an unavoidable process in large scale fusion devices operating with deuterium and tritium fuels. Neutron activation leads to the production of radioactive materials. Activated materials can generate heat and ionizing radiation in their environment. Consequently, values of activation characteristics need to be determined in order to ensure safety and performance of the fusion devices.

This paper presents the analyses performed within the SAE (Safety and Environment) project of EUROfusion/PPPT regarding the activity and decay heat inventories for the DEMOnstration power plant (DEMO) 2015 baseline model divertor. Operation scenario of 1898 days was considered as well as three different breeder blanket configurations of DEMO reactor: single-module segmentation water cooled lithium lead (WCLL SMS), multi-module segmentation water cooled lithium lead (WCLL MMS) and helium cooled pebble bed (HCPB). Divertor model is divided into 62 segments and 4 layers with different material compositions. Neutron transport and activation inventory calculations were performed respectively with MCNP and FISPACT codes. In addition to the activation calculation results, divertor radionuclide analysis is also presented in this work.

Keywords: *DEMO, divertor, neutron activation, breeding blanket, MCNP, FISPACT*

Corresponding author: *Andrius.Tidikas@lei.lt*

The Flow Instability Phenomenon in Loss of Coolant Accident of Water Cooling Blanket

P2-022

Dianle Wang, Changhong Peng, Yun Guo*

Department of Engineering and Applied Physics, School of Physical Sciences, University of Science and Technology of China, Hefei, China

The application of fusion energy is considered to be the best way to solve future energy crisis. In the current fusion reactor design, the blanket transfers the energy generated by the fusion to the steam turbine for power generation. Thus, the accident of the blanket is a subject of great concern to scientists. In this paper, the RELAP5 model of water cooling blanket is reasonably established, and then the sensitivity of node number is analyzed. The flow instability phenomenon in loss of coolant accident (LOCA) of the blanket is found in the analysis procedure. In the case of double-end shearing break of the flow channel and within a certain area of the break, obvious instability occurs. According to the results such instability man be caused by the reverse flow phenomenon. In the case of small break LOCA, the blanket may be dried and damaged earlier because of thermal stress induced by huge flow fluctuation. This study has certain reference significance for the safety design of water cooling blanket.

Keywords: *Water cooling blanket, LOCA, RELAP5, Flow instability*

Corresponding author: *guoyun79@ ustc.edu.cn*

Influence of Different Equivalent Methods of Pulsed Neutron Irradiation on ACPs Source Term Calculation in Fusion Reactor

P2-023

Jingyu Zhang [a]*, Sheng Fang [b,c], Qingyang Guo [a], Bin Zhang [a], Yixue Chen [a]

[a] *School of Nuclear Science and Engineering, North China Electric Power University, Beijing, China*
[b] *Institute of Nuclear and New Energy Technology, Tsinghua University, Beijing, China*
[c] *Key Laboratory of Advanced Reactor Engineering and Safety, Ministry of Education, Beijing, China*

Fusion reactors are mostly designed to run in pulsed condition, and pulsed neutron irradiation is important to source term calculation of activated corrosion products (ACPs), which is the dominant radiation hazard of water-cooled loops in fusion reactors. Considering that the quantity of pulses is huge, some equivalent methods of pulses are usually adopted to improve the computational efficiency, but the computational accuracy of different equivalent methods has not been fully evaluated for ACPs source term. In this paper, three typical equivalent methods of pulses are performed in the ACPs source term analysis code CATE 2.0, which are steady state method (SS), equivalent steady state method (ESS), and continuous pulse method (CP). And the influence of different equivalent methods on ACPs source term is analyzed based on the preliminary design scheme of water-cooled blanket of China Fusion Engineering Test Reactor (CFETR). The calculation results show that: (1) compared with ACPs radioactivity of the actual pulses, the computational error of SS is largest, followed by ESS, whose values both exceed 10%; (2) the computational error of CP is small only when enough pulses are reserved in the end; (3) specific to different nuclides, the computational accuracy of equivalent methods is closely related to the nuclides half-lives, and the computational error of long-lived nuclides (Co-58, Co-60, etc.) is very small, while the computational error of short-lived nuclides (Mn-56, Ni-65, etc.) is large, even bigger than an order of magnitude. The above research can provide support for the engineering design and radiation protection of fusion reactors.

Keywords: *Pulsed neutron irradiation, Equivalent method, ACPs source term, Fusion reactor, CFETR*

Corresponding author: *poptnt@163.com*

Consideration on Defence-In-Depth Applied in CFETR

P2-024

He Zhang *, Yijie Qian, Jiqiang Su

China Nuclear Power Engineering Corporation, Beijing, China

For fission reactors, defense in depth as a top-level design principle has important guiding significance for reactor design. Analogous to the fusion reactor, it also needs to optimize the system design from the perspective of defense in depth and improve the safety of the design. This paper attempts to analyze the existing China Fusion Engineering Experimental Reactor (CFETR) system design from the perspective of the defense-in-depth principle which is widely used in the fission reactor, and discusses the analysis results and proposes some suggestions for the defense-in-depth principle of the fusion reactor.

Corresponding author: zhhcnpe2014@163.com

Development of Li, Na, K, Rb and Cs Thermionic Ion Sources Using SiC Block Heater Technology

P2-025

Gábor Anda *, Sándor Bató

Wigner RCP, Association EURATOM, Budapest, Hungary

At Alkali Beam Emission Spectroscopy [1] and at Atomic Beam Probe [2] (or also called imaging heavy ion beam probe [3]) systems the ion emission capacity of the ion source is critical to obtain reasonable information about fusion plasmas. The common way to heat up a thermionic ion emission material is resistive, usually using Molybdenum or Tungsten filaments. The filaments are embedded into alumina to increase their lifetime, but the power density and the temperature are strongly limited to about150 W/cm^2 and 1200 °C.

A new method is developed for ion source heating based on Silicon Carbide (SiC) volume heater. The used SiC disc is also heated resistively, but the power density and temperature limit is much higher, about 500W/cm^2 and 1400 °C. Additionally, the heating system is very robust, resists immediate power shutdowns or abruptly increased heating power, as well.

The production method of the ion emission material surface is also improved. The conventionally used β-Eucryptite (or Spodumen) and melting technique limited the ion current density to about 1.5 – 2mA/cm^2 (at Lithium ion sources). With small modifications of the constituent and the production, it was enhanced to about 3-3.5mA/cm^2 (at Lithium ion sources). These modifications can be used similarly at the mentioned alkali ion sources, as well.

Two different sizes of the ion sources were manufactured, with 14 and 19 mm diameter (1.5 and 3 cm^2) reaching up to 5/10 mA ion current (at Lithium), but theoretically the size of the ion sources shall be increased arbitrarily.

In this paper the SiC block heater and the production technology of the emission surface of the alkali ion sources are described. Ion emission capacity of Li, Na, Rb and Cz sources are also presented.

[1] *Development of a high current 60 keV neutral lithium beam injector for beam emission spectroscopy measurements on fusion experiments, G. Anda et al, Review of Scientific Instruments, 89, 013503 (2018)*

[2] *Micro-Faraday cup matrix detector for ion beam measurements in fusion plasmas, D. Réfy at al, submitted to Review of Scientific Instruments*

[3] *Hardware design and beam modelling of the imaging heavy ion beam probe diagnostic at ASDEX Upgrade, G. Birkenmeier at al, ECPD 2019, poster*

Keywords: *thermionic alkali ion sources, plasma diagnostic*

****Corresponding author:*** *anda.gabor@wigner.mta.hu*

RAMI Evaluation of the Beam Source for the DEMO Neutral Beam Injectors

P2-026

Piero Agostinetti [a]*, Thomas Franke [b,c], Ursel Fantz [c], Christian Hopf [c], Nicolas Mantel [d], Minh Quang Tran [e]

[a] *Consorzio RFX, Euratom-ENEA Association, Padova, Italy*
[b] *EUROfusion Consortium, Power Plant Physics and Technology (PPPT), Garching, Germany*
[c] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[d] *Culham Centre for Fusion Energy, Science Centre, Abingdon, UK*
[e] *Ecole Polytechnique Federale de Lausanne (EPFL), Swiss Plasma Center (SPC), Lausanne, Switzerland*

DEMO is a first-of-a-kind DEMOnstration Fusion Power Plant [1] and is intended to follow the ITER experimental reactor. The main goal of DEMO will be to demonstrate the possibility to produce electric energy for the grid from the fusion reaction early in the second half of the century. The injection of high energy neutral (1 MeV) particle beams is one of the main tools to heat the plasma up to fusion conditions, control the plasma burn phase and ramp the plasma down.

Within the EUROfusion Framework a conceptual design of the Neutral Beam Injector (NBI) for the DEMO fusion reactor is currently being developed. Thereby, Reliability, Availability, Maintainability and Inspectability (RAMI) have to be taken into consideration for the conceptual design of the DEMO NBI, together with the exploitation of the currently available return of experience from the ITER NBIs.

A novel design of the beam source for the DEMO NBI [2] has been developed by Consorzio RFX between 2016 and 2018, featuring multiple sub-sources and following a modular design concept, with each sub-source featuring its expansion chamber and radio frequency driver. In view of RAMI analysis, this design approach has been compared to the beam source of the ITER heating and current drive NBI, which features a single expansion chamber with eight RF drivers [3]. This study is based on experience gained during operation of ITER relevant sources. Comparing the failure risk of the two different source concepts due to the considered failure modes has allowed for further design developments aiming at exploiting the advantages of the modular approach while minimizing its drawbacks.

[1] *T. Donné, Journal of Instrum. 2017, Vol. 12, https://doi.org/10.1088/1748-0221/12/10/C10008.*

[2] *P. Sonato et al., Nucl. Fusion, vol. 57 (2017) 056026.*

[3] *Marcuzzi et al., Rev. Sci. Instrum. 87 (2016) 02B309.*

Keywords: *RAMI, evaluation, beam source, negative ions*

****Corresponding author:*** *piero.agostinetti@igi.cnr.it*

Measurement of Delayed Neutron Emission from Water Activated by 14 MeV Neutrons in a FW Mock-Up of ITER

P2-027

Maurizio Angelone [a]*, Mario Pillon [a], Stefano Loreti [a], Andrea Colangeli [a], Giuseppe Mazzitelli [a], Paolo Del Prete [a], Rosaria Villari [a], Jonathan Naish [b], Chantal Nobs [b], Lee Packer [b]

[a] *ENEA C.R. Frascati, Department of Fusion and Technology for Nuclear Safety and Security, Frascati, Italy*
[b] *Nuclear Technology Department, Culham Centre for Fusion Energy, Culham Science Centre, Abingdon, UK*

Water activated by 14 MeV neutrons emits both high energy gammas from ^{16}N and delayed neutrons from ^{17}N. The isotopes $^{16,17}N$ are produced through the $^{16}O(n,p)^{16}N$ ($T_{1/2}$= 7.13 s) and $^{17}O(n,p)^{17}N$ ($T_{1/2}$ = 4.14 s) reactions, respectively. These sources of radiation represent one of the main safety issues for ITER having large impact on the schedule, commissioning and licensing of the machine. To validate the methodology for water activation assessment used for ITER and to reduce the safety factors applied to the calculations results a water activation experiment is ongoing at the 14 MeV Frascati neutron generator (FNG). The experimental set-up consists of a closed water loop, about 40 m long, where the cooling water, while transiting through an ITER FW mock-up, is irradiated by the D-T neutrons produced by FNG. Gamma-rays (6.73 MeV) from ^{16}N and delayed neutrons from ^{17}N are measured using a large CsI gamma-ray detector and an array of calibrated ^{3}He detectors, respectively. Calculations are performed using the MCNP Monte Carlo code with FENDL3.1 data library as well as the FISPACT-II inventory code.

The paper reports about the measurements of the delayed neutrons emission. The experimental procedure, including the calibration of ^{3}He detectors performed at the National Institute of Metrology of Ionizing Radiation (INRIM) of ENEA, the experimental results and the comparison with the calculation prediction are reported and discussed.

Keywords: *ITER, Water Activation, Delayed neutrons, FNG, Fusion neutronics*

**Corresponding author:* *aurizio.angelone@enea.it*

Taking to Service and First Results of the Q-PETE/D2 Hydrogen Permeation Setup

P2-028

Frederik Arbeiter, Dmitry Klimenko, Volker Pasler, Georg Schlindwein, Axel von der Weth

Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany

The Q-PETE hydrogen permeation setup has been developed to (i) provide validation cases for DEMO HCPB (Helium Cooled Pebble Bed) breeder-zone relevant cases of hydrogen permeation, and (ii) to derive hydrogen transport properties of the purge gas contacting structural materials like ferritic-martensitic 9%-Cr steels and austenitic steels. The experiment features two gas-purged chambers separated by a membrane (made from the material under test), an experimental control periphery (with supply of hydrogen-containing feed gas and hydrogen-free sweep gas), as well as a quadrupole mass spectrometer for time-resolved measurement of hydrogen species concentration. Analysis of the time-dependent concentration signals provides insight to the involved hydrogen transport processes.

In the reported first series of experiments, a 1.2 mm thick membrane of X2 CrNiMo 17-12-2 (316L) austenitic steel was used, which was also the material of the permeator chambers. Experiments were performed at several temperatures and purge gas conditions in the blanket-relevant range. The experimental results (time constants, permeation levels and species distribution) are discussed and compared to numerical simulation results. An iterative procedure was applied to derive the hydrogen transport properties for the used austenitic steel.

Acknowledgments: This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *Hydrogen, Tritium, Breeder, Permeation, Diffusion, Solubility, Experiment*

**Corresponding author:* *frederik.arbeiter@kit.edu*

Application of BES Synthetic Diagnostics for the Study of SOL Filament Dynamics on the EAST Tokamak

P2-029

Ors Asztalos [a*], A.H. Nielsen [b], S. Zoletnik [c], B. Tal [d], N. Yan [e], G.H. Hu [e], G.I. Pokol [a,c], EUROfusion-IM Team [f]

[a] *Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary*
[b] *Department of Physics, Technical University of Denmark, Kgs. Lyngby, Denmark*
[c] *Wigner Research Center for Physics, Budapest, Hungary*
[d] *Max Planck Institute for Plasma Physics, Garching, Germany*
[e] *Institute of Plasma Physics, Chinese Academy of Science, Hefei, China*
[f] *See author list at http://euro-fusionscipub.org/eu-im*

Fluctuation beam emission spectroscopy (BES) is an active plasma diagnostic used for density measurements which has sufficient spatial and temporal resolution for the study of turbulent density fluctuations and associated flows. A high energy neutral beam consisting of hydrogen isotopes or light alkali metal atoms is shot into the plasma. Through various collisional processes with plasma particles, the beam atoms get excited and the photons originated from their spontaneous emitted is collected by an observation system.

RENATE is a fluctuation BES modelling code, featuring 3D beam and 3D observation geometry modelling capabilities as well as accounting for the underlying magnetic geometry, thereby incorporating all relevant spatial artefacts of the diagnostic. Time-dependent density and temperature fluctuations are taken as input, provided by 2D fluid model, HESEL, used to study interchange dynamics in the SOL. Flux tube expansion of turbulent structures within the beam geometry allows for 3D modelling of the synthetic diagnostic signal.

In present contribution, the statistical properties of synthetic BES signals are discussed. Filament frequencies, amplitudes, sizes and velocities are acquired from synthetic BES signals. Effect of BES specific

artefacts on filament detection and overall dynamics is studied. Emphasis was placed on the atomic physics processes, smearing information along the beam; filament detection dependence on noise added to synthetic signals as well as filament size and velocity dependence on the alignment of magnetic field lines to the lines of sight within the measurement volume. The HESEL code was run in a Kepler workflow, developed within the EUROfusion Integrated Modelling framework. A workflow for passing HESEL fluctuation data via integrated data structures to the RENATE BES code will be discussed. A comparison of synthetic LiBES SOL observations with the corresponding experimental observations in L-mode plasmas was conducted on EAST tokamak data, showing a reasonable agreement of modelled and experimental SOL turbulence dynamics.

Keywords: *BES, filament, SOL, turbulence*

****Corresponding author:*** *asztalos@reak.bme.hu*

Parameter Study and Dynamic Simulation of Current DEMOnstration Intermediate Heat Transfer and Storage System Design via MATLAB/Simulink

P2-030

Maria-Victoria Bologa [a]*, Robert Stieglitz [a], Wolfgang Hering [a], Evaldas Bubelis [a], Joel Oliveira [b]

[a] *Karlsruhe Institute of Technology, Institute for Neutron Physics and Reactor Technology, Eggenstein-Leopoldshafen, Germany*
[b] *Kraftanlagen Heidelberg GmbH, Heidelberg, Germany*

Nuclear fusion, as a real sustainable and renewable energy source, offers significant benefits like limitless fuel reserves and inherent nuclear safety. Due to pulsed operation of the DEMOnstration Fusion Power Plant (DEMO FPP), a steady energy supply, as demanded by energy grids, is not compatible with the pulsation of the TOKAMAK plasma. Hence, effective decoupling of the contradicting requests is necessary, realized by an intermediate system comprising a thermal energy storage system (ESS).

A detailed design of that Intermediate Heat Transfer and Storage System (IHTS) for DEMO FPP is developed. The IHTS concept includes a two-tank direct storage system with HITEC molten salt as a heat transfer fluid. A review of the results of research and development of the IHTS with a parameter study for the specific DEMO IHTS design was carried out. The DEMO IHTS design model was simulated in stationary state via EBSILON® Professional code. The simulation program MATLAB / Simulink was used for the evaluation of DEMO dynamic behavior parameters such as frequent transitions and thermal losses. Based on thermo- and fluid dynamic principles, the dynamic IHTS design model was successfully validated by MATLAB / Simulink. In order to attain stable power output during the burn and dwell phases, the instabilities arising during the burn phase in the hot and cold tanks were evaluated. As a function of time, the operation parameters such as temperature, filling level and thermal losses in the tanks were optimized. Next steps are an extension of IHTS model by a power conversion system (PCS), definition of the interfaces and coupling of PCS with a corresponding generator model, also further simulation of the completed system with SIMULINK.

Keywords: *DEMOnstration Fusion Power Plant (DEMO FPP), Energy Storage System (ESS), EBSILON® Professional, MATLAB/Simulink, Power Conversion System (PCS)*

****Corresponding author:*** *maria-victoria.bologa@kit.edu*

On the Vertical Uniformity of an ITER-Like Large Beam

P2-031

Federica Bonomo *, Isabella Mario, Dirk Wünderlich, Ursel Fantz and the NNBI Team

Max-Planck-Institut für Plasmaphysik, Garching, Germany

The main challenges for the ITER Neutral Beam Injection (NBI) system is to combine high extracted current densities (329 A/m^2 for H, 286 A/m^2 for D) with a limited amount of co-extracted electrons (electron-ion ratio < 1) for pulses up to 1 hour together with a small beam core divergence (< 7 mrad) and with an high beam uniformity of better than 90 % over the whole beam area (1 m × 2 m). This will allow to deliver 16.5 MW of 1 MeV D^0 (and 870 keV of H^0 in the first ITER operation phase) into the fusion plasma for each of the two beam lines.

The ELISE test facility is foreseen to demonstrate the feasibility of achieving the ITER NBI requirements in terms of currents for pulses of up to 1 hour at the required source filling pressure of 0.3 Pa. ELISE is equipped with an ITER-like 3-grid system and has half the size of the ITER NBI source in the vertical direction (1 m × 1 m). The whole beam results formed by two (instead of four as for the ITER NBI) *"beam segments"*, each one hosting 4 rectangular beamlet groups.

Several beam diagnostics allow for a deep insight of the properties of the large beam in terms of beam segments. Since beam inhomogeneity is expected mostly in vertical direction due to the usage of a horizontal magnetic filter field, the focus is on the vertical uniformity of the accelerated current among the two beam segments. This is measured via IR analysis of the surface temperature of the diagnostic calorimeter. Beam divergence (from Beam Emission Spectroscopy) and beam segment widths at the calorimeter are monitored too. Some knobs have been identified to modify and improve the vertical homogeneity in terms of accelerated current (via RF power) and beam divergence (via biasing the grid system with respect to the source potential); other key parameter affecting the vertical uniformity are still under investigation, such as different magnetic fields configuration in front of the grid system. Significant cases will be discussed, with proposal given for possibilities to improve the vertical homogeneity in large beams.

Keywords: *Neutral Beam Injection, ELISE, beam, beam diagnostics, homogeneity, ITER*

**Corresponding author: federica.bonomo@ipp.mpg.de*

Nuclear Analyses in Support of the Conceptual Design of the DTT Tokamak Neutron Diagnostics

P2-032

Barbara Caiffi [a]*, M. Angelone [a], D. Flammini [a], F. Moro [a], M. Tardocchi [b], R. Villari [a]

[a] *ENEA, Department of Fusion and Nuclear Safety Technology, Frascati, Rome, Italy*

[b] *Istituto di Fisica del Plasma "P. Caldirola", CNR, Milano, Italy*

The Divertor Tokamak Test (DTT) facility, whose design phase is currently under finalization, is an Italian project aimed to investigate alternative power exhaust solutions for DEMO. It is designed to operate with significant power loads and enough flexibility to test innovative divertor configurations, different plasma edge and bulk conditions approaching, as much as possible, those planned for DEMO. DTT will be equipped with a set of diagnostics for real time control of the plasma parameters and for machine protection. For neutron diagnostics a set of neutron yield monitors, based upon fission chambers, and a multi-channel neutron camera, equipped with liquid scintillators and/or diamond detectors, to investigate the variation of the neutron emission profiles along collimated lines of sights, are foreseen.

The present paper presents preliminary study performed in support of the DTT neutron diagnostic systems design. A detailed MCNP model representing a 20° sector of the machine integrating its main components and detectors' assemblies has been developed and used for these studies. Three-dimensional neutron and gamma transport simulations have been carried out by means of the MCNP Monte Carlo code coupled with the FENDL nuclear data libraries in order to assess the expected neutron and gamma flux spectra, fluences and doses inside the detectors. Simulation of detectors' response, assessment of the performances and design optimization study including shielding have been performed as well. The outcomes of this analysis provide detectors' requirements, guidelines for the development of the above-mentioned diagnostics, investigating their feasibility and suitability with the neutron emissivity foreseen for the DTT operational scenarios.

Keywords: *DTT, MCNP, neutronics, diagnostics*

**Corresponding author: barbara.caiffi@enea.it*

Design, Manufacturing and Tests of the LIPAc High Energy Beam Transport Line

P2-033

B. Brañas [a], Jesus Castellanos [a]*, O. Nomen [b], C. Oliver [a], F. Arranz [a], F. Aragón [a], D. Duglue [c], J. A. Ferreira [a], D. Iglesias [a], D. Jiménez-Rey [a], J. Marroncle [d], A. Mas [a], L. Melón [a], J. Mollá [a], F. Ogando [e], I. Podadera [a], P. Sauvan [e], A. Soleto [a], F. Toral [a]

[a] *CIEMAT, Madrid, Spain*

[b] *IREC, Sant Adrià de Besòs, Barcelona, Spain*

[c] *Fusion for Energy, BA/IFMIF, Garching, Germany*

[d] *CEA, Gif-sur-Yvette, France*

[e] *Departamento de Ingeniería Energética, ETSII-UNED, Madrid, Spain*

The International Fusion Materials Irradiation Facility (IFMIF) is a projected accelerator-based, D-Li neutron source for DEMO materials qualification. LIPAc is an accelerator aiming to reach 125 mA, 9 MeV continuous wave deuteron beam, which is currently being commissioned in Rokkasho (Japan) with the final objective of validating the IFMIF accelerator design.

In the LIPAc accelerator, a 10 m long High Energy Beam Transport line (HEBT) will connect the exit of the superconducting linac to the beam dump. The HEBT line must accommodate the diagnostics for beam characterization and open the beam at its exit to allow its stopping at the beam dump. The beam line contains quadrupoles to control the beam shape and its trajectory maintaining beam losses below 1W/m so that activation of surrounding elements is limited and hands-on maintenance is allowed.

The manufacturing and procurement of the main components of the HEBT line has been completed in 2018 and the systems have been shipped to the LIPAc site in Rokkasho. Special attention was given to the materials selection, with the aim of guaranteeing the performance of the different components under the radiation environment as well as minimizing their activation.

In this work, the whole project will be described since its origins. A summary of the beam dynamic calculations and other studies (vacuum, radioprotection, assembly, alignment) that led to the conceptual design of the line will be presented. After that, the detailed design of the line, including that of the vacuum chambers and supports will be described, justifying the main design decisions taken. Finally, the manufacturing and procurement process and the acceptance tests performed will be summarized.

Keywords: *IFMIF, accelerator, magnets, vacuum, alignment*

Corresponding author: *jesus.castellanos@ciemat.es*

Virtual DEMO for Korean Fusion Program

P2-034

Jae Min Kwon, Hyunseok Kim, Hansoo Chang, **Hyun-Kyung Chung** *

National Fusion Research Institute, Daejeon, Korea

Aiming to demonstrate the feasibility of fusion electricity generation, K-DEMO is an important step toward a commercial fusion reactor in Korea. Due to physical uncertainties of fusion core, the design of K-DEMO has relied heavily on empirical scalings while entailing many issues ranging from engineering difficulties and uncertainties to excessive construction costs. Therefore, it is highly desirable to find new ways to optimize a K-DEMO design based on better understanding of physics and engineering. With the rapid advances of supercomputers and artificial intelligence (AI) technologies, we propose to develop a virtual DEMO design that combines comprehensive fusion core and tokamak simulations with power plant simulators. We present a plan and strategies to develop Korean Virtual DEMO by reviewing the past and on-going researches on fusion simulation in Korea. A connection between the Virtual DEMO and other Korean fusion programs e.g. KSTAR and ITER etc is discussed. The development will initially involve a whole device modelling of KSTAR-sized tokamak plasma and its validation with KSTAR experiments. Then the physics and engineering elements for burning plasma will be later added to realize a Virtual DEMO eventually as the final goal. Potential roles of AI technologies such as neural-net based machine learning in accelerating fusion simulation will be also discussed.

Keywords: *tokamak simulation, DEMO design, machine learning, modeling*

Corresponding author: *hkchung@nfri.re.kr*

Analysis of Stress Induced by Plasma Disruption on Vacuum Vessel through Multi-physics Modeling

P2-035

Valter Cocilovo [a]*, R. Fresa[b]

[a] *ENEA-FSN-FUSTEC, c.r.e. Frascati, Frascati, Roma, Italy*
[b] *Scuola di Ingegneria- Università della Basilicata, Potenza, Italy*

The analysis of the stresses induced on the vacuum vessel and its internal components by the plasma instabilities, such as disruptions, also following a plasma Vertical Displacement Event (VDE), are one of the major effort in mechanical design of a tokamak machine, because of the great number of possible variations in inputs and parameters set as basis of the dimensions design. So the availability of a fast tool for evaluating different design options, also able to perform parametric analysis, is necessary to define the project requirements and to verify the conceptual design. To respond to this need a methodology based on the multi-physics modeling capability offered by the Comsol® software was developed in the frame of the Italian DTT project. It consists in coupled simulations carried out through the data sharing (one way) between 2D axisymmetric and 3D models where two poloidal sections, with and without ports, are modeled. In the first 2D axisymmetric model, a poloidal section without ports, the magnetic and electric fields generated by the VDE and disruption are calculated imposing the plasma time evolution as input, not self-consistently calculated. In the 3D models only the mechanical structures are present and on them the E&M fields are extruded, determining so the eddy currents diffusion in the passive conductors. The mechanical loads are the Lorentz's forces obtained by the cross product of the 3D eddy currents and the 2D axisymmetric poloidal magnetic field and are imposed as body loads on the mechanical structures. Then a linear stress analysis is carried out after the constraints assignment. The same procedure is followed with the 2D model complete of the ports to check the overall correctness of the model by comparing some proper global physical values, such as the reaction forces and moments and the total elastic strain energy. In this work the methodology is presented by reporting the simulation of a double null plasma VDE, lasting c.a. 100 ms, followed by a full plasma current (5.5 MA) in c.a. 40 ms.

Keywords: *Stress Analysis, Vacuum Vessel, VDE, Disruption, Tokamak*

Corresponding author: *valter.cocilovo@enea.it*

3D SURO Modelling of Beryllium Erosion and Deposition on Tungsten Rough Surfaces under ITER-Relevant Plasma Conditions

P2-036

S.Y. Dai [a]*, M. Wang [a], Donghong Liu [a], Q. Shi [b], B. Cui [c], S.G. Liu [a], D.Z. Wang [a]

[a] *Key Laboratory of Materials Modification by Laser, Ion and Electron Beams (Ministry of Education), School of Physics and Optoelectronic Technology, Dalian University of Technology, Dalian, China*
[b] *Graduate School of Engineering, Nagoya University, Nagoya, Japan*
[c] *School of Science, Dalian University of Technology, Panjin, China*

The impurity erosion-deposition distribution on rough divertor of fusion device is inhomogeneous due to surface roughness, with a high erosion or small deposition on protruding parts of rough surface, and a lower erosion or larger deposition at the far side of ridges and at the bottom of recessions. This phenomenon is observed in different tokamak facilities, which may finally lead to a smoothing of initially rough surface. This effect of long pulse and steady state operation is difficult to study in present tokamak facilities, but predictive modelling can be performed to investigate it.

The three-dimensional (3D) Monte-Carlo code SURO [1-3] has been developed to study the impurity erosion and deposition on rough surfaces under ITER-relevant plasma conditions. The properties of background plasma and impurity near the divertor target are studied by the 1D ParticleIn-Cell Monte-Carlo collision (PIC-MCC) code SDPIC, which are used as the input data for SURO code. The SURO code uses the test particle approach to describe the bombardment of background plasma and the deposition of impurity particles on the 3D surface topography. The dynamic change of surface topography as well as surface concentrations of different species due to erosion and deposition are taken into account in SURO, which has a very good flexibility for treating the process of material mixing. In this study, the beryllium impurity erosion-deposition on the tungsten substrate of divertor target relevant to ITER has been studied by SDPIC/SURO modelling. The detailed analysis of the impact of the background beryllium flux on the rough surface evolution is conducted. The areal densities of the background beryllium deposition on rough tungsten substrate are calculated with different beryllium fluxes. The effects of the rough surface topography on the beryllium erosion and deposition are discussed.

[1] *Shuyu Dai, Shengguang Liu, Jizhong Sun, et al., J. Nucl. Mater. 463, 372-376, (2015).*

[2] *Shuyu Dai, A. Kirschner, Jizhong Sun, et al., Nuclear Fusion, 54, 123015 (2014).*

[3] *Shuyu Dai, A. Kirschner, et al., Plasma Physics and Controlled Fusion,55,055004 (2013).*

Keywords: *rough surface, impurity transport, Monte-Carlo*

Corresponding author: *daishuyu@dlut.edu.cn*

Beam Formation and Transport in the BATMAN Upgrade Test Facility

P2-037

Niek den Harder *, Federica Bonomo, Andrew Hurlbatt, Christian Hopf, Ursel Fantz and the NNBI Team

Max-Planck-Institut für Plasmaphysik, Garching bei München, Germany

The neutral beam heating in ITER is designed to supply 16.5 MW of injected power per beam box, which requires 40 A of accelerated D^- current at 1 MeV. In order to achieve this, an accelerated D^- current density of 200 A/m^2 is needed, distributed homogeneously over the 1280 apertures. The required current density has been demonstrated in the test bed BATMAN, although in a source with fewer and smaller apertures. BATMAN has been upgraded with an ITER-like grid system with 5 x 14 apertures (ø 14 mm), and a diagnostic suite which consists of a tungsten wire calorimeter close to the grids, a diagnostic calorimeter further downstream, and two beam emission spectroscopy viewing arrays. In this contribution, the influence of spatial variations of the extracted current density on the beam formation and transport is studied.

Discharges with varying current density and deviation from uniformity were performed in BATMAN Upgrade by scanning the RF-power at different magnetic field strengths. Beam emission spectroscopy measurements were used to benchmark IBSimu single aperture extraction calculations with the ITER-like grid system. The vertical current density profile over the grid system was estimated by integrating the beamlet power density profiles in the tungsten wire calorimetry and by comparing the perveance curves for the different beam emission spectroscopy lines of sight. If the beam formation is understood, projecting the beamlets towards the calorimeter should reproduce the measured profile when transport effects are small. For uniform discharges the profile is well matched, but the shift is underpredicted, because of beam deflection in the magnetic field between grid system and calorimeter as seen in more advanced beam transport codes such as BBCNI. When the beam emission spectroscopy and tungsten wire calorimetry show that the vertical current density profile becomes non-uniform, the different intensity and divergence per beamlet need to be taken into account to correctly describe the beam profile and qualitative beam shift behaviour.

Keywords: *Neutral Beam Injection, Ion-optics, IBSimu, Tungsten Wire Calorimetry, BATMAN*

**Corresponding author: Niek.den.Harder@ipp.mpg.de*

Test Series D Experimental Results for SIMMER Code Validation of WCLL BB In-Box LOCA in LIFUS5/Mod3 Facility

P2-038

Marica Eboli [a]*, Samad Khani Moghanaki [b], Nicola Forgione [b], Riccardo Maria Crugnola [c], Alessandro Del Nevo [a]

[a] *ENEA FSN-ING-PAN, Camugnano, Italy*
[b] *DICI – University of Pisa, Pisa, Italy*
[c] *Department of Energy, Politecnico di Milano, Milano, Italy*

The in-box LOCA (Loss of Coolant Accident) is a relevant safety issue for the design of the WCLL BB design. Research activities are ongoing to master phenomena and processes occurring during the postulated accident, to enhance the predictive capability and reliability of numerical tools, and to validate computer models, codes and procedures for their applications. Current status of knowledge requires the availability of qualified and reliable experimental data to support these activities. In view of this, the new separate effect test facility LIFUS5/Mod3 has been commissioned and an experimental campaign has been designed (Series D). The main objective of the tests are the generation of reliable experimental data for the validation of the modified version SIMMER codes for fusion application. Moreover, the data will be also used to investigate the dynamic effects of energy release on the structures and to provide relevant feedbacks for the follow up experimental campaigns.

The experiments of the series D test matrix are focused on the validation of the chemical model of SIMMER code. Therefore, focus is given to the relevance of parameter ranges relevant for the WCLL BB and the model object of the validation, as well as, to parameter ranges suitable for the reliable quantification of the test data. A pre-defined amount of water is injected into the reaction tank at a pressure of 155 bar and different liquids temperatures, accordingly with the selected test matrix. The experimental data and results of the executed tests (i.e. pressures, temperatures, amount of injected water, and hydrogen production quantification) are reported and critically discussed.

Keywords: *Chemical Reaction, Code Validation, Test Experiments, LIFUS5/Mod3, WCLL breeding blanket*

**Corresponding author: marica.eboli@enea.it*

Thermal Fatigue Tests on Functionally Graded W/EUROFER-Layer Systems in a Newly Constructed Testing Apparatus

P2-039

Thomas Emmerich [a], Robert Vaßen [b], Jarir Aktaa [a]

[a] *Karlsruhe Institute of Technology (KIT), Institute for Applied Materials, Eggenstein-Leopoldshafen, Germany*
[b] *Forschungszentrum Jülich GmbH (FZJ), Institute of Energy and Climate Research, Jülich, Germany*

For future fusion reactors it is envisaged to protect First Wall components, made of reduced activation ferritic martensitic steel, against the plasma by tungsten coatings because of its favorable thermo-mechanical properties. By implementing functionally graded material layers between the coating and steel substrate the difference in coefficient of thermal expansion can be compensated. Respective layer systems were successfully produced by vacuum plasma spraying and tested, among other aspects, in regard to thermal fatigue for 500 cycles in a vacuum furnace. For future fusion reactors a higher number of thermal cycles are anticipated and therefore, a less time consuming approach for thermal fatigue testing is required.

Hence, a new testing apparatus was constructed to achieve shorter heating and especially cooling times, while preventing also sample oxidation, and is presented in this work. Furthermore, first thermal fatigue experiments on different functionally graded tungsten/steel-layers systems are carried out and their microstructural behavior is investigated.

Keywords: *tungsten, first wall, functionally graded material (FGM), vacuum plasma spraying, thermal fatigue*

**Corresponding author: Thomas.Emmerich@kit.edu*

Development Progress of Functionally Graded W/EUROFER-Layers for First Wall Components

P2-040

Thomas Emmerich [a], Jörg Rey [b], Bradut-Eugen Ghidersa [b], Robert Vaßen [c], Jarir Aktaa [a]

[a] *Karlsruhe Institute of Technology (KIT), Institute for Applied Materials, Eggenstein-Leopoldshafen, Germany*
[b] *Karlsruhe Institute of Technology (KIT), Institute for Neutron Physics and Reactor Technology, Eggenstein-Leopoldshafen, Germany*
[c] *Forschungszentrum Jülich GmbH (FZJ), Institute of Energy and Climate Research, Jülich, Germany*

For future fusion reactors it is planned to use tungsten as protective coating on First Wall components, made of reduced activation ferritic martensitic steel, due to its favorable thermo-mechanical properties. Functionally graded material layers, inserted between the tungsten coating and steel substrate, have been recently developed to compensate the large difference in the coefficient of thermal expansion between these two materials.

An overview of functionally graded W/EUROFER-layer systems, successfully produced by vacuum plasma spraying on laboratory scale will be given in this paper as well as their achieved properties and current development status with regard to the transferability to First Wall components. Special attention is paid to the challenges of a limited temperature window during coating, to not exceed the tempering temperature of the steel. Particularly in case of more complex substrate structures, like First Wall Mock-ups, temperature monitoring and supportive finite element simulations are utilized. Furthermore, several techniques for cleaning and inspection of the layer systems after fabrication are evaluated. Finally, first results of testing coated First Wall Mock-ups in HELOKA and production of the layer systems on industrial scale will be presented and discussed.

Keywords: *tungsten, finite element method (FEM), first wall, functionally graded material (FGM), vacuum plasma spraying*

Corresponding author: *Thomas.Emmerich@kit.edu*

Pre-Analysis of the WCLL-Mock Up Neutronics Experiment at the Frascati Neutron Generator

P2-041

Davide Flammini [a]*, M. Angelone [a], A. Colangeli [a,b], U. Fischer [c], N. Fonnesu [a], G. Mariano [d], F. Moro [a], R. Villari [a]

[a] *ENEA, Department of Fusion and Nuclear Safety Technology, Frascati, Rome, Italy*
[b] *SBAI Department, Sapienza University of Rome, Rome, Italy*
[c] *Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany*
[d] *DIAEE Department, Sapienza University of Rome, Roma, Italy*

The accuracy of the prediction in calculating relevant nuclear quantities such as Tritium Production Rate (TPR) and shielding capability of a breeder blanket is a key requirement for the design of a fusion power reactor. Experimental validation of these nuclear quantities is fundamental to assess the reliability of numerical tools and nuclear data and the related uncertainties. The Water Cooled Lithium Lead (WCLL) is one of the breeding blanket concept candidate for the future European Demonstration Fusion Power Reactor (DEMO). In the frame of the EUROfusion Consortium, a WCLL mock up neutronics experiment will be performed at the 14 MeV Frascati Neutron Generator (FNG) irradiation facility, devoted to the validation of the TPR and the shielding capability predicted by neutronics design calculations. It will include the assessment of the uncertainties on the calculated TPR and the neutron/gamma attenuation which will be useful to evaluate design margins required to guarantee tritium self-sufficiency and to prove the predicted shielding efficiency. In preparation of the future experiment, a detailed pre-analysis has been performed with the MCNP Monte Carlo code for the definition of the mock-up assembly and for the optimisation of the experimental setup. The WCLL mock-up has been designed in such a way that the essential nuclear features of the WCLL in DEMO can be represented in the experiment and the relevant nuclear quantities can be measurable with sufficient accuracy under FNG irradiation conditions, with several experimental techniques (scintillation techniques in Li-ceramic pellets, activation foils, diamonds detectors and spectrometers, e.g. ^{3}He). This study reports the results of the pre-analysis: the optimised experimental layout and its representativeness of WCLL DEMO blanket.

Keywords: *DEMO, WCLL, Breeding Blanket, Neutronics, TPR, Benchmark, MCNP*

Corresponding author: *davide.flammini@enea.it*

Shutdown Dose Rate Studies for the TT and DTE2 Campaigns at JET

P2-042

Nicola Fonnesu [a]*, R. Villari [a], P. Batistoni [a], U. Fischer [b], P. Pereslavtsev [b] and JET Contributors [c]

EUROfusion Consortium, JET, Culham Science Centre, Abingdon, UK
[a] *ENEA, Department of Fusion and Nuclear Safety Technology, Frascati, Rome, Italy*
[b] *Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Karlsruhe, Germany*
[c] *See the author list of "X. Litaudon et al 2017 Nucl. Fusion 57 102001"*

The EUROfusion Work Package JET3 programme, established to enable the technological exploitation of the future Deuterium-Tritium (DT) operations at JET over the next years, includes, within the NEXP subproject, a novel Shutdown Dose Rate (SDR) benchmark experiment. The measurement of the SDR due to neutron activation in a fusion machine operating with Deuterium and Tritium is of primary importance for planning its operation in respect of dose limits for the external radiation exposure. The next high-performance DT campaign at JET (DTE2) which will follow a Tritium-Tritium (TT) operation phase, is a unique opportunity to validate the numerical tools for ITER shutdown dose rate analysis, through the comparison between numerical predictions and measured quantities in terms of C/E (Calculation/Experiment) ratios. Within this framework, this work is a pre-analysis of the impact of the future TT and DTE2 campaigns on the dose rate. Threedimensional simulations are performed with the Advanced D1S (Direct 1-step) and the R2Smesh (Rigorous 2-step) approaches to predict the shutdown dose rate levels at the considered experimental positions and the dependence on the JET operation conditions. The results are presented and discussed in the paper with the major objective to contribute to the optimization of the planned SDR benchmark experiments.

Keywords: *JET, Deuterium Tritium Campaign, DTE2, TT, Shutdown Dose Rate*

Corresponding author: *nicola.fonnesu@enea.it*

Development of Small Specimen Test Technologies on Joints of Fusion Structural Materials in SWIP

P2-043

Haiying Fu *, Pengfei Zheng, Liwen Zhang, Ran Wei, Jiming Chen

Southwestern Institute of Physics (SWIP), Chengdu, China

Welding of structural materials is essential to construction and safe operation of fusion reactors. Usually, new material will be produced after welding; the material in the weld metal of a joint should be different from that in the base metal. Considering bonding properties evaluation especially neutron irradiation properties maybe involved in the future, similar to structural materials themselves, small specimen test technologies (SSTTs) are also important for the joints of fusion structural materials. In Southwestern Institute of Physics (SWIP), several SSTTs have been developed for bonding properties evaluation, such as tensile, shear, bending, and small punch test technologies. Specific fixtures have been designed to fit for the different kind of tests above-mentioned. Joint of CLF-1 reduced-activation ferritic/martensitic steel after electron beam welding and post-weld heat treatment is utilized for the bonding properties evaluation. Specimens with different sizes and shapes are used to study the effects on bonding properties such as strain, stress, and stress triaxiality, etc. By combining experiments and finite element method simulation, effects of the different kinds of SSTTs on bonding properties are investigated. Relationships between them will be tried to reveal. Finally, suitable SSTTs will be selected for evaluation of bonding properties for fusion structural materials.

Corresponding author: *fuhy@swip.ac.cn*

RELAP5-SIMMER-III Code Coupling Development of PbLi-Water Interaction

P2-044

Francesco Galleni [a]*, Daniele Martelli [b], Marica Eboli [b], Alessandro Del Nevo [b], Samad Khani Moghanaki [a], Sandro Paci [a], Nicola Forgione [a]

[a] *University of Pisa, DICI, Pisa, Italy*
[b] *ENEA FSN-ING C.R. Brasimone, Camugnano, Italy*

A major safety issue in the Water-Cooled Lead-Lithium Breeding Blanket (WCLL-BB) system foreseen for fusion reactor is the interaction concerning the primary coolant (water) and the neutron multiplier (PbLi), due to a rupture in the coolant circuit. This scenario involves an exothermic chemical reaction between PbLi and water with the production of hydrogen, in addition to critical interactions in a complex multiphase system in non-thermal equilibrium.

In recent years the PbLi/water reaction was successfully implemented in the SIMMER-III code and validated against data from the LIFUS5 experimental campaign. However – due to limitations of SIMMER-III – this work was restricted to the prediction of the phenomena inside the vessel, neglecting the simulation of the injection line. Nevertheless, since the injection line may actually have an important effect on the development of the transient, the simulation of the whole facility would be highly desirable. Indeed, University of Pisa recently developed a coupling methodology between the SIMMER-III and RELAP5/Mod3.3 codes and applied it to simple single-phase cases.

In this paper the complete simulation of the LIFUS5/Mod3 facility is presented, with the injection line modelled through RELAP5. Furthermore, all the complex aspects of the phenomena inside the vessel were included: the multiphase system and the interaction between water and PbLi with the chemical reaction and the production of hydrogen modelled by SIMMER.

Preliminary results are presented, showing that the coupling methodology can be effectively employed for the prediction of the chemical and thermal-hydraulic behaviour of complex loop experimental facilities.

Keywords: *Coupling codes, SIMMER-III code, RELAP5 code, WCLL, PbLi*

Corresponding author: *francescog.galleni@dici.unipi.it*

Development of an On-Line Sensor for Hydrogen Isotopes Monitoring in Flowing Lithium at DONES

P2-045

Belit Garcinuño *, Joaquín Mollá, David Rapisarda, Marta Malo, Jesús Herranz, Ángel Ibarra

CIEMAT, Madrid, Spain

In the frame of EUROfusion project, large R&D activities are being performed for the development of DONES. Since impurities enhance the erosion/corrosion effect, one of the challenging aspects on the development of the neutron source is the purity of the lithium. The main process in DONES is the deuteron impact on the lithium target. As consequence, the liquid lithium is contaminated by deuterium. Protium and tritium are also produced in Li as sub-product of the reaction. The requirement imposed to protium and deuterium content in Li is 10 wppm. On the other hand, the radiotoxicity of tritium imposes even more severe requirements to its control in the plant. The reference requirement for tritium is 1 wppm but any achievable value below that limit will have an important impact in safety related systems, safety procedures, licensing and in the end in the cost of the plant. Thus, there is a need to monitor the content of these isotopes during the operation of the plant. The reference solution for monitoring is the chemical analysis off-beam by means of Li specimens extracted from the loop. An on-line monitoring of the tritium inventory in the loop would be preferable from the operation point of view and much more acceptable from safety point of view. There is no availability of commercial sensors that could work under the Li environment; therefore, in this work the development of an on-line sensor based on the permeation against vacuum technology is presented. Its design is conditioned by the materials employed, needing a high permeability and compatibility with liquid lithium. The main goal is to design a device which maximizes the contact area between membrane and liquid by keeping the vacuum volume at minimum to reduce the response time and improve the detection limit.

Keywords: *DONES, lithium purity, hydrogen isotopes, permeation against vacuum*

Corresponding author: *belit.garcinuno@ciemat.es*

Application of the Best-Estimate Model Calibration and Prediction through Experimental Data Assimilation Methodology to the Tests Performed on a Helium Cooled First Wall Mock-Up

P2-046

Belit Gonfiotti *, B. Ghidersa, M. Ionescu-Bujor, X. Z. Jin, R. Stieglitz

Karlsruhe Institute of Technology, Karlsruhe, Germany

As part of the general strategy for the qualification of the DEMO design in view of licensing and operation, experiments are under progress at the Karlsruhe Institute of Technology (KIT). These experiments are part of the program on the qualification of components and on the validation and calibration of the RELAP5-3D models reproducing such components.

In 2018, thermal-hydraulic investigations have been performed looking into the transient response of a helium-cooled blanket First Wall (FW) mock-up under LOFA conditions. The results obtained through these investigations captured the quick temperature rise on the surfaces exposed to heat loads (such as those coming from the plasma).

The outcomes of this experimental activity are the basis of the present work, which describes the validation and calibration of the FW mock-up RELAP5-3D model using the Best-Estimate Model Calibration and Prediction through Experimental Data Assimilation methodology. This methodology is a rigorous procedure based on the maximum entropy principle and Bayes' theorem to compute best-estimate predictive results: it assimilates the computed results and the experimental data together with their uncertainties to provide "best-estimate" responses (i.e. computed results) and parameters (i.e. input deck data). In a previous work, the same methodology was used to calibrate a RELAP5-3D model of the HELOKA-HP electrical heater under normal operation. In this regard, the present work marks the first application of this methodology against incidental (fast) transients in the field of fusion reactors.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *First Wall mock-up, Best-estimate, Pressure drops, LOFA*

****Corresponding author:*** *bruno.gonfiotti@kit.edu*

Residual Ion Energy Recovery for the DEMO NBI – a Conceptual Design Study

P2-047

Christian Hopf [a]*, Giuseppe Starnella [a], Niek den Harder [a], Piero Agostinetti [b], Ursel Fantz [a]

[a] *Max-Planck-Institut für Plasmaphysik, Garching bei München, Germany*

[b] *Consorzio RFX, CNR, ENEA, INFN, Università di Padova, Padova, Italy*

The requirements for a neutral beam injection system for DEMO go significantly beyond those on ITER, particularly regarding availability and energy efficiency. The latter is mostly limited by the neutralisation efficiency, which at energies around 1 MeV is only about 55 % with a gas neutraliser like on ITER. Besides the technically challenging alternative by photoneutralisation – with a theoretically possible neutralisation efficiency of up to 100 %, but undemonstrated at any relevant scale – residual ion (RI) energy recovery (ER) has been proposed as a means of improving energy efficiency with a less efficient neutraliser. For this purpose the negative and positive RIs are first deflected out of the neutral beam in opposite directions, and subsequently decelerated by an electrically biased collector. The possible gain in energy efficiency depends not only on the fraction of the kinetic energy to which the RIs can be decelerated while still being effectively collected, but also on the additional neutral beam losses by reionisation due to the additional beamline length. We present a CAD model of the conceptual design of an energy recovery system integrated into a DEMO beamline with ITER-like parameters and beam shape. We use detailed 3D ion optics simulations to study charged particle trajectories, taking the effects of finite beamlet divergence and space charge into account. Heat loads on beamline components and transmission losses are an output of these simulations as is the beamline's energy efficiency gain. The dependence of the efficiency gain on a variety of design parameters, such as the neutralisation efficiency, which could e.g. be mildly increased with a plasma neutraliser, are studied using a simpler 0D model, in order to show under which conditions the integration of ER is economically attractive.

Keywords: *Neutral Beam Injection, Energy Recovery, Neutraliser, Efficiency, Plasma Heating*

****Corresponding author:*** *christian.hopf@ipp.mpg.de*

Improved Characterisation of ITER-Relevant Large Negative Ion Beams through Forward Modelling of their Diagnostics

P2-048

Andrew Hurlbatt *, Niek den Harder, Ursel Fantz and the NNBI Team

Max-Planck-Institut für Plasmaphysik, Garching, Germany

The two NBI systems at ITER are planned to inject up to 16.5 MW each of either 1 MeV D or 870 keV H atoms. These beams will be formed by the neutralisation of ion beams of either D^- or H^-, extracted from a D_2 or H_2 plasma. The targets for the accelerated ion currents are 40 A for D^- and 46 A for H^- from a source 1 m × 2 m in width and height, with extraction and acceleration grids containing 1280 apertures. Each aperture generates an individual 'beamlet', each of which expands with a characteristic divergence to merge downstream into a single, large, ion beam. Total beam divergence is also limited to a maximum of 7 mrad, for an efficient transmission into the tokamak, and to avoid damage to beamline components.

The beam divergence should be measured accurately by beam diagnostics. Power densities within the beam are high: the target parameters (above) give 20 $MW{\cdot}m^{-2}$, which precludes small probe type diagnostics. This limits diagnostics to calorimetric, electrical, or beam emission spectroscopy, all of which have low, if any, spatial resolution. Experimental data can be difficult to interpret due to mixing of information from many beamlets, which may have different properties or propagation directions. Separation of beamlet and whole-beam information is necessary to understand the divergence values obtained, so that efforts can be focussed for its reduction.

To assist with the resulting complex data interpretation, a whole-beam simulation has been developed for large ion beams, including the interaction with beam diagnostics and generation of synthetic emission spectra. Thanks to the traceability of information, the origins of phenomena in the diagnostic results can be found. Results are given for experimental and synthetic diagnostics of the BATMAN Upgrade test facility at IPP Garching. It is shown how different contributions to the

beam divergence can be identified in a way that is not possible with experimental data alone. Conclusions are also given for the implications on diagnostics of full-scale ITER-like ion beams.

Keywords: *NBI, Diagnostics, Simulation*

***Corresponding author:** *andrew.hurlbatt@ipp.mpg.de*

The SIMMER-III Code Validation: Post-Test Analysis of Test D1.1 on the LIFUS5/Mod3 Facility for In-Box LOCA of WCLL-BB

P2-049

Samad Khani Moghanaki [a]*, Marica Eboli [b], Nicola Forgione [a], Alessandro Del Nevo [b]

[a] *DICI – University of Pisa, Pisa, Italy*
[b] *ENEA FSN-ING-PAN, Camugnano, Italy*

The Breeding Blanket (BB) is a major system of a fusion reactor and the Water Coolant Lithium Lead (WCLL) type is one of the potential candidates for the DEMO (DEMOnstration) fusion reactor. The in-box LOCA (Loss of Coolant Accident) for the WCLL is considered as a highly affecting safety concern, therefore, transient behavior of it shall be carefully investigated and studied during the design phase, to evaluate the consequences and to adopt the necessary mitigating countermeasures. This requires also a numerical predictive tool, which is capable to simulate the safety phenomena and parameters occurring during a LOCA transient. Following this objective, the SIMMER-III code was firstly improved by implementing the chemical reaction between PbLi and water at the University of Pisa. Then, SIMMER-III Verification and Validation (V&V) procedures were established and conducted to obtain a qualified code for deterministic safety analysis. The verification activity was successfully completed and documented within the past numerical analytical and experimental activities, while the validation phase requires further efforts according with the R&D plan set up in the framework of the EUROfusion Project. In this way, an experimental campaign and a test-matrix has been designed according to the LIFUS5/Mod3 facility to perform a series of experiment and post-test analyses. The experimental data of the Test D1.1 is used for the validation of the SIMMER-III code according with a standard procedure. A qualitative analysis of obtained results is performed according to the available time trends. It aims to interpret the resulting sequence of main events and the identification of phenomenological windows and aspects, relevant to pressure transient and hydrogen production due to the chemical reaction between PbLi and water.

Keywords: *WCLL-BB, DEMO reactor, in-box LOCA, SIMMER-III*

***Corresponding author:** *s.khanimoghanaki@studenti.unipi.it*

Validation of Source Term Descriptions in MCNP and MCUNED Code Models for SINBAD Fusion Benchmark Compilations

P2-050

Ivan Kodeli *, Aljaž Čufar

Reactor Physics Department, Jožef Stefan Institute, Ljubljana, Slovenia

In order to preserve and make available the information on the performed radiation shielding benchmarks the OECD Nuclear Energy Agency Data Bank (OECD/NEA-DB) and the Radiation Safety Information Computational Center (RSICC) started in the early 1990's the Shielding Integral Benchmark Archive and Database (SINBAD) project. Currently, SINBAD database comprises 100 benchmark compilations and evaluations, of which 31 are of relevance to fusion blanket neutronics and further 23 concern accelerator shielding. 25 of these benchmarks were performed using D-T neutron source, particularly the FNG, OKTAVIAN, FNS and IPPE benchmarks. The present status and the future plans for SINBAD will be presented.

In addition to the characterization of the radiation source, description of the shielding set-up, the instrumentation and the relevant detector measurements, most sets in SINBAD contain also the deterministic or Monte Carlo radiation transport computer model used for the interpretation of the experiment. Since the existing transport codes, such as the Monte Carlo codes MCNP or TRIPOLI, do now allow explicit modeling of the DT reaction the MCNP code inputs presently available in the SINBAD database make use of the DT neutron source subroutine. However, this approach requires the recompilation of the MCNP code therefore it can be used only by the users who have access to the MCNP source code. Furthermore the subroutine becomes obsolete each time a new version of MCNP becomes available and has to be updated. To avoid these inconveniences alternative possibilities were tested and validated as well as the required CPU times compared. The first alternative is the MCUNED code, a new extension of MCNPX, and the second, the neutron source that is provided explicitly using the SDEF card, similarly as it was done in the past to perform deterministic transport calculations using the DORT/TORT S_N codes. Consistent results were obtained by the three methods. The new inputs will be available in the SINBAD database and are expected to facilitate the use of these benchmarks for many MCNP users who do not have access to its source code. This revision will ultimately concerns altogether 18 SINBAD benchmarks that presently include the MCNP input with the DT neutron source subroutine.

Keywords: *benchmark experiments, DT neutron source, MCNP*

***Corresponding author:** *ivan.kodeli@ijs.si*

Analysis of Thermal Response of New Diagnostic Probe in TCV

P2-051

Boštjan Končar [a*], Martin Draksler [a], Jernej Kovačič [a], Bernd S. Schneider [b], Codrina Ionita [b], Tomaž Gyergyek [c,a], Cedric Kar-Wai Tsui [d,e], Roman Schrittwieser [b]

[a] *Jožef Stefan Institute, Ljubljana, Slovenia*
[b] *Institute for Ion Physics and Applied Physics, University of Innsbruck, Innsbruck, Austria*
[c] *Faculty of Electrical Engineering, University of Ljubljana, Slovenia*
[d] *University of California-San Diego, La Jolla, California, USA*
[e] *Swiss Plasma Center (SPC), Ecole Poytechinque Federale de Lausanne (EPFL), Lausanne, Switzerland*

Thermal response of the new multi-purpose diagnostic probe inside the TCV (Tokamak à Configuration Variable) tokamak is analyzed numerically. The new diagnostics consists of two Cold Langmuir Probes (CLP), one Electron-Emissive Probe (EEP), two Retarding Field Analysers (RFA) and two Magnetic Pickup Coils, all mounted on a single probe head, called the New Probe Head (NPH). The probe head design, followed by the prototype, has to go through a number of tests (numerical and experimental), before it can be used in the tokamak environment, where it has to withstand repeated exposure to the high heat flux of the Scrape-Off-Layer (SOL) plasma. The NPH is constructed as a universal probe, that can be used in all MST tokamaks by applying corresponding adapters. In this study the thermal stress on the NPH inserted into the TCV environment is studied.

The probe exposure time to the SOL plasma during single insertion will be in the range of 100 milliseconds. Up to three probe insertions with the delay of 0.5 s are expected during one plasma discharge in TCV. The incident heat flux on the top of the NPH is modelled according to the radial distribution of the parallel heat flux in the SOL. Based on this, thermal response of the probe head to the transient heat loads will be simulated. The NPH will be cooled to the outside by thermal radiation and by the heat conduction through the probe body. Thermal analysis will give us a rather accurate estimation of NPH behavior in the SOL of the TCV. The probe structural integrity during such thermal response will be evaluated. The design of the graphite shroud protecting the diagnostics will be evaluated in particular, focusing on the gap effect between the upper cap and the cylindrical part of the shroud that may lead to excessive overheating of the shroud cap.

Keywords: *new probe head, thermal response, numerical simulation, TCV, scrape-off-layer*

***Corresponding author:** Bostjan.Koncar@ijs.si*

Understanding and Investigating the Relationships between Geometrical Errors and Lost Particles in MCNP

P2-052

Davide Laghi [b*], Marco Fabbri [a], Raul Pampin [a], Alfredo Portone [a]

[a] *Fusion for Energy, Barcelona, Spain*
[b] *Industrial Engineering Department, University of Bologna, Bologna, Italy*

Generating Constructive Solid Geometry (CSG) ready for MCNP input using automated programs (such as SuperMC or MC-CAD), and the use of universe structures, may give rise to geometrical inconsistencies leading to numerical phenomena known as ***lost particles***, which perturb the statistical reliability of the transport solution. Hence it is important to identify, understand and correct these inconsistencies as much as possible.

In the context of the reference nuclear analysis models developed for ITER, there is a need to better understand these phenomena, define a sufficiently explicit standard way to test the models, and provide clear and uniform parameters indicating their technical quality and reliability in terms of geometrical errors.

After identifying the key parameters relating these errors to the lost particles phenomenon, using a source able to generate an isotropic and uniform fluence inside its closed surface, analytical relationships were derived in order to predict the lost particle rate that would be caused by a specific source and a specific error size. The result was that the lost particle rate depends only on the ratio between error size and source surface area. Conversely, these relationships can also be used to estimate error size knowing the lost particle rate and the specific source used. Additionally, it is possible to associate an uncertainty to this estimation knowing the absolute number of particles that were lost. All these relationships were verified with multiple MCNP runs and good agreement was found between simulations results and analytical predictions.

This work allows defining more explicit and uniform procedures to debug for lost particles, and provides a way to grade model quality based on objective parameters, like the size of the errors, instead of using others that heavily depend on the setup on which the simulation was run.

Keywords: *lost particles, geometrical error, MCNP, debugging*

***Corresponding author:** davide.laghi01@gmail.com*

Tritium Decontamination Scenario from Plasma Facing Materials Under Vacuum Condition in DEMO

P2-053

Naoko Ashikawa [a,b]*, Y. Torikai [c], K. Katayama [d], M. Nakayama [e], A. Taguchi [e], M. Matsuyama [e]

[a] *National Institute for Fusion Science, Toki, Japan*
[b] *The Graduate University of Advanced Studies, Toki, Japan*
[c] *Ibaraki University, Mito, Japan*
[d] *Kyushu University, Kasuga, Japan*
[e] *Toyama University, Toyama, Japan*

In DEMO, maintenances after plasma operation as power productions are required to open plasma vacuum vessel. Hence, it is important that tritium (T) decontamination under vacuum conditions before open plasma vacuum vessel. At present, allowable value of remained T in the vacuum vessel did not decide yet, but to show a candidate T decontamination technique is required.

Three kinds of candidate techniques of T decontamination are considered in DEMO; 1) temperature control by decay heat and baking/cooling, 2) active wall conditionings, such as glow discharge, ion cyclotron wall conditioning and electron cyclotron wall conditioning, 3) a selection of working gas and vacuum pressure. Mainly retained tritium on the surface of materials are important for T decontamination. Since T is easily replaced with hydrogen, it is well known that the process of replacing H with T and desorption. Therefore, it is thought that the ratio of water molecules present in the space has a great influence on T decontamination.

In this study, coated tungsten on tungsten (W/W) specimens were exposed to T gas (7%) at temperature of 573 K in a glove box. Two kinds of different process were done for W/W specimens after T gas exposure under the following conditions. 1) W/W specimens exposed to atmosphere and remained surface T was measured by T imaging plate (TIP). 2) W/W specimens holding in the glove box and remained surface T was measured by TIP. For both W/W specimens, additional baking treatments were done from 373 K to 1073 K. Retained surface T on W/W specimens in atmosphere was reached about half of an initial surface T at 473K, but the W / W specimens holding the glove box reached half of the initial T at 773K. Surface T was exchanged by H or H2O in atmosphere. Hence, W/W specimens in atmosphere show quick T decontamination to compare with that in the glove box.

Keywords: *tritium decontamination, DEMO, vacuum condition, Surface tritium*

***Corresponding author:** ashikawa@nifs.ac.jp*

First Neutral Beam Injection Experiments in Versatile Experimental Spherical Torus

P2-054

Kihyun Lee, Seongcheol Kim, Yoosung Kim, Jongin Wang, Doyeon Kim, Wonik Jeong, Jung-hwa Kim, Jae-Young Jang, Y.S. Hwang *

Department of Nuclear Engineering, Seoul National University, Seoul, Korea

The neutral beam (NB) injection system of Versatile Experimental Spherical Torus (VEST) was installed, and commissioning was also successfully conducted in order to obtain ~0.6MW ion beam power. After commissioning process, NB injection experiments with VEST have been carried out. The ion source consists of arc plasma source and multi-aperture accelerator assembly which is developed by Korea Atomic Energy Research Institute (KAERI). The KAERI accelerator system was designed to deliver a 0.6MW with beam energy of 15kV. The arc plasma source can make high power of ~60kW Arc plasmas, which is enough for beam current of ~40A in this accelerator system. By using this NB system, a hydrogen NB power of 0.6MW was successfully injected to the VEST plasma. Difference of the plasma parameters by NB injection such as ion temperature, rotation, and plasma density was measured by using thomson scattering system, interferometry, passive emission spectroscopy and so on. From these experiment results, neutral beam injection efficiency was evaluated in comparison with those of the NUBEAM – ASTRA simulation.

Keywords: *Neutral Beam Injection, Versatile Experimental Spherical Torus, NB coupling, NUBEAM, ST*

***Corresponding author:** yhwang@snu.ac.kr*

Tripoli-4 Simulation of the FNG Copper Benchmark Experiment and the Tritium Production in the Lithium Diamond Detector for ITER-TBM

P2-055

Yi-Kang Lee *

DEN, Service d'étude des réacteurs et de mathématiques appliquées (SERMA), CEA, Université Paris-Saclay, Gif-sur-Yvette, France

The TRIPOLI-4 Monte Carlo radiation transport code has been widely used on the radiation shielding, criticality safety and fission reactor physics fields to support French nuclear energy research and industrial applications. With the growing interest in using the continuous-energy TRIPOLI-4 code for International Thermonuclear Experimental Reactor

(ITER) applications, a key issue that arises is whether or not the released TRIPOLI-4 code and its associated nuclear data libraries are verified and validated for the Deuterium-Tritium (D-T) fusion neutronics calculations. Two new fusion neutronics benchmark experiments were performed at the 14 MeV Frascati Neutron Generator (FNG). These experiments are utilized in this TRIPOLI-4 code benchmark study. The first experiment was performed using pure Copper block (60 x 70 x 60 cm^3), aimed at testing and validating the recent nuclear data for ITER design calculations. To study the measurement of tritium production in the ITER Test Blanket Module (TBM), the second FNG experiment considered a single crystal diamond detector covered with a thin layer of ^{6}LiF (LiDia detector) placed inside a polyethylene cylinder (25 cm diameter and 30 cm height) and irradiated with 14 MeV neutrons. TRIPOLI-4 calculations will be performed to simulate the reaction rates of various activation foils (^{93}Nb(n,2n), ^{27}Al(n,α), ^{58}Ni(n,p), ^{115}In(n,n'), ^{197}Au(n,2n), ^{186}W(n,γ), and ^{197}Au(n,γ)) versus penetration depth in the Cu block for the first benchmark and to evaluate the tritium production in the LiDia detector for the second benchmark. Investigation of the 3D modeling of experimental configurations is the first part of the paper. Study the neutron source models and the variance reduction options of TRIPOLI-4 for the two above experimental benchmarks is the second part. Using different nuclear data libraries to verify their impact on the calculation results is the third part. Sensitivity study on materials and geometry models is the last part by using different calculation options.

Keywords: *TRIPOLI-4 Monte Carlo code, 14 MeV neutrons, FNG benchmarks, Radiation shielding*

Corresponding author: *yi-kang.lee@cea.fr*

Transport Calculations with the JET Torus MCNP Models for Characterization of the Neutron Field

P2-056

Igor Lengar [a]*, Sean Conroy [b], Andrej Žohar [a], Paola Batistoni [c,d], Aljaž Čufar [a], Lee Packer [c], Rosaria Villari [d], Jonathan Naish [c], Sergey Popovichev [c], Bor Kos [a], Luka Snoj [a] and JET Contributors [e]

[a] *JSI, Reactor physics division, Jožef Stefan Institute, Ljubljana, Slovenia*
[b] *VR, Department of Physics and Astronomy, Uppsala University, Uppsala, Sweden*
[c] *CCFE, Culham Science Centre, Abingdon, UK*
[d] *ENEA Fusion Technical Unit, Frascati, Italy*
[e] *See the author list of X. Litaudon et al. 2017 Nucl. Fusion 57 102001*

A large number of experiments involving measurements of neutron flux and spectra are performed at the Joint European Torus (JET) tokamak. For their optimal exploitation i.e. to determine fusion neutron yield for example they are supported by radiation transport calculations. The large majority of transport calculations are performed with the Monte Carlo code MCNP, which is the reference code for several of the larger tokamak analyses including ITER and DEMO.

Several MCNP computational models of the JET torus exist, some covering a part of the torus and some full reactor. Three of the models, which have been built individually from different roots and exhibit different levels of detail, cover the full 360° of the torus and have been, in some variants and upgrades, used for a longer time either for in-vessel or torus-hall calculations. Important JET projects, including the recent 14 MeV neutron source calibration, have been supported by MCNP calculations with these models.

The type of calculations and characteristics of the different MCNP models of JET are outlined in this paper. Their suitability for calculations in different energy ranges, from 14 MeV down to thermal energies, is presented. The accuracy of calculations is partly checked by comparison of results between different models and in some locations from experimental data from activation foils. The calculations of the neutron field and the obtainable accuracy is presented for the JET tours geometry, including the study of variance reduction techniques.

Keywords: *JET, MCNP model, neutron transport, neutron flux, variance reduction*

Corresponding author: *igor.lengar@ijs.si*

First Simulation of Edge Impurity Transport and Divertor Fluxes on HL-2M with EMC3-EIRENE

P2-057

Z. Liang [a], S. Y. Dai [a]*, G. Y. Zheng [b], J. X. Li [b], Y. Feng [c], **Donghong Liu** [a], D. Z. Wang [a]*

[a] *Key Laboratory of Materials Modification by Laser, Ion and Electron Beams (Ministry of Education), School of Physics and Optoelectronic Technology, Dalian University of Technology, Dalian 116024, China*
[b] *Southwestern Institute of Physics, Chengdu, China*
[c] *Max-Planck-Institut für Plasmaphysik, Greifswald, Germany*

The intense heat load deposition on divertor targets and resulting erosion of wall material can degrade the divertor performance and reduce its lifetime, which has a strong impact on the steady state operation of fusion devices [1]. HL-2M is a new tokamak under construction at the Southwestern Institute of Physics, which is designed with a heating power of 4~15 MW and an upstream density of 2.0-3.5×10^{19} m^{-3} for routine operation [2]. In order to achieve a steady state and better performance, it is necessary to study edge impurity transport and heat loads on divertor targets for HL-2M. In this work, the properties of edge impurity transport and heat loads on divertor targets on HL-2M are investigated by the three-dimensional edge transport code EMC3-EIRENE [3,4]. The detailed analysis of the upstream density on edge carbon impurity transport and heat loads has been performed by EMC3-EIRENE modelling. It is found the C^{1+}-C^{3+} are mainly in the divertor region and C^{4+}-C^{6+} mostly distribute at the upstream for a low upstream plasma density. Furthermore, the higher upstream plasma density causes enhanced friction, which leads to a decrease of carbon impurity at upstream and an increase at divertor region. In addition, the increase of upstream plasma density leads to a suppression of heat loads on divertor targets.

[1] Federici G et al 2001 Nucl. Fusion 41 1967-2137

[2] G.Y. Zheng et al 2016 Nucl. Fusion 56 126013

[3] Feng Y et al 2004 Contrib. Plasma Phys. 44 57 [4] Reiter D et al 2005 Fusion Sci. Technol. 47 172

Keywords: *HL-2M, heat flux, divertor, EMC3-EIRENE*

Corresponding author: *daishuyu@dlut.edu.cn and wangdez@dlut.edu.cn*

SURO-FUZZ Modeling of Hydrogen Reflection on Tungsten Nano-Structure Surface

P2-058

Donghong Liu [a], S.Y. Dai [a]*, M. Wang [a], Q. Shi [b], D.Z. Wang [a]*

[a] *Key Laboratory of Materials Modification by Laser, Ion and Electron Beams (Ministry of E ducation), School of Physics and Optoelectronic Technology, Dalian University of Technology, Dalian, China*

[b] *Graduate School of Engineering, Nagoya University, Nagoya, Japan*

Since tungsten has a high melting point, low sputtering yield and tritium retention, it is currently a leading choice for the divertor material in ITER. However, experiments performed in a linear device with helium-containing plasma reveal that a fiber form nanostructure generally called "fuzz" [1], which has some advantages and disadvantages. Beneficial aspects compared to a smooth surface are lower sputtering yield, lower secondary electron emission yield, lower fuel retention, and better resistance to surface cracking in response to transient plasma loads [2]. On the other hand, tungsten fuzz decreases the optical reflectivity and the thermal conductivity [3]. The threedimensional (3D) kinetic Monte Carlo (KMC) code SURO-FUZZ has been developed to investigate the formation of fuzzy nanostructure in the micro- and second-scales [4]. In this work, the SUROFUZZ code has been upgraded to simulate the reflection characteristics of H ions and the sputtering yield of tungsten fuzz surface. It is found that the reflectivity of H ion and the sputtering yield of a fuzzy surface decreases with the porosity increasing, which is a reasonable agreement with experimental data [5].

[1] Takamura S, et al., Plasma and Fusion Research, 2006, 1:051-051.

[2] D.Nishijima, et al., Journal of Nuclear Materials 434 (2013) 230–234.

[3] S.Kajita, et al., Nuclear Fusion 49 (2009) 095005.

[4] Q. Shi, et al., Fusion Eng. Des. 136, 554-557 (2018).

[5] K.Doi, et al., Fusion Eng. Des. 136, 100-105 (2018).

Keywords: *fuzz, tungsten, Monte Carlo*

****Corresponding author:*** *daishuyu@dlut.edu.cn and wangdez@dlut.edu.cn*

EMC3-EIRENE Modelling of Impacts of the Injected Neon Amount on Heat Flux Deposition on EAST

P2-059

B. Liu [a], S.Y. Dai [a]*, G. Kawamura [b,c], **Donghong Liu** [a], R. Ding [d], Z.S. Yang [d], Y. Feng [e], D.Z. Wang [a]

[a] *Key Laboratory of Materials Modification by Laser, Ion and Electron Beams (Ministry of Education), School of Physics, Dalian University of Technology, Dalian, China*

[b] *National Institute for Fusion Science, National Institutes of Natural Sciences, Toki, Japan*

[c] *Department of Fusion Science, SOKENDAI (the Graduate University for Advanced Studies), Toki, Japan*

[d] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*

[e] *Max-Planck-Institut für Plasmaphysik, Greifswald, Germany*

One of the most critical concerns for fusion devices is the heat flux deposition on divertor targets, which have an impact on the divertor performance and lifetime. It has been found that by means of extrinsic impurity seeding, the heat flux deposition on divertor targets is suppressed effectively [1-3]. However, studies have found that spatial localization of the impurity source may result in toroidal asymmetries in the radiation power and the heat flux deposition on divertor targets [2].

In this work, the simulation of the heat flux deposition on divertor targets has been conducted with the three-dimensional edge transport code EMC3-EIRENE [4]. Impacts of gas injection on the heat flux deposition on divertor targets of EAST tokamak are studied with different gas puffing fluxes. The heat flux deposition on divertor targets can be reduced with increasing the gas puffing flux. Toroidal asymmetry of the heat flux deposition on divertor targets is negligible according to the modelling results. The decay length of the heat flux on divertor targets is used to illustrate the impacts of impurity gas injection on the distribution of the heat flux deposition. The simulation results show that an almost linear relationship between the decay length and gas puffing flux is obtained.

[1] G Kawamura, et al., Plasma Physics and Controlled Fusion, 60,084005 (2018).
[2] J. D. Lore, et al., Physics of Plasmas, 22, 056106 (2015).
[3] J. D. Lore, et al., J. Nucl. Mater. 463, 515-518 (2015).
[4] Shuyu Dai, et al., Plasma Physics and Controlled Fusion, 59, 85013 (2017).

Keywords: EAST, neon impurity seeding, heat flux deposition

**Corresponding author: daishuyu@dlut.edu.cn*

Photon Tally Convergence Acceleration in D1S Calculation

P2-060

Víctor López Ochoa *, Patrick Sauvan, Aljaž Kolšek

Departamento de Ingeniería Energética, Escuela Técnica Superior de Ingenieros Industriales, Universidad Nacional de Educación a Distancia (UNED), Madrid, Spain

The D1S methodology for shutdown dose rate calculations is widely used in fusion installations like ITER, JET or DEMO. This methodology allows to obtain response functions associated to decay photon flux with a single coupled neutron-photon Monte Carlo transport calculation. In order to produce enough photons for a correct evaluation of the photon field in the MC simulation, a decay photon is produced at each neutron collision. To conserve the correct photon statistics, the weight of the emitted photon is adjusted with the ration of the total photon production cross-section over the total collision cross-section. This cross-section ratio is dependent of the neutron energy and vary several orders of magnitude from 1Mev to 10Mev in most of materials used in fusion facilities. It has been observed that this large variation of photon weight produces high contribution low probability events in photon tallies leading to fluctuations of the tally value and slow convergence. In order to accelerate the photon tally convergence, a new variance reduction is proposed in this work. This variance reduction is based on neutron weight windows method with energy discrimination. The neutron weight is adjusted for each weight windows energy range so that the product of the neutron and photon weights is in the same order of magnitude for all energy bin. Two methods are proposed to produce such neutron weight windows weights. The first one is using the MCNP weight windows generator and the second based on the neutron flux level in each energy range. The compatibility of the second approach with the global variance reduction is discussed.

To evaluate the performance of the method SDR calculations have been performed the ITER cmodel using D1SUNED code.

Keywords: D1SUNED, low probability events, variance reduction, weight windows, SDR

**Corresponding author: vlopez@ind.uned.es*

Voltage Hold Off Test of the Insulating Supports for the Plasma Grid Mask of SPIDER

P2-061

Alberto Maistrello [a]*, Mauro Recchia [a], Nicolò Marconato [a], Tommaso Patton [a,b], Mauro Pavei [a], Emanuele Sartori [a], Gianluigi Serianni [a]

[a] *Consorzio RFX, Padova, Italy*
[b] *Department of Industrial Engineering, University of Padova, Padova, Italy*

SPIDER is the full scale prototype of the RadioFrequency driven negative Ion Source (IS) of the ITER Heating Neutral Beam Injector. It is equipped with a system of grids, each having 1280 apertures, composed of the Plasma Grid (PG) polarized up to -108 kV with respect to ground, the Extraction Grid (EG) polarized up to -96 kV with respect to ground and the Grounded Grid (GG). The negative ions are mainly produced by the interaction of plasma with the PG surface which is heated up to 150°C and covered with a caesium layer.

Recently the opportunity to explore the operation with a range of vessel pressures wider than the original requirements has been identified and, in view of an upgrade of the pumping system, a temporary solution consisting in a PG mask with a limited number of apertures was conceived. This solution will allow the beam operation of SPIDER, although at reduced ion beam current, in a configuration that permits the investigation of the beam optical parameters.

The PG mask will be held in position by means of about 100 PEEK insulating supports, called pushers, hooked to a stainless steel frame at the GG potential; they pass through the GG and the EG apertures and push the mask on the PG. With this temporary solution the EG and PG can be biased up to -30 kV and -34 kV respectively.

This paper deals with the design of a testbed capable of reproducing the working conditions of the pushers in SPIDER (electric field, vessel pressure, magnetic field and PG temperature), in order to verify the voltage hold off of some pusher prototypes, and reports on the results of the relevant experimental campaign.

Keywords: Voltage hold off, Pusher, PG mask, SPIDER, ITER HNBI

**Corresponding author: alberto.maistrello@igi.cnr.it*

LIFUS II Corrosion Loop Final Design and Screening of an Al Based Diffusion Coating in Stagnant PbLi Environment

P2-062

Daniele Martelli [a]*, S. Bassini [a], M. Utili [a], M. Tarantino [a], S. Lionetti [b], E. Zanin [b]

[a] *Experimental Engineering Division, Fusion and Technology for Nuclear Safety and Security Department, ENEA Brasimone Research Centre, Camugnano, Italy*

[b] *RINA Consulting-CSM S.p.A Roma Italy*

Corrosion phenomena of structural material in PbLi environment is influenced by the working fluid chemistry, temperature, velocity profile and impurities concentration dissolved in it. In the framework of the Eurofusion consortium, the Experimental Engineering division (FSN-ING) by the ENEA Brasimone Research Centre (RC) is performing scientific activities to support the development of Water-Cooled Lithium Lead Breeding Blanket (WCLL-BB) technologies. In this concept, being the PbLi used as breeder, neutron multiplier and tritium carrier, the PbLi required velocity inside the BB module is of the order of some mm/s thus minimizing corrosion and Magneto Hydro Dynamic issues. Nevertheless, higher velocities (of the order of 0.5 m/s) are reached inside components and piping of the lead lithium loop making corrosion phenomena of structural material a critical issue for the design. To investigate the corrosion rate of materials and to test coatings at different velocities (0.01, 0.1 and 1 m/s) and different exposure times at 550°C, a new experimental facility named LIFUS II was manufactured and installed in the Brasimone RC laboratories. The loop is equipped with a glove box to remove specimen in a controlled atmosphere and avoiding O_2 contamination of the loop and two cold traps arranged in a parallel configuration for the PbLi purification. The coolant chemistry composition is monitored through a dedicated sampling device. The piping and components installed in the hot leg of the experimental loop are internally coated by an Al based diffusion coating developed in collaboration with RINA Consulting-CSM to reduce the impurities concentration in the loop. A preliminary screening of the coating was performed in PbLi stagnant conditions at 550°C for 1000 h showing a good resistance. Here the final design of the facility is presented, and the experimental procedure adopted to test the Al based diffusion coating with the obtained results are discussed.

Keywords: *PbLi, Corrosion, WCLL BB, LIFUS-II, Chemistry control in PbLi*

Corresponding author: *daniele.martelli@enea.it*

Engineering Design of Wendelstein 7-X Alkali Metal Beam Diagnostic Observation System

P2-063

Domonkos Nagy [a]*, Sándor Zoletnik [a], Gábor Anda [a], Dániel Dunai [a], Matthias Otte [b], Sándor Hegedűs [a], Máté Lampert [a] and the W7-X team [b]

[a] *Wigner RCP, Budapest, Hungary*

[b] *Max Planck Institute for Plasma Physics, Greifswald, Germany*

On Wendelstein 7-X a Sodium beam emission spectroscopy (BES) diagnostic system has been installed in 2017 in order to measure plasma edge density and turbulence. The diagnostic setup consists of two parts: an alkali beam injector and an observation system through which we can observe the light emission by the alkali beam.

The observation system consists of two parts, which operate in parallel: a high sensitivity Avalanche Photodiode (APD) camera and an overview CMOS camera. The collected light is divided by a 45° angled mirror, which transmits 2.5% of the light to the CMOS camera while 97.5% goes to the APD camera. The light is focused by an optical lens system and transmitted to APD camera by optical fibres arranged along the observed beam.

To achieve sufficiently high photon flux the APD branch contains relatively big lenses (up to 244 mm diameter) therefore a robust lens holder structure had to be fixed onto a 184 mm diameter window flange at the top of the cryostat. The 45° angled mirror had to be exactly positioned to the centre of the optical axis while its holding structure should not cover the way of the light towards the APD branch. Beam light emission occurs at two spectral lines separated by 0.5 nm around 589 nm. A Carbon II line is located exactly on one of the Sodium lines, therefore cutting the Carbon emission was possible by measuring only one of the Sodium lines. A custom designed filter is used, which can be temperature tuned so as at room temperature it transmits both Sodium lines, while at about 60°C the Carbon line and one of the Sodium lines is strongly cut.

Engineering design solutions and technical developments of the W7-X alkali BES observation system are discussed in this paper.

Keywords: *Beam Emission Spectroscopy, Optics, Wigner, W7-X*

Corresponding author: *nagy.domonkos@wigner.mta.hu*

Computational Evaluation of N-16 and N-17 Measurements for a 14 MeV Neutron Irradiation of an ITER First Wall Component with Water Circuit

P2-064

Chantal Nobs [a]*, Jonathon Naish [a], Lee Packer [a], Mario Pillon [b], Maurizio Angelone [b], Stefano Loreti [b], Paolo Del Prete [b], Jesus Izquierdo [c], Raul Pampin [c], Luca Zanibellato [c]

[a] *Nuclear Technology Department, Culham Centre for Fusion Energy, Culham Science Centre, Abingdon, Oxon, UK*
[b] *ENEA – Department of Fusion and Technology for Nuclear Safety and Security, Frascati, Rome, Italy*
[c] *Fusion for Energy, The European Joint Undertaking for ITER and the Development of Fusion Energy, Barcelona, Spain*

During ITER operations the water coolant flowing through components such as the first wall, blanket modules, divertor cassettes and vacuum vessel will become activated by high energy neutrons. Two key neutron-induced reactions will occur with oxygen in the water producing the radioactive isotopes N-16 and N-17, which have relatively short half-lives of a few seconds. These nuclides are transported in coolant loops and, unmitigated, their decay emissions will induce additional nuclear heat in components, potentially including superconducting magnets, and lead to an increase in the occupational dose for workers and sensitive equipment outside the biological shield. A new experiment is being planned at the Frascati Neutron Generator to accurately measure N-16 and N-17 produced by irradiating an ITER first wall mock-up, with the aim to provide scientific justification to reduce safety factors. This paper will provide a detailed description of the neutronics calculations performed using GammaFlow to model the temporal evolution of activated water, along with MCNP6 and FISPACT-II to calculate the detector response and optimise the experimental design. To define the uncertainty of the calculated reaction rates associated with nuclear data the results calculated from six libraries will be compared with measured data, the results will also be prepared for an NEA SINBAD benchmark.

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

Keywords: *neutronics, activation, FNG, FISPACT-II, ITER*

Corresponding author: *chantal.nobs@ukaea.uk*

Preliminary Design of the HEBT of IFMIF DONES

P2-065

Oriol Nomen [a]*, Daniel Sánchez-Herranz [a], Concepción Oliver [b], Ivan Podadera [b], Rodrigo Varela [b], Francisco Ogando [c], Volker Hauer [d], Fernando Arranz [b], Sofía Coloma [e], Roland Heidinger [f]

[a] *IREC, Sant Adrià de Besòs, Barcelona, Catalonia, Spain*
[b] *CIEMAT, Madrid, Spain*
[c] *UNED, Madrid, Spain*
[d] *Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany*
[e] *Centre for Automation and Robotics UPM-CSIC, Madrid, Spain*
[f] *Fusion For Energy, BA/IFMIF, Garching, Germany*

IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO Oriented Neutron Source) is currently being developed in the frame of the EUROfusion Early Neutron Source work package (WPENS) and will be an installation for fusion material testing, with a neutron energy of 14 MeV by Li(d,xn) nuclear reactions thanks to a deuteron beam colliding on a liquid Li flow. The accelerator system is in charge of providing such high energy deuterons in order to produce the neutron flux expected. The High Energy Beam Transport line (HEBT) is the last subsystem of the accelerator system and its main functions are to guide the deuteron beam towards the Lithium target and to model it by the use of magnetic elements to the reference shape of the beam footprint at the Lithium Target.

The present work summarizes the current status of the HEBT design, including beam dynamics, vacuum, radioprotection, diagnostics and remote handling studies performed, along with the layout and integration of the line, including beam shaping elements, diagnostics and the vacuum system. This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *DONES, IFMIF, DEMO, HEBT, Preliminary design*

Corresponding author: *onomen@irec.cat*

Dynamic Modelling of the Helium-Cooled DEMO Fusion Power Plant with an Intermediate Loop and Energy Storage System (Indirect Cycle)

P2-066

Sixten Norrman [a], Marton Szogradi [a], Evaldas Bubelis [b]

[a] *VTT, PO Box 1000 02044 Espoo, Finland*

[b] *Karlsruhe Institute of Technology (KIT), Institute for Neutron Physics and Reactor Technology, Eggenstein-Leopoldshafen, Germany*

The Demonstration Fusion Power Plant (DEMO) project aims to construct and successfully operate an industrial size tokamak fusion power plant generating electricity to the grid. One key challenge, among many, relates to the pulsed operation of the machine and adopting conventional power conversion systems, which are commonly designed for continuous operation. The DEMO balanceof-plant systems have to be designed to manage a periodical drop in fusion heat production during the dwell period.

A concept utilizing a molten salt Intermediate Heat Storage System (IHTS) equipped by an Energy Storage System (ESS) between the helium-cooled Primary Heat Transfer System (PHTS) and the Power Conversion System (PCS) – Indirect Cycle – have been studied over the course of several years within the EUROfusion Balance of Plant (BOP) workpackage, with the aim to smooth the transition between pulse and dwell. KIT has, with the support from the industrial turbine manufacturer Siemens, developed and optimized the PCS scheme and performed steady-state balance analysis for power and dwell operations with the simulation tool Ebsilon. To complement the static analyses VTT has made a model of the same configuration with the system code Apros. Dynamic analyses including pulse and dwell transitions have been performed to verify that the developed balance-of-plant concept is feasible. Further, a plausible control strategy has been developed and optimized, which minimizes cyclic temperature and pressure loads on components, potentially induced by the pulsation of the low temperature DEMO heat sources (e.g. divertor and vacuum vessel), whose Heat Exchangers (HXs) are integrated in the feedwater train. With the increasing maturity of the design related to the key areas of the DEMO fusion power plant, during the last years also a higher accuracy of the models have been achieved.

Keywords: *DEMO fusion power plant, Apros, dynamic modelling, tokamak, energy storage*

***Corresponding author:** *sixten.norrman@vtt.fi*

Analysis of the Ingress of Coolant Event Tests Performed in the Upgraded ICE Facility Aimed at the ECART Code Validation

P2-067

Sandro Paci [a]*, B. Gonfiotti [a,b], D. Martelli [a], M.T. Porfiri [c]

[a] *University of Pisa, Department of Civil and Industrial Engineering, Pisa, Italy*

[b] *now at Karlsruhe Institut für Technologie, Institut für Neutronenphysik und Reaktortechnik, Eggenstein-Leopoldshafen, Germany*

[c] *ENEA, Nuclear Fusion Technologies, Frascati, Rome, Italy*

In the present work, the efforts on the validation of the ECART code against the eight tests performed in the upgraded Ingress of Coolant Event (ICE) facility are discussed. This activity has been carried out to extend the validation of the ECART code to incidental sequences related to future fusion plants. The upgraded ICE facility consists of a boiler, injecting water at high pressure inside a low-pressure tank simulating the Plasma Chamber (PC). This PC is in turn connected at the Vacuum Vessel (VV) through a plate simulated a divertor. The PC is also connected to the Pressure Suppression System (PSS) by means of several relief pipes closed by magnetic valves, opening when the PC total pressure exceeds 150 kPa. The PSS is initially filled with 0.5 m^3 of water. Finally, the VV is connected, through a narrow pipe, with the Drain Tank (DT). Eight tests were performed in this upgraded ICE facility, investigating different numbers of relief pipes, different initial PC and VV temperatures, and different mass flow rates, pressures and temperatures of the injected water.

The employed ECART code couples three modules: a thermal-hydraulic module, an aerosol-vapour transport phenomena module and a chemistry one. Although, only the thermal-hydraulic code section was activated in the present work due to the ICE tests characteristics. The obtained results by ECART showed an overall good agreement with the experimental data. This confirms that ECART is also a valuable tool for the safety analysis in future fusion plants, as already pointed out in previous works.

Keywords: *Ingress of Coolant Event, ECART code, Pressure Suppression System*

***Corresponding author:** *sandro.paci@ing.unipi.it*

Backwards Extrapolation Activation Diagnostics and Their Dynamic Range for Pulsed Neutron Source Measurements

P2-068

Lee W. Packer [a]*, S. Allan [a], S.C. Bradnam [a], S. Jednorog [b], M.R. Gilbert [a], E. Laszynska [b], N.J. Roberts [c], T. Stainer [a]

[a] *Culham Centre for Fusion Energy, Culham Science Centre, Abingdon, Oxon, UK*
[b] *Institute of Plasma Physics and Laser Microfusion, Warsaw, Poland*
[c] *National Physical Laboratory, Teddington, UK*

The measurement of fusion neutron yields provides a direct relationship with fusion power and is hence an important measure of experimental performance. In pulsed neutron emission scenarios, such as those experienced in dense plasma focus devices, inertial confinement fusion and short-pulse tokamak experiments, several considerations in selecting a suitable diagnostic must be made. These include the diagnostic sensitivity and linearity of neutron fluence measurement across a dynamic range, and how to ensure calibration across this range. Integrated fluence detection systems such as activation foils, CR39 and bubble detectors are widely deployed for such fields, but due to intermediate, often manual steps, have drawbacks over pulse counting or current-based detection systems which allow for real time data acquisition. Activation materials implanted within radiation detection media and advanced pulse processing offer the potential for a system designed to maximise neutron detection sensitivity, but still operate over a dynamic fluence range spanning several orders of magnitude. In addition, threshold reactions may be utilized to yield neutron spectrum information. Calibration of these systems may be achieved using relatively low emission rate steady state neutron sources, with factors that are straightforward to translate to defined pulsed fields, and have excellent linearity for high intensity measurements, alleviating the need for a pulsed neutron field standard.

Here in a generalized parameter study, the use of activation materials and their potential for integration within radiation detection technologies such as gas counters, scintillators or Cerenkov detectors are explored, where algorithms to counter instrument paralysis effects are applied. We use the FISPACTII inventory code with multiple nuclear data libraries, and survey the nuclear response of elemental materials, including In, Ag, P, S, Nb, Au and Y, to pulsed neutron fields. Through high-fidelity modelling of residual temporal emissions, and nuclear detector radiation transport models, we provide an insight into their sensitivity and discuss potential wider deployment as activation-based diagnostics for measuring neutron fields.

We apply paralyzeable and non-paralyzable deadtime models to these systems at various incident neutron field intensities, and particularly in high neutron fluence scenarios, apply a backwards extrapolation, or adjoint, algorithm to estimate the pulse total neutron fluence, associated uncertainty and extend measurement capability across a larger dynamic range. We discuss the range of plasma focus device activation diagnostics already used at the PF-1000 facility in Poland, and compare aspects of our modelling approach to experimental data obtained using the Ag-activation detector system. Beside the clear application to inertial confinement and plasma focus fusion experiments, their suitability for short-pulse tokamak experiments at MAST-U is discussed. Work supported by RCUK [grant number EP/P012450/1] and the Euratom research and training programme.

Keywords: *activation, FISPACT-II, neutron, pulsed fields*

***Corresponding author:** lee.packer@ukaea.uk*

Spiralock Locking Function Tests for the ITER Diagnostics In-Vessel Applications

P2-069

Miklós Palánkai [a]*, Mátyás Tóth [a], Gábor Veres [a], Tamás Szabolics [b], Jenő Kádi [a], Sergey Sadakov [c], Pavel Shigin [c], George Vayakis [c], Michael Walsh [c]

[a] *GEMS Engineering Ltd, Budapest, Hungary*
[b] *WIGNER RCP, RMI, Budapest, Hungary*
[c] *ITER Organisation, St Paul Lez Durance Cedex, France*

Different kind of fasteners are used in ITER for the fixation of the functional and structural in-vessel diagnostic components. The fasteners should retain initial tightening preload in a vacuum under neutron irradiation, thermal cycling and vibrations. These conditions can lead to incidental unscrewing of some bolts and nuts if they are not locked. It shall be equipped with some locking features which in most cases shall be remote handling compatible. One of the candidate on a role of a locking mechanism is Spiralock type thread. It has proven vibration resisting properties, positive record of experience in JET [1] and it is remote handling compatible. However Spiralock behaviors under thermal cycling induced by cyclic volumetric heat load are not known.

There is a concern that loosening rate of Spiralock thread in the case of volumetric heat load (i.e. due to nuclear heating) may be faster than the normal 60° thread. In the case of "normal" thread, bolts and nuts stick each other through friction and slight elastic deformation. Spiralock-shaped female thread pre-deform tips of the male pitches, strongly limiting a freedom of the male thread to thermally expand in elastic deformation zone. It might lead to further accelerated deformation of the pitches and rapid loosening of the tightening preload. This risk is addressed in the dedicated R&D study is being performed under service contract (IO/17/CT/4300001633) with ITER Organization.

Custom designed so called First Wall Samples (CuCrZr) with M16 fine thread and standard M8 bolts (316L) are the subject of the study. ITER-like thermal cycling alternating with vibration tests are applied to the components. Fasteners loosening is monitored via periodic torque audit. For comparison with Spiralock thread, tests with normal thread are performed in the same experimental conditions. This paper outlines intermediate results of this work.

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

[1] FED 112 (2016) 42, P. Bunting, V. Thompson, V. Riccardo. Fastener investigation in JET.

Keywords: *ITER, fasteners, Spiralock, tests, diagnostics*

***Corresponding author:** *miklos.palankai@gems-engineering.com*

Estimation of Radiation Conditions in the ITER Electron Cyclotron Upper Launcher with State-of-the-Art Simulation Techniques

P2-070

Raul Pampin [a]*, Natalia Casal [b], Marco Fabbri [a], Mario Gagliardi [a], Davide Laghi [a]

[a] *Fusion for Energy, Barcelona, Spain*
[b] *ITER Organization, St. Paul lez Durance, France*

The ITER Electron Cyclotron Upper Launcher (ECUL) port plug is undergoing final design evolutions towards manufacturing, which include engineering efforts to ascertain and improve the radiation conditions during operation and maintenance. Two important aspects of these efforts are discussed here.

First, design and analysis activities undertaken to ameliorate the residual radiation field (shut-down dose rate, SDDR), present during maintenance due to activation of components by plasma neutrons, are presented. Several shielding options were proposed and analysed. It was concluded that very significant improvement is achieved by those located within the port plug itself, and that further necessary reductions are only possible by considering changes in the surrounding environment.

Second, the nuclear heating profile in the critical blanket shield module (BSM) component was computed. During this part of the work, a pioneering application of the novel MCNP6 capability for use of conformal unstructured mesh (UM) for both geometry description and results tallying was made, in addition to the conventional approach of CSG geometry plus superimposed structured mesh. Different aspects of the functionality and results of both approaches were compared in different modelling conditions, thereby optimising the evaluation. It was concluded that, whilst the UM succeeds in accelerating geometry preparation and post-processing and in removing unphysical results inherent to the conventional approach, as expected, the CSG still provides a more computationally efficient representation leading to uniform precision. These may be limiting factors for large analysis models such as those needed for ITER systems.

Keywords: *ECUL, MCNP, radiation shielding, neutronics*

***Corresponding author:** *raul.pampin@f4e.europa.eu*

SPIDER Plasma Grid masking for Reducing Gas Conductance and Pressure in the Vacuum Vessel

P2-071

Mauro Pavei *, S. Dal Bello, G. Gambetta, A. Maistrello, D. Marcuzzi, E. Sartori, G. Serianni, V. Cervaro, F. Degli Agostini, L. Franchin, M. Tollin

Consorzio RFX, Padova, Italy

SPIDER experiment is operating at the PRIMA site in Padova (I) since June 2018, with the aim of testing and optimizing the negative ion source prototype for ITER Heating Neutral Beam Injectors. In the first operational phase it was discovered that, as the in-vessel hydrogen pressure exceeds the design requirements, discharges occur on the back of the radio frequency source. A specific operational campaign allowed defining a threshold below which the discharge probability is strongly reduced. In order to extend the operational range of the source pressure above the nominal value, while a significant upgrade of the vacuum pumping system is designed and realized, it was decided to proceed with the SPIDER operations by applying a temporary solution. A mask was installed on the beam source plasma grid, closing most of its apertures, in order to reduce the gas conductance between the inside of the radio frequency source and the surrounding volume. At first only 80 over 1280 apertures are left open, with a specific layout properly arranged so as to guarantee the possibility to diagnose the beam characteristics and to evaluate its uniformity. In the paper the plasma grid masking system will be described in detail, together with the main design choices, the thermal and structural analyses and the tests that were carried out to get a validation of the whole system design. Finally, an overview of the behavior of plasma grid mask during SPIDER operations will be given.

Keywords: *Neutral beam, Beam Source, SPIDER experiment, Plasma grid*

***Corresponding author:** *mauro.pavei@igi.cnr.it*

Error Handling Method for Digital Twin Based Plasma Radiation Detection

P2-072

Attila Piros *

C3D Engineering Consultant Ltd., Budapest, Hungary

In Industry 4.0 several new methods have introduced to approximate the real and digital world. Digital twins represent real object together with their data and functions in the digital world. The latest developments in digital twin researches extend the application of following areas like simulation-based system engineering, simulation-based optimization and simulation-based control. Application of digital twin in plasma radiation detection gives a new opportunity to model the procedure of detection and discover the possible errors.

This paper proposes a fuzzy based mathematical model of metal absorber–metal resistor bolometer as a digital twin. This model follows the detection process where the radiation is absorbed by a metallic layer and the change of the layer's temperature is measured by metal resistors. Based on the measured change of the resistance, the radiated power absorbed by the metallic layer is backcalculated. With this back-calculated power value the total plasma radiation (or spatial distribution thereof) can be deducted, once the geometrical properties of the observation (direction, solid angle, etc.) are known. To ascribe a given degree of trustability to the derived plasma radiative power, the use of a fussy logic based approach is herewith proposed and implemented.

This fuzzy inference system handles such absolutely independent factors like characteristics of the metallic layer (thickness, absorption coefficient, etc.) and the geometric setup of camera (tolerances in the line-of-sight, spatial position, etc.). Our fuzzy based digital twin is tested on results of numerical plasma simulations the uncertainties (as a possible sources of detection error) of which are also handled via fuzzy functions. The resulted mathematical model (camera + plasma) is a suitable tool to be able to estimate the overall error and uncertainty of radiation detection.

Keywords: *bolometer, error estimation, digital twin, fuzzy logic*

Corresponding author: *piros@c3d.hu*

FPGA Implementation of Diamond Detector Data Acquisition System for the ITER Radial Neutron Camera Using FlexRIO PXIexpress Technology: Architecture and First Results

P2-073

Marco Riva *, Basilio Esposito, Daniele Marocco, Fulvio Pompili, Francesco Belli

ENEA, Department of Fusion and Nuclear Safety Technology, Frascati, Rome, Italy

The ITER Radial Neutron Camera (RNC) main role is to measure the uncollided 14 MeV and 2.5 MeV neutrons from deuterium-tritium (DT) and deuterium-deuterium (DD) fusion reactions through an array of detectors located in collimated lines of sight (LOS) viewing the plasma through the ITER Equatorial Port Plug #1. The line-integrated neutron fluxes will be used to evaluate the radial profile of the neutrons emitted per unit time and volume (neutron emissivity) and therefore the neutron yield and the alpha particles birth profile.

One of the detectors mounted on the LOS are Single Crystal Diamonds (SCD) sensors. They will produce pulses with a spectra that will be produced in the system by a digital detector emulator following a spectra resembling the expected ITER neutron spectra in that position.

The paper describes the system architecture and the first results of the spectra resulting from the data acquisition and processing system in comparison with the original spectra and with the results from similar data acquisition systems.

Keywords: *RNC, FPGA, FLEXRIO, ITER, Diamonds*

Corresponding author: *marco.riva@enea.it*

Design and Configurations for the Shielding of the Beam Dump of IFMIF DONES

P2-074

Daniel Sánchez-Herranz [a]*, Oriol Nomen [a], Pedro Ortego [b], Beatriz Brañas [c], Francisco Ogando [d], Patrick Sauvan [d], Fernando Arranz [c], Sofía Coloma [e]

[a] *IREC, Barcelona, Catalonia, Spain*

[b] *SEA Ingeniería, Madrid, Spain*

[c] *CIEMAT, Madrid, Spain*

[d] *UNED, Madrid, Spain*

[e] *Centre for Automation and Robotics UPM-CSIC, Madrid, Spain*

IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO Oriented Neutron Source) will be an installation for fusion material testing, with a neutron energy peak of 14 MeV generated by

Li(d,xn) reactions based on a deuteron beam bombarding a liquid Li flow.

The accelerator system will provide such high energy deuterons to produce the neutron flux expected. The objective of the Beam Dump, part of the High Energy Beam Transport Line (HEBT), is to stop the pulsed beam at low duty cycle during DONES accelerator commissioning and start-up phases.

The present work explains the radiological design of the beam dump shielding and two different configuration approaches for the materialization of the design. The radiological design considers maintenance and operation, involving the re-dimension of the nearest wall, floor and ceiling concrete thicknesses. Activation of the materials in the HEBT line, originated by the leakage of neutrons through the beam dump entrance is evaluated and an ad-hoc solution is proposed for its minimization. Regarding the mechanical design, in the first configuration, the shielding is split into two halves horizontally, the upper-half requiring external lifting capabilities for its commission and maintenance. The second approach consists in a vertical splitting into two halves, which are self-moveable, avoiding the needs of external lifting capabilities for the remote handling of the shielding.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *DONES, IFMIF, DEMO, Beam Dump, Shielding*

Corresponding author: *dsanchez@irec.cat*

Evaluation of the Neutron and Gamma-Ray Doses in the LHD Torus Hall and the Basement on the Deuterium Plasma Operation and the Neutral Beam Conditioning

P2-075

Takuya Saze [a*], Takeo Nishitani [a], Hitoshi Miyake [a], Masashi Kisaki [a,b], Makoto Kobayashi [a,b], Kunihiro Ogawa [a,b], Mitsutaka Isobe [a,b], Masaki Osakabe [a,b] and LHD Experiment Group

[a] *National Institute for Fusion Science, National Institutes of Natural Sciences, Toki, Japan*

[b] *SOKENDAI (The Graduate University for Advanced Studies), Toki, Japan*

For the purpose of radiation safety, the neutron and gamma-ray doses have been measured with optically stimulated luminescence (OSL) + CR39 dosimeters in the Large Helical Device (LHD) torus hall and the basement floors on the deuterium plasma operation and also the neutral beam conditioning. A total of 10 dosimeters were placed on the torus hall, the basement 1st floor, and the basement 2nd floor. It is confirmed that the measured dose at each site increases with t the amount of neutron generation from LHD. Also, the neutron and gamma-ray doses in the LHD torus hall and the basement on the deuterium plasma operation and the neutral beam conditioning have been estimated by the neutronics simulation using the MCNP6 code. LHD and the LHD experimental building including the basement are modeled manually from the CAD drawings. Especially, penetrations in the floor concrete slab are carefully modeled. The model of LHD includes five neutral beam injectors (NBIs) and a couple of vacuum pumping ports. For NBIs, the injection ports, the vacuum tanks and the support structure are modeled excluding internal components such as beam dumps and the ion sources. The total neutron emission of the plasma is derived from the measurement with the neutron flux monitor. Also, the total neutron emission from the NBI beam dumps is evaluated by the maximum envelop of the database of the neutron emission rate and the beam current.

Based on the neutronics simulation results, it found that there were dose hotspots in the basement of NBI No.1, No.3, and No.5. At each measurement point of the dosimeters, the dose measured with the dosimeter showed a lower value than the simulation result. One of the reasons was that equipment such as cables in the penetration part of the floor concrete slab may have shielded the neutrons.

Keywords: *LHD, deuterium plasma operation, neutral beam conditioning, neutron, radiation control*

Corresponding author: *saze.takuya@nifs.ac.jp*

Measurements of the Angle-Dependent Reflectivity of Plasma-Facing Components and Assessment of the Impact on the Estimations of Coverage of the IVVS Measurements of the ITER VV

P2-076

Thomas Siegel [a*], Efstathios Kolokotronis [b], Philip Bates [b], **Paloma Matia-Hernando** [a], Marino Maiorino [a], Maria-Pilar Urizar [a], Parthena Symeonidou [a], Carlo Damiani [b], Gregory Dubus [b], Adrian Puiu [b], Roger Reichle [c], Marta de la Fuente [a], Andres Cifuentes [a]

[a] *ASE Optics Europe, Parc UPC Ed. K2M, Barcelona, Spain*

[b] *Fusion for Energy, Barcelona, Spain*

[c] *ITER Organization, St. Paul Lez Durance Cedex, France*

The In-Vessel Viewing System (IVVS) is a metrology instrument designed for deployment inside the ITER Vacuum Vessel (VV) to assess any damage, erosion or displacements of the Plasma-Facing Components (PFCs) through the lifetime of the ITER experiment. The latest developments of the IVVS have identified the Angle-Dependent Reflectivity (ADR) of the PFCs to be critical to the assessment of the coverage of the IVVS within the VV. VV components have now been manufactured and a specific instrument has been developed to measure their ADR on site, in an IVVS-like geometry with 100dB dynamic range. This work presents the first measurements of manufactured ITER PFCs' ADR together with an assessment of the impact of the ADR on the effective coverage of the metrology channel of the IVVS operating through Frequency-Modulated Laser Optical Radar (FMLOR).

Keywords: *IVVS, metrology, vacuum vessel inspection, frequency modulated laser optical radar, ITER*

Corresponding author: *Thomas.siegel@aseoptics.com*

Neutron Detector Response Sensitivity Study for Realistic Plasma Neutron Sources at the JET Tokamak

P2-077

Žiga Štancar [a]*, Sean Conroy [b], Luka Snoj [a], JET Contributors [c]

[a] *Jožef Stefan Institute, Ljubljana, Slovenia*
[b] *Department of Physics and Astronomy, Uppsala University, Sweden*
[c] *See the author list of "Overview of the JET preparation for Deuterium-Tritium Operation" by E. Joffrin et al. to be published in Nuclear Fusion Special issue: overview and summary reports from the 27th Fusion Energy Conference (Ahmedabad, India, 22-27 October 2018)*

A methodology for the calculation of realistic plasma neutron sources has been developed, intended to be used in Monte Carlo neutron transport computations at the JET tokamak. The method has been developed around JET plasma diagnostics measurements, which serve as the basis for state-of-the-art core plasma simulations performed with the TRANSP-NUBEAM code, and neutron energy spectra calculations performed with the DRESS code, yielding a comprehensive description of plasma neutron emission properties. The plasma source data is coupled with the MCNP code, which allows for detailed and efficient sampling of neutron position, emission vector and energy distributions.

The work presented is focused on the study of the sensitivity of neutron diagnostic systems at JET, specifically the time-resolved (ex-vessel fission chambers) and time-unresolved (activation system) neutron yield monitors, on the perturbation of basic plasma source parameters, e.g. electron temperature and density profiles, ion temperature profiles and the effective ionic charge, as well as plasma transport computational parameters, e.g. emissivity profile grid. A well characterized JET baseline deuterium (DD) plasma discharge is taken as reference, with its neutron emission characteristics compared to cases in which input plasma diagnostics measurement are perturbed within their respective uncertainties. For each of the scenarios a realistic plasma neutron source is generated and the response of neutron diagnostics is computed through neutron transport calculation, using a detailed JET computational model. It is shown that the fission chambers are relatively insensitive to plasma parameter changes. Larger changes are computed in the response of the activation system, which is sensitive to variations in the DD peak region of the neutron energy spectrum, especially to the uncertainty in the plasma bulk ion temperature. For reference the computed neutron systems' responses are compared to calculations performed with a simplified plasma neutron source.

Keywords: *Plasma neutron source, Sensitivity study, TRANSP, DRESS, MCNP, JET*

Corresponding author: *ziga.stancar@ijs.si*

Dynamic Modelling of the Helium-cooled DEMO Fusion Power Plant with an Auxiliary Boiler

P2-078

Marton Szogradi [a]*, Sixten Norrman [a], Evaldas Bubelis [b]

[a] *VTT, Espoo, Finland*
[b] *Karlsruhe Institute of Technology (KIT), Institute for Neutron Physics and Reactor Technology, Eggenstein-Leopoldshafen, Germany*

The Demonstration Fusion Power Plant (DEMO) project aims to construct and successfully operate an industrial size tokamak fusion power plant generating electricity to the grid. In order to succeed in this task, several challenges in the current fusion technologies have to be overcome. One key challenge relates to the pulsed operation of the machine and adopting conventional power conversion systems, which are commonly designed for continuous operation. The DEMO balance-of-plant systems have to be designed to manage a periodical drop in fusion heat production during the dwell period.

A configuration utilizing an Auxiliary Boiler (AUXB) between the helium-cooled Primary Heat Transfer System (PHTS) and the Power Conversion System (PCS) have been studied over the course of several years within the EUROfusion Balance of Plant (BoP) workpackage. In such arrangement the AUXB provides necessary power to maintain ~50% power generation conditions in the PCS during dwell time. KIT developed a realistic PCS scheme and performed steady-state balance analysis for pulse and dwell operation of DEMO with the simulation tool Ebsilon. To complement the static analyses VTT has made a model of the same configuration with the Apros system code. Dynamic analyses have been performed to assess plant performance during power cycles consisting of transitions from pulse to dwell and back to pulse operation. The goals of present study are to verify that the developed balance-of-plant concept is feasible, to establish and optimize a plausible control strategy

minimizing various stresses on the components, i.e. enhancing reliability of DEMO operation andthus providing input for further improvements.

Keywords: *DEMO fusion power plant, Apros, dynamic modelling, tokamak, auxiliary boiler*

**Corresponding author:* *marton.szogradi@vtt.fi*

SURO-FUZZ Modelling of Tungsten Fuzz Evolution in an Erosive Helium Plasma

P2-079

M. Wang [a], S.Y. Dai [a]*, Q. Shi [b], **Donghong Liu** [a], B. Cui [c], S.G. Liu [a], D.Z. Wang [a]

[a] *Key Laboratory of Materials Modification by Laser, Ion and Electron Beams (Ministry of Education), School of Physics and Optoelectronic Technology, Dalian University of Technology, Dalian, China*

[b] *Graduate School of Engineering, Nagoya University, Nagoya, Japan*

[c] *School of Science, Dalian University of Technology, Panjin, China*

Tungsten (W) has been widely considered as a major material for the divertor and the first wall in the next-step fusion devices due to its high thermal conductivity and melting point, low sputtering yield and tritium retention. However, experiments performed in linear helium-containing plasma devices reveal that a fiber form nanostructure, generally called “fuzz”, is generated on W surface under certain conditions: the surface temperature window is from 1000 to 2000K and the lower threshold of ion energy is 30 eV [1]. The nanorods structure dramatically changes the W morphology which leads to the decrease of optical reflectivity, thermal conductivity and sputtering yield. In addition, the nanostructured W can cause the enhancement of tritium retention and floating potential. The threedimensional (3D) kinetic Monte Carlo (KMC) code SURO-FUZZ has been developed to investigate the formation of fuzzy nanostructure in the micro- and second-scales [2]. The growth rate of tungsten fuzz shows a reasonable agreement with the experimental results in a non-erosive helium plasma [3]. In this work, the SURO-FUZZ code has been upgraded to simulate the fuzz evolution in an erosive helium plasma. The simulation results are benchmarked against the experimental data on PISCES-A device. It is shown that the high energetic helium particles can suppress the fuzz growth in the erosive helium plasma.

[1] Shin Kajita, et al., Nuclear Fusion, 49, 095005 (2009).
[2] Q. Shi, et al., Fusion Eng. Des. 136, 554-557 (2018).
[3] R.P. Doerner, et al., Nuclear Fusion, 51, 043001 (2011).

Keywords: *roughness, impurity transport, Monte-Carlo*

**Corresponding author:* *daishuyu@dlut.edu.cn*

Magnet AC Loss and Stability Analysis of EAST in ELM Mode

P2-080

Weibin Xi *

Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China

The dramatic change of transient heat loads produced by high temperature plasma in H-mode when ELM burst results in rapid control of the coil current of the PF magnet by plasma control system to ensure the stability of the plasma operation.This process leads to the increase of the magnet AC loss and the reduction of stability margin, which will further leads to the quench of the magnet. The AC loss under discharge condition in recent years was acquired by analyzing some low temperature operating parameters under ELM condition during discharge.The stability margin was also analyzed at current operating temperature, which showed that the magnet can operate steadily under the current ELM operation mode.

**Corresponding author:* *xiwb@ipp.ac.cn*

Numerical Analyses of Impurity Behaviors For CFETR Scenarios of 1GW Fusion Power by the Integrated COREDIV Code

P2-081

Hai Xie [a,b]*, Roman Zagórski[c], Rui Ding[b], Irena Ivanova-Stanik[c], Jiale Chen[b], Vincent S. Chan[d], Junling Chen[b] and the CFETR team

[a] *Advanced Energy Research Center, Shenzhen University, Shenzhen, China*

[b] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, Anhui, China*

[c] *Institute of Plasma Physics and Laser Microfusion, Warsaw, Poland*

[d] *General Atomics, San Diego, USA*

Tungsten (W) will be used as plasma facing material for the next step fusion facility China Fusion Engineering Test Reactor (CFETR). As a high Z material, W impurity in plasma can cause large core radiation losses, which will strongly affect the core plasma performance. External impurity seeding is considered as an approach to reduce heat load at divertor plate. Therefore, it is essential for CFETR to study the effects of impurity seeding on W concentration and core plasma performance. To this end, the self-consistent core-SOL integrated COREDIV code has been used to simulate CFETR scenarios of 1 GW fusion power with different types of impurity seeding (Ne, Ar, Kr). W source is determined by the sputtering processes from divertor. In the case of Ne seeding, operation in semidetachment condition is predicted in which plasma temperature at divertor target and W erosion can be strongly mitigated. However, achievement of this condition leads to significant reduction in fusion power (from P_{fus}≈1015 MW to 700 MW) due to plasma dilution by Ne. Higher Kr seeding can effectively reduce plasma temperature at divertor, but can barely affect W erosion rate due to the high sputtering yield by Kr. What's worse, Kr can result in strong core radiation losses (Kr radiation in core is ~ 90 MW when Z_{eff} = 2.5). Due to the high radiation in the core, power to SOL will be smaller than L-H transition threshold which is incapable of sustaining H mode. Among all considered seeding impurities, Ar seems to be an optimal choice which can effectively reduce power to divertor with sustaining high plasma performance in the core.

In order to have a more comprehensive evaluation of the CFETR operational window for Ar seeding, several key parameters have been analyzed. Impurity pinch in the core seems to have small effects on the global parameters, due to the synergy effect of enhancing power radiation and reducing W erosion. Simulations with different transport coefficients (D_{SOL}=0.15 m^2/s and 0.5 m^2/s) in ScrapeOff Layer (SOL) have been performed. Higher SOL transport of impurity can enhance the screening effect in the SOL region which results in less radiation and dilution in the core. But higher SOL transport of fuel ions tends to the opposite direction, which increases impurity concentration in the core. This is due to the fact of lower plasma temperature and thus longer impurity ionization lengths in the SOL region. If the puffing location changes from the divertor region to midplane, more seeded impurity can get into the core plasma which is inappropriate for CFETR operation.

Keywords: *CFETR, impurity effects, COREDIV modelling, radiation and dilution*

****Corresponding author:*** *xiehai@ipp.ac.cn*

Gamma-Ray Irradiation Effects on Optical Coatings and Polarizers for Edge Thomson Scattering System in ITER

P2-082

Eiichi Yatsuka *, Kazuhiro Torimoto, Masao Ishikawa, Takaki Hatae

National Institutes for Quantum and Radiological Science and Technology, Naka Ibaraki, Japan

Irradiation effects on structural materials, electric insulators, optical fibers, optical glass substrate and electronics have been investigated for a long time. On the other hand, the scope of this study is gammaray irradiation effects on optically coated windows, dielectric mirrors and a wire-grid polarizer for the edge Thomson scattering system in ITER (ETS). In ETS, high power laser beams with the wavelengths of 1064 nm and 694 nm are injected and spectra of scattered light within the wavelength of 590-1070 nm are measured. Therefore, windows with anti-reflection (AR) coating specialized for laser wavelengths and with broadband AR coating are investigated. For assessment of gamma-ray irradiation effect on the laser injection optics, five types of mirrors having different dependencies on the angle of incidence and the polarization at the laser wavelengths are investigated. All substrates for optical components investigated are fused silica. Cobalt-60 sources capable to generate the dose rate of 10 kGy/h is used. The total gamma-ray dose irradiated onto the samples so far is up to 3 MGy, which is comparable with that throughout 20 years of ITER operation outside equatorial diagnostic ports. Regarding optically coated windows, degradation of a few percent of the transmission at visible wavelength after 1 MGy of gamma dose is observed, whereas no degradation is observed at near infrared wavelength even after 3 MGy gamma dose. Regarding the dielectric mirrors and the wire-grid polarizer, no degradation is observed up to the gamma dose of 3 MGy. Investigation of laser-induced damage-threshold after irradiation is required to validate usability of optically coated windows and mirrors. A wire-grid polarizer would improve S/N ratio because its transmission is maintained against the level of gamma dose in ITER and it halves background light which has random polarization.

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

Keywords: *Gamma irradiation, Optical coating, Laser-induced damage, Thomson scattering*

****Corresponding author:*** *yatsuka.eiichi@qst.go.jp*

Dynamic Modelling of a Solid Energy Storage Concept for Pulsed Operation DEMO Fusion Power Plant (Direct Cycle)

P2-083

Jaakko Ylätalo [a]*, Antti Teräsvirta [a], **Sixten Norrman** [b], Marton Szogradi [b], Evaldas Bubelis [c]

[a] *Fortum Power and Heat Oy, Espoo, Finland*
[b] *VTT, Espoo, Finland*
[c] *Karlsruhe Institute of Technology (KIT), Institute for Neutron Physics and Reactor Technology, Eggenstein-Leopoldshafen, Germany*

The Demonstration Fusion Power Plant (DEMO) project aims to construct and successfully operate an industrial size fusion power plant. In order to succeed in this task, several challenges in the current fusion technologies have to be overcome. One key challenge is related to the pulsed operation of the Tokamak-fusion reactor. It is estimated that the tokamak reactor can run only in pulsed mode introducing challenges to conventional power conversion systems, which are commonly designed for continuous operation. The DEMO power conversion system has to be designed to manage a periodical drop in fusion heat production during the dwell period. The DEMO-project work packages Plant Level System Engineering (PMI) and Balance of Plant (BOP) aim to design a thermal- and cost-effective Primary Heat Transfer System (PHTS) and Power Conversion System (PCS) for the DEMO fusion reactor. As part of these work packages, modelling studies were done by KIT, VTT and Fortum during 2018. Personnel in KIT performed a steady-state balance analysis of several PCS concepts directly thermally connected to the Helium Cooled Pebble Bed PHTS (Direct Cycle) both during normal pulse (power) operation and during 10 minutes dwell time operation with the Ebsilon simulation tool. Ebsilon results were used to dimension and construct a dynamic 1-D Apros model for a solid energy storage concept analyzed by VTT and Fortum. In the solid energy storage concept, thermal accumulators and a steam drum are used to store heat during operation and are unloaded during the dwell period. The solid energy storage system succeeded in keeping turbine power around 50% during the dwell period; however the Apros model also showed strong transients in process parameters, which could be problematic when considering plant safety and process component wear.

Keywords: *DEMO fusion power plant, Apros, dynamic modelling, tokamak, small energy storage*

Corresponding author: *jaakko.ylatalo@fortum.com*

The Electron Temperature Fluctuation on EAST Tokamak

P2-084

Hailin Zhao *, Tianfu Zhou, Yong Liu, Liqun Hu

Institute of Plasma Physics, Chinese Academy of Science, Hefei, China

In future burning plasma like ITER, electron heat transport can be driven by turbulence in the IT G/TEM mode wavenumber range as well as in the short scale ETG mode range. Turbulent electron te mperature fluctuation is measured on EAST tokamak using correlation ECE (CECE) system. The CECE system consists of eight closely spaced narrow-band heterodyne radiometer channels, can cover the low field side region of ρ from 0.5 to 1.0. The system can measure TT_{ee} fluctuation at

BB_{tt} of 2.2 and 2.4 T. The CECE system can measure turbulence in the ITG and TEM mode range.

During density ramp up in the Ohmic discharge, the TT_{ee} fluctuation begins to play an important role when nn_{ee} is above $2.8 \times 10^{19}\ mm^{-3}$, when nn_{ee} becomes higher than $4.5 \times 10^{19} mm^{-3}$, the TT_{ee} fluctuation is below the noise level. This seems to be connected with turbulence transition between TEM and ITG. In the L mode discharge with ECH and NBI, the TT_{ee} fluctuation power spectrum changes at different power stages. During ELM-free H mode period, quasi coherent mode can be found, the frequency of the mode is about 19 kHz. The mode exsits at a wide radial range with ρ from 0.7 to 0.95. The quasi-coherent mode can periodically exhaust particles and heat to the last closed surface, so that H mode without ELM can be sustained.

Keywords: *electron temperature fluctuation, electron heat transport, ECH, H mode*

Corresponding author: *zhaohailin@ipp.ac.cn*

CFD Evaluation and Optimization of the HEMJ Divertor Cooling Design

P2-085

Meng Zhao *, B. Ghidersa, R. Stieglitz

Institute for Neutron Physics and Reactor Technology, Karlsruhe Institute of Technology, Karlsruhe, Germany

The designs of He-cooled divertor with multiple-jet cooling (HEMJ) systems are going-on for future fusions reactor DEMO at the Karlsruhe Institute of Technologies (KIT). One of the most important goals of the cooling design is to remove the heat flux of more than 10MW/m2 from the Tungsten target plate. To evaluate and optimize the HEMJ cooling design, computational fluid dynamics (CFD) method was implemented into the present work. Various parameters such as the jet-hole diameter, jet-hole numbers of the cartridge as well as the jet-to-wall distance were simulated and optimized by CFD program CFX-18.6. The results show that jet-to-wall distance and jet pattern are sensitive to the local heat transfer and jet-hole diameter has less influence on divertor cooling performance.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *HEMJ, jet-to-wall distance, CFD*

Corresponding author: *meng.zhao@kit.edu*

Fabrication of ITER LCTS Assembly Tools and Load Test for Lift Adapter

P2-086

Jinho Bae *, Min-Su Ha, Kyoungo Nam, Sikun Chung, Wooho Chung

National Fusion Research Institute, Daejeon, Korea

Three kinds LCTS (Lower Cryostat Thermal Shield) assembly tools are purpose-built tool for installation of ITER LCTS cylinder. Those tools are specially designed by ITER Korea Domestic Agency (KODA) for lifting and aligning LCTS cylinders. In January 2019, the manufacturing of LCTS assembly tools started at the factory of KODA's supplier. The assembly tools consist of LCTS cylinder lift adapter, LCTS cylinder pre-assembly Tool and LCTS cylinder align unit. Among them, especially, LCTS cylinder lift adapter is identified as lifting accessory. For that reason, it should be verified by load test using 1.5 times payload. After the load test, dimension inspection should be carried out to check the permanent deformation.

In this paper, the fabrication progress is summarized with description on the function of LCTS assembly tools. And also, the test procedure is described with load test result.

Keywords: ITER, Lower Cryostat Thermal Shield, Assembly Tools, Lift Adapter

**Corresponding author: jhbae@nfri.re.kr*

Preliminary Study and Selection of CFETR In-Vessel Component Tritium Dust Decontamination Method in Hot Cell

P2-088

Zheng Gong [a,b,c]*, Xiang Gao [a,c], Jianjun Huang [a,b], Bin Yu [a,b] Xiaodong Lin[a], Jiangang Li [a,c], Xiongyi Huang [c], Kun Lu [c], Sumei Liu [c], Yinxian Jie [a,c], YunTao Song [c]

[a] *Advanced Energy Research Center, Shenzhen University, Shenzhen, China*
[b] *Key Laboratory of Optoelectronic Devices and Systems of Ministry of Education and Guangdong Province, College of Physics and Optoelectronic Engineering, Shenzhen University, Shenzhen, China*
[c] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*

Decontamination is a necessary and effective way of radioactive contamination reduction in the IVC (In-Vessel-Component) maintenance and decommissioning for CFETR, due to the plasma unstable events result in the reactor produces a large amount of tritium dust in the vacuum vessel. It also plays an extremely important role in reducing the radioactive pollutants diffusion, cumulative radiation dose and staff occupational exposure level, controlling radioactive effluent of nuclear facilities and enhancing remote handling equipment life and reliability and even series environmental protection problems. In this study, we carry out selection and analysis of decontaminate method for CFETR hot cell, comparing to exiting mature cleaning and decontamination methods for radioactive contaminated surface, with the comprehensive consideration of the characteristics of PFC pollution and the limitation of hot cell remote handling operation, analyze the adaptability and requirements of the fusion facility, selecting the most suitable cleaning method satisfied with necessary requirements for CFETR hot cell. Establishment of PFC decontamination method evaluation system provides technical support for the design of hot cell remote handling cleaning device.

Keywords: CFETR, Dust, Decontamination, In-vessel Component, Hot Cell

**Corresponding author: g828z@qq.com*

The DTT Device: Preliminary Remote Maintenance Strategy

P2-089

Stanislao Grazioso *, Giuseppe Di Gironimo, Domenico Marzullo

CREATE/University of Naples Federico II-DII, Napoli, Italy

Divertor Tokamak Test (DTT) is the next Italian facility for nuclear fusion research aiming at bringing alternative divertor solutions to a sufficient readiness level to be adopted by the European DEMOnstrating fusion power reactor (EU-DEMO). Since a non-negligible activation is expected on plasma-facing components after DTT shutdown, remote maintenance is mandatory.

To date, the tokamak building internal layout, the number and size of ports, the segmentation and size of in-vessel components have been the result of a compromise between operational and maintenance needs. In-vessel main components to be remotely installed/removed are first wall and divertor. A special effort has been devoted in their design, and it is still in progress for ensuring their remote maintenance compatibility.

DTT device foresees 18 sectors of 20°, each with five ports. In the current configuration, the first wall modules at the inboard side are expected to be removed from all the upper ports; the first wall modules at the outboard side are segmented such that they can be removed from four equatorial ports; the divertor cassettes are expected to be removed from four ports next to their location (the 25° upper and lower ports), even if an alternative solution via the equatorial ports will be discussed. Conceptual design of remote maintenance equipment which can implement this strategy will be presented as well.

This work aims at presenting an overview of the current strategy planned for DTT remote maintenance, as the result of the development work which was possible only considering all the major aspects of the operational capability expected for the machine.

Keywords: DTT, remote maintenance

**Corresponding author: stanislao.grazioso@unina.it*

Fabrication and Load Test of ITER PF5 & 6 Coil Assembly Tools

P2-090

Min-Su Ha *, Jinho Bae, Kyoungo Nam, Sikun Chung, Soo-Hyeon Park, Wooho Chung, Hyun-Sung Hwang

National Fusion Research Institute, Daejeon, Korea

PF5 and 6 coil assembly tool is used to transfer, support and align the PF 5 & 6 coil. The tools are comprised of PF5 lifting adapters, PF5 temporary support and align units, PF6 lifting adapters, and PF6 temporary support and align unit. PF 5 and 6 coil will be lifted from assembly hall to the tokamak pit using PF 5 and 6 lifting adapter. Then, PF5 and 6 will be temporarily placed on their temporary support and align unit. These temporary supports will be used until completion of sector assembly in tokamak pit. Then, PF5 and 6 coil will be aligned using the hydraulic jacks to their final position.

Since PF5 and 6 lifting adapters are classified as lifting accessories, a load test of 1.5 times the nominal load (PF 5 and 6 Coil weight) is required to verify their structural integrity. For PF 5 and 6 support and align unit, the partial load test and function test are required to verify their alignment system.

This paper provides the test procedure and the test results on PF5 and 6 coil assembly tools.

Keywords: *ITER, PF Coil Assembly Tool, Structural Analysis, FEM*

****Corresponding author:*** *msha12@nfri.re.kr*

Improved Reconstruction and Anomaly Detection in JET Using LIDAR-Vision Fusion

P2-091

Emil T. Jonasson [a]*, Jonathan Beouf [b], Stephen Kyberd [b], Robert Skilton [a]

[a] *RACE (UK Atomic Energy Authority), Culham Science Centre, Abingdon, Oxfordshire, UK*

[b] *Oxford Robotics Institute, University of Oxford, Oxford, UK*

Just like most industrial or scientific installations, future fusion reactors will require more or less frequent maintenance. The expected environmental conditions, as well as the necessity of carrying out many maintenance tasks in parallel in result in remote robotic maintenance becoming a necessity in order to minimize the maintenance shutdown durations. Advanced technologies will be required to carry out the automated inspection and maintenance tasks. LIDAR is a promising technology which is only just starting to be applied in Fusion contexts. Though it is presently not radiation tolerant enough to be utilized in future reactor designs such as ITER or DEMO without further development, the low radiation levels in JET have presented an opportunity to evaluate the technology for use in fusion environments. In a previous publication, we have presented initial results using data captured in JET in the form of a coloured 3D-pointcloud created by LIDAR-Vision sensor fusion.

In this paper, we will present further results obtained by processing and utilising this data. This will include details on the improvement of model quality using recorded JET RH Boom kinematics data, updated pointcloud-CAD data comparisons using numerical methods, as well as the segmentation of the vessel interior into tiles and the detection of discrepancies between the CAD model and the dataset. Finally, the results will be discussed and the relevance of this technology for future remote maintenance system inspection / navigation tasks will be discussed.

Keywords: *Remote Maintenance, Inspection, LIDAR, Vision, Sensor Fusion*

****Corresponding author:*** *emil.jonasson@ukaea.uk*

Remote Handling Strategy and Prototype Tooling of the ITER Vacuum Vessel Pressure Suppression System Bleed Line Valve Assembly and Rupture Disk Assembly

P2-092

Keelan Keogh [a]*, Steven Craig [a], Stewart Hockley [a], Paul Talbot [a], Chris Lamb [a], David Hamilton [b], Chang-Hwan Choi [b]

[a] *UKAEA, Culham Science Centre, Oxfordshire, UK*

[b] *ITER Organization, St. Paul Lez Durance Cedex, France*

As part of the development work performed under the ITER Robotic Test Facility (IRTF) the development and substantiation of the maintenance strategy for the Vacuum Vessel Pressure Suppression System (VVPSS) must be investigated. The VVPSS is a key safety aspect of the vacuum vessel of ITER. It ensures after a coolant leak, and subsequent pressure event, the vessel is protected from the expanding gasses by controlling and venting them appropriately. After an Ingress of Coolant Event (ICE) or during maintenance the bleed line valve and rupture disk assemblies will require removal and replacement. During removal the function of vessel confinement must be retained. The remote handling of these components while maintaining the first confinement boundary is challenging due to the access restrictions around the pipe flange and environmental conditions in the area. To understand and develop the remote handling strategy for ITER the creation of prototype tooling and mock up environment has been done. This allows development and confidence in the proposed maintenance strategy leading to the final solution for use at ITER. In this paper we will

discuss the results of these prototype tests, the strategies and challenges which need to be overcome for the ITER application, and design considerations that must be taken to future designs to achieve a feasible solution for ITER.

Keywords: *Remote Handling, WPSS, ITER maintenance, Confinement*

**Corresponding author:* *keelan.keogh@ukaea.uk*

Fabrication and Factory Acceptance Test of ITER UPSE Installation Tool

P2-093

Kyoungo Nam *, Jinho Bae, Min-Su Ha, Sikun Chung, Wooho Chung

National Fusion Research Institute, Daejeon, Korea

The UPSE (Upper Port Stub Extension) installation tool is purpose-built tool to transfer, support and align the UPSE in assembly building and Tokamak pit. The UPSE installation tool consists of the main frame, OB support beam, moving unit, align unit and inside support bracket. The UPSE pre-assembled to the installation tool in assembly building is lifted and moved to Tokamak pit. And the installation tool is supported on the TF coil and radial beam supports. The UPSE is moved and aligned to its final position through manipulation of moving unit and align unit according to assembly procedure.

Since UPSE installation tool is classified as lifting accessories, a load test of 1.5 times the nominal weight as a factory acceptance test is required to verify their structural integrity according to French regulation.

This paper provides the fabrication, structural analysis and the test results of installation tool.

Keywords: *ITER, UPSE Installation Tool, Factory Acceptance Test, Fabrication, Structural Analysis*

**Corresponding author:* *namko@nfri.re.kr*

Development of Bolting Tool for Remote Handling of ITER First Wall Central Bolt

P2-094

Yuto Noguchi *, Kentaro Nakata, Nobukazu Takeda

National Institutes for Quantum and Radiological Science and Technology, Ibaraki-ken, Japan

In ITER, the plasma facing First Wall (FW) panels are positioned by its keys and fixed to the Shield Block (SB) with an M64 Central Bolt. As the Central Bolt is primary load support component between FW and SB, tightening torque up to 10kNm is required for the bolting operation of the FW central bolt to achieve necessary pre-load to the bolt. In addition to significant torque required, due to expected misalignment between Bolt and thread, the wrench needs to have compliance mechanism. Also, narrow access hole excludes the use of ordinary material.

This article presents design of remote handling tool for FW central bolt bolting to solve those exclusive design challenges, and demonstration of the functional requirements via prototype testing. A compliance mechanism was newly designed and successfully demonstrated for the capacity to handle the misalignment of the thread part and to withstand 10 kNm torque. Maraging steel wrench with out-of-standard Torx bit was designed and fabricated. Strength analysis of the bit, irradiation test of the maraging steel material, and accuracy of the tightening torque and axial force are reported in the article.

Keywords: *ITER, Remote maintenance, Bolting, Compliance mechanism*

**Corresponding author:* *noguchi.yuto@qst.go.jp*

A Taxonomic Approach to Failure Mode Analysis for Use in Predictive Condition Monitoring

P2-095

Luigi Pangione *, R. Skilton, R.Powell

RACE, Culham Science Centre, Abingdon, UK

An extensive knowledge of a system's failures is crucial for identifying areas where the reliability of the system can benefit from improvements, as well as informing the design of new systems. Moreover, relationships between faults and failures can be used to enhance the maintenance of the system.

In this paper we present a taxonomy of failure modes of the JET Remote Handling System. This system is used during maintenance and enhancements of in-vessel systems, and consists of two transporters (booms) and a two-armed manipulator, along with a number of supporting systems. In this work we first present a failure taxonomy suitable for our specific system, and then we apply this taxonomy model to our failure database. The presented failures have been collected during commissioning and operations over a period of over 5 years. Cataloged failures are extracted from the logs produced by the control system and from the daily log books recorded by the system operator.

Keywords: *fault taxonomy, condition monitoring,*

**Corresponding author:* *Luigi.Pangione@ukaea.uk*

Study of Decontamination and Maintenance for ITER Blanket Remote Handling System

P2-096

Makiko Saito*, Nobukazu Takeda

National Institutes for Quantum and Radiological Science and Technology, Ibaraki-ken, Japan

The ITER blanket remote handling system (BRHS) will operate in the ITER vacuum vessel (VV) after plasma operation to exchange the blanket first walls (FWs) and shield blocks (SBs), which can weigh up to 4.5 ton and be larger than 1.5 m. The most important operation for the BRHS is the two-year maintenance campaign in which all 440 FWs must be replaced. The in-vessel components will have been activated by neutrons produced by the D-T reaction, with the total dose estimated to be 5 MGy. Moreover, activated dust will have accumulated in the VV and will contaminate the BRHS. The BRHS must first be transported to the red zone of the hot cell facility where it will be decontaminated remotely. The BRHS will then be transported to the hands-on area to be decontaminated manually and undergo maintenance. The ITER Organization has defined the dose rate criteria in the hands-on area as 100 μSv/h, thus, decontamination and maintenance activities must be performed in compliance with this criteria. In this paper, the parts that will require replacement in the two-year maintenance campaign are estimated based on the results of past irradiation experiments studies; areas of the BRHS that could not be decontaminated remotely are specified; and the dose rate to workers standing near the BRHS are calculated by using Monte Carlo N–Particle Transport Code analysis to establish the BRHS maintenance plan.

Keywords: *ITER Blanket Remote Handling System, Remote maintenance, Radioactive dust, Decontamination*

Corresponding author: *saito.makiko@qst.go.jp*

Visual Anomaly Detection in Tokamak Components Using Generative Adversarial Networks

P2-097

Robert Skilton [a]*, Yang Gao [b]

[a] *RACE, UKAEA, Culham Science Centre, UK*
[b] *University of Surrey, Guildford, UK*

Present inspection techniques in place at the Joint European Torus (JET), as well as some of those planned for ITER make use of telerobotic deployed inspection systems, which typically collect data for offline analysis. This can be a slow, laborious and subjective or error-prone process. There are significant benefits to be gained through some level of automation or user assistance through prioritization of samples for analysis.

Automated visual anomaly detection is a highly challenging problem due to high dimensionality of the input data, meaning that the normal statistical distribution cannot be directly modelled. We provide a robotic and algorithmic framework that utilizes Generative Adversarial Networks (GANs) to indirectly model this distribution, and hence provide a mechanism to quantify the anomalousness of given image data samples from a tokamak environment.

This paper presents an overview of the architecture and algorithms employed as well as quantitative and qualitative assessments of the performance against data from both a benchmark dataset, and real data gathered from JET components.

Keywords: *Maintenance, Inspection, Neural Networks, Computer Vision, Automation*

Corresponding author: *robert.skilton@ukaea.uk*

A New Array Eddy Current Probe for Inspection of Small-Diameter Tubes in Tokamak Fusion Nuclear Devices

P2-098

Yingsong Zhao [a], Pan Qi [b], Zheng Xie [a], Peigen Bai [a], Zhenmao Chen [a]*

[a] *State Key Laboratory for Strength and Vibration of Mechanical Structures, Shaanxi Engineering, Research Center for NDT and Structural Integrity Evaluation, Xi'an Jiaotong University, Xi'an, China*
[b] *China Nuclear Power Operation Technology Co. , Ltd. , Wuhan, China*

Nuclear power plant (NPP) is the final goal of the research and development in the controlled nuclear fusion technology. Not only in the future fusion NPPs, but also in the demonstration Tokamak fusion reactors and even the Test Tokamak devices, including ITER and EAST, small-diameter tubes are widely used, such as the coolant tubes of water or liquid helium in ITER and the steam generator tubes in the future Demo plant. To ensure safety of controlled fusion reactors, pre-service and in-service nondestructive testing for small-diameter tubes is very important. This paper proposes a new array eddy current probe with higher efficiency and detectability aiming for inspection of small-diameter tubes. The exciting part of the probe contains several spiral coils, which makes the induced eddy current be significantly disturbed by the major defects in tube wall, i.e. axial or circumferential cracks. The pickup part comprises four pancake coils with square arrangement. The final detection signal is the specific difference of outputs from these four coils. This special arrangement and output mode can obviously reduce the wobbling noise due to probe lift-off change and inclination. The validity and efficiency of the probe were verified through numerical simulation and experiments. A hybrid FEM-BEM code of A-ϕ method is adopted in eddy current testing (ECT) signals simulation, and the related

experiments were conducted on the array ECT equipment (ECTANE). Both the simulation and experiment results reveal that the proposed new array probe has outstanding performance for the inspection of both axial and circumferential crack in small-diameter tubes. In addition, owing to the application of array technique of new coil structures, the new ECT probe shown much better detectability than the conventional Bobbin type ECT probe but with almost the same high inspection speed.

Keywords: Fusion nuclear devices, small-diameter tubes, eddy current testing, array probe

***Corresponding author:** chenzm@mail.xjtu.edu.cn*

Progress of Design and Analysis of Divertor Remote Maintenance System for CFETR

P2-099

Wenlong Zhao [a,b], Yuntao Song [c,d]*, Xiaodong Lin [a], Jianjun Huang [a], Huibin Sun [a], Yong Cheng [c], Kun Lu [c], Xiang Gao [c], Jiangang Li [c], Junwei Li [c], **Huapeng Wu** [e]

[a] *Advanced Energy Research Center, Shenzhen University, Shenzhen, China*
[b] *Key Laboratory of Optoelectronic Devices and Systems of Ministry of Education and Guangdong Province, College of Physics and Optoelectronic Engineering, Shenzhen University, Shenzhen, China*
[c] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*
[d] *University of Science and Technology of China, Hefei, China*
[e] *School of Energy Systems, Lappeenranta University of Technology, Lappeenranta, Finland*

China Fusion Engineering Testing Reactor (CFETR) will be built to test and verify the feasibility of engineering and technology in practice for the future fusion reactor. The in-vessel components divertor's remote replacement is the key maintenance operations for the CFETR to meet the requirements of duty cycle time (≥30~50%). This paper concentrates on the CFETR divertor maintenance approach together with recent efforts towards the design and development of the associated remote handling equipment and procedures. First, an overview on the remote handling (RH) system is given and key design features are described. Then, Comparative study of various maintenance schemes for divertor is done in order to reduce the maintenance time, increasing the availability of the Tokamak. In addition, the high load capacity and reliability of the RH system analysis is presented in this paper.

Keywords: CFETR, divertor, maintenance strategy, remote handling, Reliability

**Corresponding author: zhwl@ipp.ac.cn*

Functioning of GEM Based Detector for Poloidal Tomography under Plasma Radiation Requirements

P2-100

Maryna Chernyshova [a]*, Karol Malinowski [a], Tomasz Czarski [a], Iraida N. Demchenko [b], Yevgen Melikhov [c], Kamila Zarębska [b], Ewa Kowalska-Strzęciwilk [a], Marcin Strawski [b], Andrzej Wojeński [d], Paweł Linczuk [a,d], Rafał Krawczyk [d,e]

[a] *Institute of Plasma Physics and Laser Microfusion, Warsaw, Poland*
[b] *Laboratory of Electrochemistry, Department of Chemistry, University of Warsaw, Warsaw, Poland*
[c] *School of Engineering, Cardiff University, Cardiff, UK*
[d] *Warsaw University of Technology, Institute of Electronic Systems, Warsaw, Poland*
[e] *CERN, Geneva, Switzerland*

After the problem of high-temperature plasma confinement, its contamination with impurities is another critically important problem, solution of which would allow continuation towards the success in controlled thermonuclear fusion. Basic information about impurities is obtained from studying their characteristic emission. The solution of most tasks within the problem of impurities depends, to a decisive degree, on the knowledge on the dynamics of the emission of impurities in time and space (over the cross section of the plasma cord). Recently the metallic walls have been installed on many machines and this issue has become more urgent with W, in particular, due to the high emissivity of high-Z materials. Therefore, there is a need to develop diagnostics, that suffers much less to a neutron damage than standard semiconductor diodes, to be able to reconstruct the impurity distribution. Information from spatial and energy distribution of soft X-ray (SXR) plasma radiation should allow also better estimation of the plasma state and plasma parameter optimization for future research fusion reactors. The aim of this work is to design a new diagnostics based on Gas Electron Multiplier technology for poloidal tomography focused on the metal impurities radiation monitoring in SXR region. This contribution highlights the latest achievements in the construction of the detector with the curved detecting surface. The results of progressive laboratory tests of the constructed detector will be presented demonstrating the potential of such diagnostics.

Keywords: Nuclear instruments for hot plasma diagnostics, X-ray detectors, Electron multipliers (gas), Micropattern gaseous detectors

***Corresponding author:** maryna.chernyshova@ipplm.pl*

Utilizing Silicon Photomultiplier Detectors for Low Light Level High Frequency Measurements in Fusion Diagnostics

P2-101

Dániel Dunai [a*], S. Zoletnik [a], M. Lampert [a], D.I. Refy [a] and W7-X Team [b]

[a] *Wigner Research Centre for Physics, Budapest, Hungary*

[b] *Max-Planck-Insitut für Plasmaphysik, Greifswald, Germany*

In several fusion diagnostics (BES, FILD, GPI, ultrafast spectroscopy, etc) similar criteria are present for the optimal detector choice. In these applications typically visible light is measured and the incoming photon flux varies between 10^7 -10^{11} photons/sec. The number of channels is usually 16-128, which is small compared to a standard CMOS. From constraints of the optics mostly relatively large surface detectors are needed (1-20 mm^2). The detectors are either coupled to fiber optics or they are placed in the image plane of a direct imaging optics. The required bandwidth is between 100kH and 1MHz. Traditionally Photomultiplier tubes (PMT) and photo diodes (PD) were used for fast systems, later avalanche photo diodes (APD) were also applied. The selected detector will determine the resulting signal quality and therefore for the physics capabilities of the diagnostic.

A new detector type has been developed in recent years; the silicon photomultiplier (SiPM). SiPM detectors are solid state photodetectors, which were developed for photon counting applications. Based on an actual BES application a study has been conducted to test and characterize the SiPM detectors. In some BES measurements the detectable photon flux is only be 10^7-10^8 photons / sec. It was shown that these detectors can also be applied to continuous measurements not just to photon counting applications. After careful tests the APD matrix detector of JET lithium BES diagnostic was replaced with SiPM matrix detectors. Also the scrape off layer channels of the Wendelstein 7-X alkali BES diagnostics were built partially with SiPM detectors. In this paper the performance of these detectors in fusion environment will be presented in comparison with APD detectors via some examples from the laboratory and from actual plasma measurements. The photon flux range where the SiPM should be considered will also be discussed with actual experimental examples.

Keywords: *fusion diagnostics, fast detectors, avalanche photodiode, silicon photomultiplier*

**Corresponding author:* *dunai.daniel@wigner.mta.hu*

Temperature Dependence of the Hall Coefficient of Sensing Layer Materials Considered for DEMO Hall Sensors

P2-102

Slavomir Entler [a*], Ivan Duran [a], Karel Kovarik [a], Ondrej Grover [a], Michal Kohout [b], Vincent Mortet [b], Andrew Taylor [b], Karel Vyborny [c], Zbynek Soban [c], Josef Sebek [b], Dominik Najman [d]

[a] *Institute of Plasma Physics of the CAS., Prague, Czech Republic*

[b] *Institute of Physics of the CAS, Prague, Czech Republic*

[c] *Institute of Physics of the CAS, Prague, Czech Republic*

[d] *Czech Technical University in Prague, Praha, Czech Republic*

The Hall sensors are proposed as a part of the DEMO magnetic diagnostics performing an absolute measurement of the steady-state magnetic field. The Hall sensors will contribute to the measurement of the plasma current, plasma-wall clearance, and local perturbations of the magnetic flux surfaces near the wall. Overall, 240 in-vessel Hall sensors should be installed between the blanket and vacuum vessel, and 544 ex-vessel Hall sensors on the outer skin of the vacuum vessel. On that account, the sensors operating temperature range is in the order of 400 - 500 °C (He-cooled blanket), 300 - 400 °C (water-cooled blanket), or 190 - 210 °C (ex-vessel sensors).

The Hall coefficient as one of the main parameters of Hall sensors is of decisive importance to provide sufficient measurement accuracy and noise immunity. However, the magnitude of the Hall coefficient depends on the temperature. The paper brings an evaluation of the temperature dependencies of the Hall coefficients of sensing layer materials considered for DEMO Hall sensors.

The consequences on sensor measurement accuracy are analysed.

Keywords: *magnetic diagnostics, Hall sensors, Hall coefficient, temperature, DEMO*

**Corresponding author:* *entler@ipp.cas.cz*

EU DEMO EC Equatorial Launcher Pre-Conceptual Performance Studies

P2-103

Saul Garavaglia [a*], Benedetta Baiocchi [a], Alessandro Bruschi [a], Aljaž Čufar [b], Francesco Fanale [a], Gustavo Granucci [a], Alessandro Moro [a], Emanuele Poli [c], Natale Rispoli [a], Theo Scherer [d], Mattia Siccinio [c,e], Peter Späh [d], Dirk Strauß [d], Ioannis Tigelis [f], Christos Tsironis [g], Thomas Franke [c,e], Minh Quang Tran [h]

[a] *Istituto di Fisica del Plasma, CNR, Milano, Italy*
[b] *Reactor Physics Department, Jožef Stefan Institute, Ljubljana, Slovenia*
[c] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[d] *KIT, IAM-AWP, Karlsruhe, Germany*
[e] *EUROfusion Consortium, Garching, Germany*
[f] *Faculty of Physics, NKUA, Athens, Greece*
[g] *School of Electrical and Computer Engineering, NTUA, Athens, Greece*
[h] *Swiss Plasma Center, EPFL, Lausanne, Switzerland*

The preliminary conceptual design for the Electron Cyclotron (EC) system of the future European DEMOnstration Fusion Power Plant is ongoing inside EUROfusion Consortium, and it is encompassed the evaluation of the EC launcher. This represents one of the key aspects in order to assess the performances and the integration capabilities of such a system in EU DEMO as well as in the alternative reactor configuration Flexi-DEMO. Three different options for the antenna, namely Remote Steering Antenna (RSA), Mid Steering Antenna (MSA) and Open Ended Waveguides (OEWG), are investigated, analyzing their performance for several injection angles and launch points. This activity takes into account the constraints given by physics and engineering requirements, as for example the maximum power per port and the necessary local current drive to stabilize neo-classical tearing modes (NTMs) with a proper deposition width. The beam tracing calculations have been performed on different scenarios, providing information on plasma accessibility and deposited power. The microwave design and initial ideas about the ex-vessel EC transmission lines routing will be shown. Based on former neutronics analyses which were performed to define the radiation effect on the launching structure, conclusions are drawn to quantify and to mitigate the effect of penetrations in the equatorial port plug for the EC heating system and DEMO machine.

Keywords: *DEMO, Electron Cyclotron Heating & Current Drive, Gyrotron, Launcher*

Corresponding author: *garavaglia@ifp.cnr.it*

Stabilization of Resistive Wall Modes for ITER Using Model Predictive Control

P2-104

Samo Gerkšič [a*], Boštjan Pregelj [a], Marco Ariola [b]

[a] *Jožef Stefan Institute, Ljubljana, Slovenia*
[b] *Consorzio CREATE / Università degli Studi di Napoli Parthenope, Naples, Italy*

Tokamak control systems have to deal with different kinds of plasma instabilities. The most common such instability is the vertical instability due to an axisymmetric ($n = 0$) mode. A resistive wall mode (RWM) is an instability due to plasma kink at higher plasma pressure, moderated by the presence of a resistive wall that surrounds the plasma. The subject of this work is the kink instability associated with the main non-axisymmetric ($n = 1$) RWM, dominant in advanced tokamaks.

Model Predictive Control (MPC) is an advanced process control method established in the process industry, facilitating a systematic approach to control of large-scale multivariable systems and efficient handling of constraints on process variables. These advantages are beneficial for tokamak control, but the relatively long sample computation time typically needed for the on-line optimization represents a difficult obstacle. Speeding up MPC has been a topic of a recent intensive research.

RWM control with MPC is challenging because fast dynamics are combined with high system dimensions (27 actuators, 6 sensors, 50 dynamic states). Therefore, a pre-computed MPC approach is not applicable, and the dual Fast Gradient Method (FGM) solver [1] exceeds the allowed computation time. However, an infinite-horizon MPC considering only actuator constraints using a solver such as the primal FGM appears viable. The MPC RWM controller is based on a prior LQG RWM scheme for ITER [2]. In simulations, the MPC controller stabilises the system with narrower voltage constraints and using less power than the LQG controller and with an acceptable computation time 0.1 ms.

[1] *Gerkšič S. et al., Model predictive control of ITER plasma current and shape using singular value decomposition, Fus. Eng. Des. 129 (2018) 158–163*

[2] *Ariola M., Pironti A. (2016) Control of the Resistive Wall Modes for the ITER Tokamak. In: Magnetic Control of Tokamak Plasmas. Advances in Industrial Control, Springer*

Keywords: *predictive control, plasma magnetic control, quadratic programming, fast gradient method*

Corresponding author: *samo.gerksic@ijs.si*

Design of High Current Cesium-Free Negative Ion Source by Sheet Plasma

P2-105

Keito Hanai [a]*, Toshikio Takimoto [a], Hiroki Kaminaga [a], Akira Tonegawa [a], Kohnosuke Sato [b], Kazutaka Kawamura [a]

[a] *Tokai University, Kanagawa, Japan*

[b] *Tokyo University of Science, Tokyo, Japan*

Neutral Beam Injection (NBI) is most powerful external heating system in International Thermonuclear Experimental Reactor (ITER). Negative ion source plays an essential role for NBI system of steady state magnetic nuclear fusion. These negative hydrogen ion sources require cesium seeding to achieve a high ion density. However, cesium degrades the time stability and uniformity of the beam. Therefore, the development of a negative ion source without cesium seeding is strongly desired, and we have development of negative ion source in a cesium-free discharge by the magnetized sheet plasma device, TPDsheet-U [1]. Negative hydrogen/deuterium ion beams are extracted through the small single hole (diameter; 4 mm) at a neutral gas pressure P_s of 0.3 Pa [2,3]. At an extraction voltage of 7 kV, the maximum negative current densities J_c is about 11.7 mA/cm^2 and 6.03 mA/cm^2 with hydrogen and deuterium discharge, respectively. We have developed the large extraction electrode to extract high current negative ion beam from the sheet plasma. The developed electrode has been increased in size (length; 124 mm, width; 24 mm, extraction hole diameter; 4 mm, 3×13 holes) to get negative ions from larger area of sheet plasma. The position of the extraction holes was settled in consideration of the uniformity of the sheet plasma and the width of it. In addition, we will report results of the extraction experiments of multi-electrodes.

This study had been supported by the LHD Joint Project of the National Institute for Fusion Science (NIFS16KOAR021).

[1] A. Tonegawa et al., Japanese Jornal of Applied Physics, 45, 8212, (2006).

[2] R. Endo, S. Ishihara, T. Takimoto, A. Tonegawa, K. Sato, K. Kawamura, AIP Conference Proceedings. 020009 (2018).

[3] S. Ishihara, T. Takimoto, R. Endo, A. Tonegawa, K. Sato, K. Kawamura, AIP Conference Proceedings, 050015 (2018).

Keywords: *neutral beam heating, negative ion source, volume production, Cs-free, sheet plasma, TPD-type plasma source*

****Corresponding author:*** *8bsnm008@mail.u-tokai.ac.jp*

Detail Design of In-Vessel Components of ITER Neutron Flux Monitor Equipped with Microfission Chambers

P2-106

Masao Ishikawa [a]*, Marin Suda [a], Vitaly Krasilnikov [b], Luciano Bertalot [c], Takaki Hatae [a]

[a] *National Institutes for Quantum and Radiological Science and Technology, Ibaraki, Japan*

[b] *Tokamak Energy Ltd, Milton Park, Oxfordshire, UK*

[c] *ITER Organization, St. Paul Lez Durance Cedex, France*

A neutron flux monitor is one of the most important systems in ITER because it provides total neutron source strength and fusion power of ITER. The in-vessel neutron flux monitor equipped with Microfission Chambers (MFCs) is designed by Japan Domestic Agency. In-vessel components of the MFC are exposed to the extreme ITER environment, such as high radiation and high electromagnetic (EM) forces. Furthermore, various accidents are also assumed in ITER Therefore, the in-vessel components need to withstand such ITER environment.

In this study, various analyses and tests have been carried out for the in-vessel components in order to show that they can be applied under ITER harsh conditions. Soundness verification tests such as high-temperature and noise immunity test of in-vessel components show that the MFCs can be operated under high temperature up to 550°C and have the noise resistance in ITER condition. Neutronic analysis also shows the in-vessel components of the MFC can withstand for high radiation environment in ITER for 20 years. An electrical feedthrough is one of the most important components of the MFC because it forms boundary for not only vacuum but also all radioactive and toxic substance of ITER tokamak. Therefore, even if accidents happen such as earthquake and/or fire, the feedthrough needs to keep its confinement function. Environmental tests were conducted for the feedthrough. The results indicate that the feedthrough can maintain its confinement function even if accidents assumed in ITER occur.

Above results indicate that the in-vessel components of the MFC can be used in the extreme ITER environments without any replacements.

Keywords: *Neutron diagnostics, fission chamber, In-vessel components, electrical feedthrough*

****Corresponding author:*** *ishikawa.masao@qst.go.jp*

Analysis of the Influence of the Different Impurity Seeding on the Burn-up Fraction and Plasma Confinement in the EU DEMO Reactor

P2-107

Irena Ivanova-Stanik [a]*, Michal Poradziński [a], Mattia Siccinio [b,c], Roman Zagórski [a]

[a] *Institute of Plasma Physics and Laser Microfusion, Warsaw, Poland*
[b] *Max-Planck Institute for Plasma Physics, Garching, Germany*
[c] *EUROfusion Programme Management Unit, Garching, Germany*

This work describes integrated numerical modelling applied to EU 2018 DEMO discharges with tungsten wall and SN divertor, using the COREDIV code, which self-consistently solves 1D radial transport equations for plasma and impurities in the core region and 2D multi-fluid transport in the SOL. The model is self-consistent with respect to both the effects of impurities on the fusion power level and the interaction between seeded and intrinsic impurities (tungsten, helium). Focus of the work is to analyze the influence of the impurity seeding on the burn-up fraction ($f_b=2\Gamma_\alpha/\Gamma_{sep}$, where Γ_α indicates the α-particle source intensity whereas Γ_{sep} represents the plasma flux through the separatrix) and plasma confinement in EU DEMO reactor. For the simulation with constant electron density on the separatrix, it is found that impurity seeding has a small influence on the burn-up fraction, which remains around 6 – 7%., but fueling source is reduced from 2.7 to $2\text{x}10^{22}$ 1/s when moving from lowest to highest seeding level. The operating point of a future DEMO fusion reactor with impurity seeding needs to be optimized with respect to the following criteria: first - the power density on the divertor needs to be smaller than technological limit (5 MW/m^2); second – operating in H-mode requires that the power to the scrape-off layer SOL (P_{SOL}) remains higher than the L-H threshold ($P_{SOL} > P_{L-H}$) and third – the fusion power should be maximized. For the reactor, the fulfillment of the first criterion requires the use of high Z impurities (low dilution, higher P_α) in the core, but high Z impurity might radiate strongly in the core region and reduce confinement, thus deteriorating the power generation. The impact of an increase in the impurity seeding depends on which parameter is assumed constant in the simulations. For this reason, we have done simulations both for cases: with fixed H_{98} factor and with constant particle transport coefficients for different impurity seeding (Ar, Kr and Xe). Strong degradation of confinement, which is correlated to radiation in core (due to intrinsic W and Xe as high Z seeded impurity is observed.

Keywords: *DEMO, impurity, integrated modelling, core plasma, edge plasma, burn-up*

Corresponding author: *irena.ivanova-stanik@ifpilm.pl*

Analysis of Transmutation of Candidate Sensitive Layer Materials of Hall Detectors under DEMO Like Neutron Fluxes

P2-108

Karel Kovarik [a]*, Slavomir Entler [a], Ivan Duran [a], Tim Eade [b]

[a] *Institute of Plasma Physics of the Czech Academy of Sciences, Prague, Czech Republic*
[b] *Culham Centre for Fusion Energy, Abingdon, Oxfordshire, UK*

Steady state magnetic sensors based on the Hall effect are one of the candidate diagnostic concepts for measurement of plasma current, plasma position and shape on the future European demonstration fusion reactor (DEMO). Design of any diagnostic component has to take into account the foreseen high neutron fluences which will be accumulated over DEMO's operational time. The total DEMO life time neutron fluence is expected to reach up to 10^{26} m^{-2} for locations at the back side of the blanket and up to 10^{23} m^{-2} for locations on the outer vacuum vessel surface. At these levels of neutron loads transmutation of materials becomes an effect which has to be considered, particularly for sensors placed in-vessel.

First, the neutron spectra and flux rates are determined using MCNP6 code [1] for a selected locations considered to be instrumented by Hall sensors on DEMO. Then, using the realistic neutron spectra, inventory simulations are conducted with FISPACT-II [2] for a set of materials presently considered for sensitive layers of the DEMO Hall sensors. Resulting information on evolution of material composition of sensitive layers of Hall detectors along the DEMO operation will be summarized. Comparison of simulations of various irradiation time patterns e.g. continuous versus more realistic DEMO pulse regime will be presented.

[1] *D. Pelowitz et al. MCNP6 Users' Manual – Code Version 1.0, LA-CP-13-00634 Rev 0, May 2013.*

[2] *J.C. Sublet et al., 'The FISPACT-II User Manual', CCFE-R(11)11 Issue 7, Sept 2015.*

Keywords: *Hall sensors, neutron irradiation, DEMO*

Corresponding author: *kovarik@ipp.cas.cz*

Deconvolution of Neutron Spectrum for the DT Neutron Generator Based on a Combination of Activation Method, Unfolding Processes and Numerical Simulations

P2-109

Ewa Laszynska *, Katarzyna Mikszuta

Institute of Plasma Physics and Laser Microfusion, Warsaw, Poland

Neutron generators (NGs) are available worldwide as convenient and attractive alternatives to nuclear reactors, fusion devices and isotope-based neutron sources. The technology has developed during recent years and various types of NGs are increasingly used in a variety of applications. They are widely applied in research, medicine, security, nutrition, environmental and geological areas. NGs are also used as the calibration neutron sources during in-vessel calibration of neutron detectors for fusion devices. In virtually all of these applications, precise knowledge of the total neutron emission Y_n, contribution and its energy distribution is required. When this information is not provided by the manufacturer, it should be determined based on measurements and numerical simulations. The method of neutron spectra determination of the DT NG based on combination of activation measurements, unfolding processes and numerical simulations is described here. The threshold nuclear reactions, selected based on many requirements, have been applied during the experimental stage. Additionally, the numerical simulations using the FISPACT-II code have been performed in order to estimate of default neutron spectrum. Obtained results have been applied as an input data to unfolding codes: Gravel and MAXED. Different values of changeable deconvolution parameters (the number of iterations, chi-square for the Gravel code and temperature, reduced temperature factor in the case of the MAXED code) defined by the user have been investigated. The final reconstructed neutron spectrum of the DT NG is a combination of spectra obtained for both deconvolution codes. The agreement with experimental data within 5% has been achieved.

Keywords: *DT neutron generator, activation method, FISPACT-II simulations, deconvolution of neutron spectrum*

Corresponding author: *ewa.laszynska@ifpilm.pl*

Commissioning and Initial Operation of the W7-X Neutral Beam Injection Heating System

P2-110

Paul McNeely [a]*, S. Äkäslompolo [a], W. Auerweck [c], Y. Drider [a], O. Ford [a], D. Hartmann [a], B. Heinemann [b], S. Heinrich [a], C. Hopf [b], R. Kairys [a], S. Obermeyer [b], R. Riedl [b], P. Rong [a], N. Rust [a], R. Schroeder [a], R. Wolf [a] and the W7-X Team

[a] *Max-Planck-Institut für Plasmaphysik-Teilinstitut Greifswald, Greifswald, Germany*
[b] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[c] *Siemens AG, München, Germany*

The first, of two planned, neutral beam injectors for the stellerator W7-X was commissioned for and participated in the experimental campaign (OP1.2b) from July to October 2018. The injector is a slightly modified version of the ASDEX Upgrade Tokamak NBI system that has been in operation since 1990. The injector was equipped with two RF driven ion sources from which 90A of hydrogen ions (H^+, H_2^+, H_3^+) were extracted at 55 kV. After neutralization, the two sources provided >3 MW of neutral beam heating power to the stellerator plasma. During the experimental campaign >300 shots were successfully performed for plasma heating or to allow for measurement of the ion temperature profile by charge exchange recombination spectroscopy (CXRS). This paper will present a summary of the source conditioning and performance, plus describe one of the challenges overcome during commissioning: failure to couple RF power into one of the sources. The initial operation of the NBI system on W7-X was very successful, demonstrating both increased central plasma density and stored plasma energy, or allowing for the collection of the time resolved ion temperature over the bulk of the plasma. An example of each, from the initial operation phase, will be given in the paper. Lastly, a short summary of the upgrades for the next experimental campaign. The primary of which will be the doubling of the available power by commissioning of the second injector box.

Keywords: *NBI, Ion Source, Plasma Heating*

Corresponding author: *p.mcneely@ipp.mpg.de*

A Total Neutron Yield Constraint Implemented to the RNC Emissivity Reconstruction on ITER Tokamak

P2-111

Katarzyna Mikszuta [a,b]*, Martin Imrisek [b], Daniele Marocco [c], Jan Mlynar [b,d], Ondrej Ficker [b,d]

[a] *Institute of Plasma Physics and Laser Microfusion, Warsaw, Poland*
[b] *Institute of Plasma Physics of the CAS, Prague, Czech Republic*
[c] *Association Euratom-ENEA, C.R. Frascati, Frascati, Roma, Italy*
[d] *Czech Technical University in Prague, Prague, Czech Republic*

The purpose of the ITER Radial Neutron Camera (RNC) is the measurement of the plasma neutron emissivity profile for burn control. The RNC is a multichannel and fan-shaped detection system. The present design consists of 22 collimated detector systems and provides a line-integrated neutron measurement with full coverage of the plasma poloidal cross-section. The neutron spatial distribution determination requires to use the sophisticated tomography methods. This paper concerns the evaluation of the improvement in the RNC neutron emissivity reconstruction capability obtained by implementation of the total neutron yield provided by an independent diagnostics as an additional constraint in the tomography procedure. The simulated count rates data for the ITER DT 15 MA scenario and tomography code based on the Minimum Fisher Regularization (MFR) method were used to test the improvement in reconstruction. The original code has been adapted specially to the RNC case. The paper contains the comparison of the tomography for two RNC layouts. The reconstruction accuracy obtained with total neutron yield constraint is generally better than that without this value except that in several pixels on the last closed flux surface. Additionally, a robustness test has been made. The results show that the reconstruction with added constraint was more stable and calibration of the profile's absolute values was better.

Acknowledgments: The work leading to this publication has been funded partially by Fusion for Energy under the Contract F4E-FPA-327. This publication reflects the views only of the author, and Fusion for Energy cannot be held responsible for any use which may be made of the information contained therein.

Keywords: *ITER, diagnostics, Radial Neutron Camera, tomography*

Corresponding author: *Katarzyna.mikszuta@ifpilm.pl*

Deployment of Multiple Shattered Pellet Injection System in KSTAR

P2-112

SooHwan Park [a]*, KunSu Lee [a], Larry R. Baylor [b], Steven J. Meitner [b], HyunMyung Lee [a], JaeIn Song [a], SangWon Yun [a], Jayhyun Kim [a], KwangPyo Kim [a], KapRai Park [a], SiWoo Yoon [a]

[a] *National Fusion Research Institute, Daejeon, Korea*
[b] *Oak Ridge National Laboratory, Tennessee, USA*

Shattered pellet injection system is one of attractive ways to mitigate the plasma disruption in fusion research facilities up to now. DIII-D has already installed it and achieved the meaningful results. ITER (International Thermonuclear Experimental Reactor) decides to adopt this system as a DMS (disruption mitigation system) for PFPO-1 (Pre-Fusion Power Operation phase 1). The validation between model and experiment, and technology development should be carried out to meet the DMS's requirement and reliability.

KSTAR (Korea Superconducting Tokamak Advanced Research) is a possible candidate to test the urgent issues of plasma disruption for ITER. KSTAR can install two injectors in toroidal opposite positions. For this work, ORNL (Oak Ridge National Laboratory) will deliver the two injectors, the shatter tubes and auxiliary system. NFRI is preparing the infrastructure of a pumping system, control and data acquisition system, and changing the diagnostic system and the heating system for the injector location. This presentation describes the requirements for multi SPI injectors and the engineering challenges to be solved for successful deployment and operation in 2019.

Keywords: *plasma, disruption, mitigation, shattered, pellet*

Corresponding author: *parksh@nfri.re.kr*

Progress of Plasma Scenario Modeling for JA DEMO

P2-113

Yoshiteru Sakamoto, Nobuhiko Hayashi, Shinsuke Tokunaga, Nobuyuki Aiba, Akinobu Matsuyama, Nobuyuki Asakura, Ryoji Hiwatari, Hiroyasu Utoh, Youji Someya, Yuki Homma, Yuya Miyoshi, Kenji Tobita and the Joint Special Design Team for Fusion DEMO

National Institutes for Quantum and Radiological Science and Technology, Japan

In order to proceed the integrated core plasma scenario modeling of JA DEMO, (i) core transport simulation, (ii) vertical stability and (iii) MHD stability analyses have been performed. On the core transport simulation

by 1.5-D time-dependent transport code (TOPICS), the steady-state operation condition with HH_{98y2} = 1.41, betaN = 3.6, f_{BS} = 0.69 is demonstrated by optimizing the heating scenario, where CDBM transport model is used. It is also demonstrated that off-axis ECCD (30MW, O-mode, 190GHz) has important roles for maintaining the internal transport barriers (ITBs) for steadystate condition and for controlling the fusion power by control of ITB location. On the vertical stability analysis, the ramp-up scenario of high elongated plasma has been developed by using the plasma equilibrium simulator MECS with 3D eddy current effects. The temporal evolutions of the poloidal beta and internal inductance are evaluated using TOPICS. The result indicates that plasma elongation at 95% of poloidal flux of 1.75 is achievable in JA DEMO. Regarding the MHD stability analysis, the beta limit of JA DEMO plasma has been evaluated by using the linear ideal MHD stability code MARG2D. The beta limit without conducting wall is betaN ~ 2.6, while that with conducting wall can be improved to ~3.5 at the wall radius of r_W/a = 1.35. Further improvements are observed with decreasing the wall radius, for example betaN ~ 3.9 at r_W/a = 1.30.

Keywords: *DEMO, plasma scenario modeling, vertical stability, MHD stability*

****Corresponding author:*** *sakamoto.yoshiteru@qst.go.jp*

Automatic Line Identification Algorithm of Impurity Spectra for Real-Time Feedback in Fusion Plasmas

P2-114

Haewon Shin [a,b], Inwoo Song [a,b], YoungHwa An [c], C. R. Seon [d], Wonho Choe [a,b] *

[a] *Korea Advanced Institute of Science and Technology (KAIST), Daejeon, Korea*
[b] *Impurity and Edge plasma Research Center, KAIST, Daejeon, Korea*
[c] *National Fusion Research Institute, Daejeon, Korea*
[d] *ITER Organization, Saint Paul lez Durance, France*

Accumulation of impurities in fusion plasmas may lead to fuel dilution and severe degradation of plasma confinement. Therefore, real-time monitoring and identification of impurities inside the plasma are strongly required for long stable plasma operation in fusion devices including ITER. These can be achieved by using impurity related spectroscopic diagnostic systems, and it is important for such diagnostics to have a capability of reduction of radiation noise level and in situ calibration of wavelength from the acquired measurement data. In this work, a numerical code for automatic identification of impurity emission lines, equipped with radiation noise reduction and in-situ wavelength calibration modules, has been developed for the VUV spectroscopic diagnostic system in KSTAR tokamak. Firstly, the noise reduction module utilizes a higher-order derivative method that distinguishes relatively sharp noises from the measured line spectra, and phantom tests showed excellent noise reduction over 90%. Secondly, an in-situ wavelength calibration algorithm was developed to minimize the possible miscalibration due to mechanical vibration of the diagnostics. This was performed by using representative carbon emission lines as markers that appear during the current ramp-up phase of the plasma. After these processes of the measured data, a matching algorithm was applied that annotates an ionization stage of an impurity on a spectral image by referencing from the NIST atomic database. Using the developed integrated algorithm, the measured VUV spectra show successful automatic annotation of the spectral lines with significant noise reduction.

Validation of real-time performance will be conducted in the following KSTAR campaign.

Keywords: *real-time post processing, VUV spectroscopy, noise removal*

****Corresponding author:*** *wchoe@kaist.ac.kr*

Wide-Angle Visible Video Diagnostics for JT-60SA Utilizing EDICAM

P2-115

Tamás Szepesi [a]*, Sam Davis [b], Nándor Hajnal [b], Kensaku Kamiya [c], Gábor Kocsis [a], Ákos Kovácsik [d], Naoyuki Oyama [c], Carlo Sozzi [e], Sándor Zoletnik [a]

[a] *Wigner RCP, Budapest, Hungary*
[b] *Fusion for Energy, Garching bei München, Germany*
[c] *QST, Naka, Japan*
[d] *Institute of Nuclear Techniques, BME, Budapest, Hungary*
[e] *Istituto di Fisica del Plasma CNR, Milano, Italy*

A fast wide-angle visible video diagnostic system, based on the Event Detection Intelligent Camera (EDICAM), featuring real-time image processing capabilities, is produced in Europe for the JT-60SA tokamak, to measure also the visible light emission associated with fast phenomena such as plasma start up, disruptions, plasma-wall interaction or gas injection. The EDICAM camera system, including data acquisition and control software, optics, mechanics, as well as the vacuum port plug and protective measures such as a pin-hole heat shield and shutter, is manufactured by Wigner RCP, Hungary. As the EDICAM is able to tolerate magnetic fields of at least 3 T, the camera is located within a horizontal, radial immersion tube ("port plug") under atmospheric conditions. The field-of-view is 80 degrees, with a depth-of-field of 5 m (3-8 m distance from the pin-hole). The special, custom-made optics is also compatible with the tokamak environment, providing also the possibility to apply an interference filter. A water-cooled stainless steel heat shield is protecting the vacuum window at the plasma end of the port plug, equipped with a 5 mm diameter pin-hole, which is an integral part of the optical system. The whole detector assembly (camera + optics) is mounted on a retractable carriage, allowing easy access to these components at any time. This contribution summarizes the design and the installation of the diagnostic system at JT-60SA.

Keywords: *EDICAM, video diagnostics, JT-60SA, real-time, image processing*

Corresponding author: *szepesi.tamas@wigner.mta.hu*

Achievement of DC 1 MV Insulation in High-Voltage Power Supply for ITER Neutral Beam Test Facility

P2-116

Hiroyuki Tobari [a], Mieko Kashiwagi [a], Kazuhiro Watanabe [a], Tetsuya Maejima [a], Yasuo Yamashita [a], Marco Boldrin [b], Loris Zanotto [b], Samuele Dal Bello [b], Vanni Toigo [b], Muriel Simon [c], Gerard Escudero Gomez [c], Francesco Paolucci [c], Tullio Bonicelli [c], Hans Decamps [d]

[a] *National Institute of Quantum and Radiological Science and Technology, Naka, Japan*
[b] *Consorzio RFX (CNR, ENEA, INFN, UNIPD, Acciaierie Venete SpA), Padova, Italy*
[c] *Fusion for Energy, Barcelona, Spain*
[d] *ITER Organization, St. Paul-lez-Durance, France*

Toward the ITER neutral beam (NB) system operating at DC -1 MV, several insulating techniques with oil, gas (air, SF_6) and in vacuum have been developed. For the NB Test Facility (NBTF) established in Padua (Italy), the DC 1 MV power supply (PS) components such as transformers, transmission line, HV Deck and HV bushing have been manufactured, installed and the on-site acceptance tests have started to ensure compliance with the ITER requirement in advance of the ITER. The 1 MV acceleration voltage to produce deuterium negative ion beams with 1 MeV, 40 A for 3600 sec is obtained with five-stage (200kV each) DC generator (transformer and rectifier) and transmitted to the beam source in a single SF_6-gas insulated transmission line (pressurized tanks) with length of ~100 m. During the installation of the transmission line, increase of contact resistance at connection points of the inner conductor inside the transmission line were observed. For such a long component, the tanks cannot be closed just after connecting the conductor. Thus, exposure to the air for several months induced the increase of the contact resistance. To avoid an electric and thermal impact on high-current conductors, a recovery technique of the contact resistance without disconnection was investigated. Finally, most of PS component have been installed and the voltage holding test for each section of the PS system as the on-site acceptance test has started. In this paper, recent progress of voltage holding test of the ITER NBTF DC 1 MV high-voltage PS including European component is reported.

Keywords: *ITER, Neutral beam injector (NBI), NBTF, High-voltage power supply*

Corresponding author: *tobari.hiroyuki@qst.go.jp*

Estimation of Magnetic Error Field with Alleviating Fabrication Tolerance of Large Superconducting Magnets on JA DEMO Reactor

P2-117

Hiroyasu Utoh [a]*, Go Matsunaga [b], Hiro Togashi [c], Ryoji Hiwatari [a], Yoshiteru Sakamoto [a], Kenji Tobita [a] and Joint Special Design Team for Fusion DEMO

[a] *National Institutes for Quantum and Radiological Science and Technology (QST), Rokkasho, Aomori, Japan*
[b] *National Institutes for Quantum and Radiological Science and Technology (QST), Naka, Ibaraki, Japan*
[c] *Hitachi, Ltd., Hitachi, Ibaraki, Japan*

In order to describe the relation between error field and superconducting coil tolerance, evaluation of error field based on accuracy of manufacture/assembly of superconducting coils on DEMO were investigated by using analysis code on JT-60SA. As the probabilistic evaluations, the mitigated target of error field B_{err} of about 0.6 mT provides a mitigation in the tolerance in coil fabrication and installation by a factor of 2.5-5 compared with that of ITER, contributing to alleviating the fabrication of larger coils on JA DEMO.

In JA DEMO design activity, superconducting magnet design has been studied for medium DEMO focusing on larger coil size and higher magnetic field (magnetic energy) than ITER superconducting magnets. The fabrication tolerance has an effect on engineering difficulties of large coil system. JA DEMO adopts the design strategy of toroidal field (TF) coils of mitigating the error field requirement and the error field is corrected by using an e Error Field Correction Coil (EFCC) if needed to avoid locked modes. In order to determine the EFCC currents, we have applied a least square method to minimize three error field components of m = 1, 2 and 3 with n = 1 at the q = 2 surface. In the estimations, anti-series

connections between coils located at the opposite side toroidally are assumed. As a result, the EFCC current required to correct B_{err} up to 0.1 mT was 200 kAT even when a set of EFCCs are arranged outside the vacuum vessel in a toroidally non-periodic manner to avoid interference with maintenance ports and NBI ports. The result indicates that the error field caused by mitigated tolerances of TF coil fabrication and installation can be corrected with ex-vessel EFCCs at realistic coil currents.

Keywords: *DEMO design, superconducting magnet, error field*

Corresponding author: *uto.hiroyasu@qst.go.jp*

Organizing Wendelstein 7-X Device Operation

P2-118

Paul van Eeten *, H.S. Bosch, S. Degenkolbe and W7-X Team

Max-Planck-Institut für Plasmaphysik, Greifswald, Germany

The super conducting stellarator Wendelstein 7-X started its first operational phase in October 2015 at the Max-Planck-Institute for Plasma Physics in Greifswald with the goal to verify that a stellarator magnetic confinement concept is a viable option for a fusion power plant, i.e. showing confinement comparable to tokamaks and running in steady state operation. Between 2015 and 2018 the first three experimental campaigns (operational phases) OP1.1, OP1.2a and OP1.2b of the W7-X stellarator have been successfully completed. Roughly 13 Months of operational time have been accumulated and have already shown the impressive capability and reliability of W7-X in achieving the physical and technical goals as set by the project.

The department Device Operation (DO) has implemented an organizational structure and workflow to ensure safe and reliable operation of the W7-X device. The device operation team (DOT) consists of the following roles / teams all being present in the central control room:

- Technical Leader
- Technical Leader in charge (during session), TLvD
- Chief Operator (for central Safety System and central Operational Management)
- Operations Teams for Magnets / Cryogenics / Vacuum systems / Gas supply / Technical services / General safety / Radiation safety / Heating Systems / CoDaC / Engineering

DO is responsible for planning and executing the commissioning and operation of the W7-X device. The operations plan is based on the physics planning which is a congregation of individual physics proposals keeping in mind the technical limitations of W7-X. The operations plan is iterated between the lead physicists and DOT. The TLvD plays a central role within the DOT in coordinating and supervising the technical operation of W7-X.

To ensure safe and reliable operation of W7-X several procedures have been implemented and improved. This involves commissioning templates, checklists, procedures for e.g. plasma heating energy release, operation malfunction cards, logbooks, coordination meetings, duty-on-call and shiftplanning.

This paper will summarize the organizational aspects of W7-X technical operation as performed in the first three phases. It will also provide an outlook on the upcoming OP2 (actively cooled divertor) with the requirements for technical operation regarding commissioning and operation towards steady state plasma operation.

Keywords: *Fusion Experiment, Operation, Planning, Organisation*

Corresponding author: *paul.van.eeten@ipp.mpg.de*

Effect of RMP on Boundary Plasma Turbulence and Transport on J-TEXT Tokamak

P2-119

Ting Wu [a]*, L. Nie [a], M. Xu [a], J. Yang [b], R. Ke [a], S. Gong [c], D. Li [b], Y. Yu [c], T. Long [a], Y. Chen [a], Y. Shi [d] and J-TEXT Team

[a] *Southwestern Institute of Physics, Chengdu, China*
[b] *State Key Laboratory of Advanced Electromagnetic Engineering and Technology, School of Electrical and Electronic Engineering, Huazhong University of Science and Technology, Wuhan, China*
[c] *Department of Nuclear Science and Technology, University of Science and Technology of China, Hefei, China*
[d] *Department of Nuclear Engineering, Seoul National University, Seoul, Korea*

In tokamaks resonant magnetic perturbation (RMP) have changed the magnetic topology and affected plasma transport and confinement. This paper presents how RMP influences the boundary plasma turbulence and transport on J-TEXT tokamak. By comparing the difference of boundary turbulence and transport under RMP with 6 kA current and 0 kA measured by Langmuir probe, it has been found that the turbulence in the edge (radially inside last closed flux surface) and Scrape-Off Layer (SOL) has been suppressed dramatically. Blob transport, which dominates the SOL transport, has been reduced strongly in the near SOL region. These results suggest that RMP can be a useful tool to reduce edge and SOL turbulence and transport.

Keywords: *RMP, turbulence transport, blob*

Corresponding author: *wuting@swip.ac.cn*

Propagation and Suppression of Tent-Induced Perturbations in the ICF Implosion

P2-120

Zhensheng Dai *, Jianfa Gu, Wudi Zheng

Institute of Applied Physics and Computational Mathematics, Beijing, China

In the inertial confinement fusion (ICF) scheme, to trap the alpha particle products of the D-T reaction, the capsules needs to be imploded and compressed with high symmetry[1]. In the laser indirect drive schemes, the capsules are held at the center of high-Z hohlraum by thin membranes (tents). However, the tents are recognized as one of the most important contributors to hot spot asymmetries, areal density perturbations and reduced performance.[2,3]

To improve the capsule implosion performance, people have to mitigate the perturbations induced by the supports such as the tents. various alternatives such as the micro-scale rods, a larger fill-tube and a low-density foam layer around the capsule have been presented.[4]. We simulated the propagation of tents and silks induced perturbations in the introduced low-density foam layer. Our simulations show that the perturbations can be well suppressed.

[1] N.B. Meezan, M.J. Edwards, O.A. Hurricane, et al. Indirect drive ignition at the National Ignition Facility, Plasma Phys. Control. Fusion 59, 014021(2017)

[2] R. Tommasini, J. E. Field, B. A. Hammel, O. L. Landen, S. W. Haan, et al. Tent-induced perturbations on areal density of implosions at the National Ignition Facility, Phys. Plasmas 22, 056315(2015)

[3] S.R. Nagel, S.W. Haan, J.R. Rygg, et al. Effect of the mounting membrane on shape in inertial confinement fusion implosions, Phys. Plasmas 22, 022704(2017)

[4] C.R. Weber, D.T. Casey, D.S. Clark, B.A. Hammel, et al. Improving ICF implosion performance with alternative capsule supports, Phys. Plasmas 24, 056302(2017)

Keywords: *inertial confinement fusion (ICF), tent, hydrodynamic instability, suppression*

***Corresponding author:** *dai_zhensheng@iapcm.ac.cn*

Gradient Nanoporous Gold: A Potential Inertial Confinement Fusion Hohlraum Wall Material

P2-121

He Zhibing *, Huang Jinglin, Liu Yansong

Laser Fusion Research Center of CAEP, Mianyang, China

Reducing hohlraum wall loss by the optimized design of wall materials is an important way to improve hohlraum coupling efficiency in laser indirect drive inertial confinement fusion (ICF) experiments. The research direction of hohlraum wall materials mainly lies on the fabrication of low density high-Z materials. While according to a recent theoretical research, using graded Au foam as the wall material can save ~40% general wall loss. In this work, to experimentally fabricate this kind of graded hohlraum wall material, we prepared a gradient nanoporous gold (GNPG) film on Si substrate by combining the traditional magnetron sputtering with dealloying technique. The composition of Au-Ag alloy precursor was carefully controlled to create a gradient structure by adjusting sputtering power in co-sputtering process. It was found that the content of Au in Au-Ag alloy precursor presents a linear increase along the film thickness direction, which provides a composition basis for the formation of gradient structure. The surface morphology of GNPG appears a nanoporous reticular structure, while the section morphology appear a graded structure. The Young's modulus and hardness of GNPG film is as high as 122 GPa and 1.07 GPa, respectively. These results indicate that the mechanical property of GNPG structure is much better than bulk gold. At last, by using vacuum coating technology, dealloying technology and pulse plating technology, GNPG hohlraum was fabricated successfully. And the process of the formation mechanism of the GNPG hohlraum was studied.

Keywords: *Inertial Confinement Fusion, Gradient Nanoporous Gold, hohlraum, magnetron sputtering, dealloying*

***Corresponding author:** *He_zhibing@126.com*

Progress on Fabrication of High Density Carbon Capsule for ICF Target in LFRC

P2-122

Liu Yansong *, Wang Tao, He Zhibing

Laser Fusion Research Center of CAEP, Mianyang, China

Fusion promises to offer a clean, inexpensive, efficient, and abundant source of energy for future. As we all know, Inertial Confinement Fusion (ICF) is an alternative way to achieve ignition and utilize fusion energy. In ICF experiment, high density carbon (HDC) is being evaluated as an alternative to the current point-design ablator material due to its high density and optimal opacity, which leads to a higher implosion velocity. The HDC capsules are fabricated exclusively by chemical vapor deposition technology, which is well known today and largely used. But the capsules for laser fusion targets have many stringent characteristics. HDC capsules have a near perfect surface figure but a microscopically rough surface. Even after polishing, the surface becomes smooth at nanometer scales but has numerous micron-sized surface pits. So, there is still need for the research to improve HDC shells meeting all the specifications. In this paper we report the development of HDC coatings focusing on mandrel production, surface morphology and uniform properties of HDC capsules at Laser Fusion Research Center (LFRC). Firstly, Molybdenum (Mo) was selected as mandrel sphere as it is easy to remove. How to fabricate and polish Mo mandrel spheres are introduced. Secondly, HDC coatings were fabricated by chemical vapor deposition concentrated on wall uniform and crystalline properties. At last, due to microscopically rough surface of HDC coatings, polishing technology was used to polish HDC shells focus on surface roughness controlling.

Keywords: *Inertial Confinement Fusion, high density carbon, capsule, polish, surface roughness*

Corresponding author: *Jdliuyansong@163.com*

Research on the Airflow Distribution in the Laser Beam Transport Tube of the ICF Target Area

P2-123

Zhenyuan Xu *, Junwei Zhang, Yong Xiang, Liangming Chen, Xibo Sun, Ye Tian, Tianran Zheng, Ao Cui, Xiaodong Yuan, Fuquan Li

Laser Fusion Research Center, China Academy of Engineering Physics, Mianyang, China

As an important part of inertial confinement fusion device, the function of the laser beam transport tube in the target area is mainly to realize the laser beam switching and accurate transporting the high-energy fundamental frequency laser from the main amplifier system to the final optics assembly. In order to ensure the cleanliness of laser transport and the safety of personnel, the whole laser beam needs to be sealed in transport tube. However, the stimulated rotational Raman scattering effect will occur when the fundamental frequency laser transport length is too long, which will seriously degrade the beam quality and damage the optical elements. A large number of experimental studies show that filling a certain concentration of argon-oxygen mixture gas in the transport tube is an effective measure to reduce the SRRS effect. Therefore, the filling uniformity and state control of the mixed gas in the beam transport tube are the key factors affecting the beam quality. In this paper, aiming at these problems, based on the principle of fluid mechanics, combined with the flow field analysis method and numerical simulation method, the whole process of dynamic replacement of mixed gas in the beam transport tube is analyzed, and an optimized and efficient gas filling and replacement scheme is proposed. Furthermore, the static stratification of the gas mixture in the beam transport tube with a height difference is analyzed. Finally, the effectiveness and feasibility of the simulation results and schemes are confirmed by several experiments, which have important significance to the project implementation.

Keywords: *laser beam quality control, flow field analysis, mixed gas replacement, numerical simulation, inertial confinement fusion*

Corresponding author: *Xuzhenyuan.1@163.com*

Numerical Research on Forming a Uniform Ice Fuel Layer in a Cryogenic Capsule

P2-124

Hong Yang *, Kai Wang, Xiaobo Qi, Baibin Jiang, Jun Xie, Shasha Gao

Research Center of Laser Fusion, China Academy of Engineering Physics, Mianyang, Sichuan, China

In inertial confinement fusion experiments, cryogenic targets with solid hydrogen fuel attached to the inner surfaces of spherical capsules have the most possibility for ignition. Spherical symmetry of the ice fuel shell is a key factor determining the experiment success. Because of small sizes, numerical simulation is the only method of predicting and assessing the ice fuel quality before shots. Based on heat transfer theory, a finite element model with "if" functions is applied to gain ice appearance solution in steady state. Because of beta decay heat, deuterium-tritium (DT) will form a spherical shell naturally but deuterium-deuterium (DD) won't. Another transient finite element model integrating heat transfer theory, Stefan law and mass conservation is implemented to track the interface between ice and gas fuel under specified thermal scenarios. Capsule temperature difference and sustaining time are proved the two factors influencing DD ice spherical symmetry degeneration from an initially uniform layer. But for DT fuel, symmetry degeneration will reach platform after a while. The concluded relationships between ice fuel spherical symmetry and capsule thermal loads are the guidelines of cryogenic target and system engineering design with different hydrogen fuel.

Keywords: *fuel homogenization, cryogenic target, inertial confinement fusion*

Corresponding author: *ricky_oo@163.com*

Analysis of Irradiation Experiments with Activated Water Radiation Source at the JSI TRIGA Research Reactor

P2-125

Andrej Žohar [a], Anže Pungerčič [a], Klemen Ambrožič [a], Vladimir Radulović [a], Igor Lengar [a], **Luka Snoj** [a,b]*

[a] *Reactor Physic Department, Jožef Stefan Institute, Ljubljana, Slovenia*
[b] *Faculty of Mathematics and Physics, University of Ljubljana, Slovenia*

Demineralized water is used as a cooling fluid in majority of fission nuclear reactors and will be used for cooling of several fusion reactors including ITER. When water is exposed to neutrons it gets activated. The most important activation reaction is $^{16}O(n,p)^{16}N$ due to high natural abundance of ^{16}O and emission of high energy gamma rays (~ 6.13 MeV and 7.11 MeV) with 7.13 s half-life. The emitted gamma rays can cause increased doses to personnel, radiation damage to components around cooling systems and in the case of fusion reactors additional nuclear heating to superconducting coils. Fission research reactors present an opportunity to study the effects of activated water decay as currently no fusion reactors fusing deuterium and tritium are capable to perform water activation experiments with sufficient accuracy. An irradiation facility utilizing activated water is planned at the

Jožef Stefan Institute TRIGA Mark II research reactor. The facility features a closed water loop through reactor core or through radial irradiation channel adjacent to the reactor core and a separate shielded room for experiments. The intensity of gamma rays in irradiation room depends on reactor power and adjustable flow rate of water in the loop thus allowing for wide range in source intensity for different experiments.

In the paper an analysis of shielding and nuclear heating experiments due to decay of ^{16}N in the proposed irradiation facility will be analyzed. As the intensity of the gamma ray source can be varied the analysis will cover a wide range of source intensities and fusion relevant shielding materials, such as copper, Eurofer, concrete, Tungsten, etc.

Corresponding author: *luka.snoj@ijs.si*

The p(20)+Be Reaction as a Source of Fusion Relevant Neutrons

P2-126

Milan Stefanik [a,b]*, Pavel Bem [a], Eva Simeckova [a], Jan Stursa [a], Mitja Majerle [a], Vaclav Zach [a]

[a] *Nuclear Physics Institute of The Czech Academy of Sciences p.r.i., Rez, Czech Republic*
[b] *Czech Technical University in Prague, Faculty of Nuclear Sciences and Physical Engineering, Prague, Czech Republic*

The p(20)+Be neutron source reaction was investigated at the Nuclear Physics Institute of the Czech Academy of Sciences using the 20 MeV proton beam on thick beryllium target (thickness of 8 mm). Neutron field in close source-to-sample distances was determined by means of the activation foils technique; set of ten activation foils (Al, Ti, Co, Fe, Ni, Y, In, Nb, Lu, Au) was utilized. Neutron spectrum reconstruction from reaction rates obtained using the gamma-spectroscopy method (semiconductor HPGe detectors) was performed utilizing the modified version of SAND-II unfolding code and relevant activation cross-sections from the EAF-2010 nuclear database. Determined neutron energy spectra were validated against the MCNPX calculations. Developed p(20)+Be neutron field with broad spectrum up to 18 MeV represents a useful tool for the cross-section data validation within the ITER and IFMIF research programs (≥ 12 MeV), neutron-hardness tests of

microelectronics, study of material damage, and calibration of new detection systems.

Keywords: *accelerator-driven fast neutron source, beryllium target, proton induced reaction, multifoil activation technique, SAND-II unfolding, neutron spectrum*

***Corresponding author:** *milan.stefanik@fjfi.cvut.cz*

Parametric Study of S-CO2 Cycles for the DEMO Fusion Reactor

P2-127

Jan Stepanek [a]*, Slavomir Entler [b], Jan Syblik [a], Ladislav Vesely [c]

[a] *Czech Technical University in Prague, FME, Praha, Czech Republic*
[b] *Institute of Plasma Physics of CAS, Prague, Czech Republic*
[c] *Center for Advanced Turbomachinery and Energy Research, University of Central Florida, Orlando, USA*

Effective use of fusion energy goes hand in hand with effective energy recovery of the generated heat. One of the most promising working medium suitable for this purpose is supercritical carbon dioxide (S-CO2) due to its thermal properties. The intent of this paper is to analyse selected S-CO2 cycle layouts heated by multiple heat sources stated by C.Bachman [1] – the first wall, blanket, divertor and vacuum vessel. The analysis of thermodynamic cycles is performed using new custombuilt high-performance computing software. The parametric study of each layout shows the suitability of the given cycle for the heat recovery and its sensitivity on the main heat source with a wide range of source output parameters.

[1] *C. Bachman, Issues and strategies for DEMO in-vessel component integration, Fusion Eng. Des. 112, (2016), pp.527-534*

Keywords: *S-CO2, heat cycle, DEMO, fusion power plant*

***Corresponding author:** *j.stepanek@fs.cvut.cz*

Minor Actinides Transmutation in a Molten Salt Blanket in the Fusion-Fission Hybrid Reactor Core

P2-128

Jing Zhao *, Jiali Zou, Zhihong Liu, Chunlin Wei

Institute of Nuclear and New Energy Technology, Collaborative Innovation Center of Advanced Nuclear Energy Technology, Key Laboratory of Advanced Reactor Engineering and Safety of Ministry of Education, Tsinghua University, Beijing

With the continuous development of nuclear power, the spent fuel accumulation grows rapidly. Proper disposal of spent fuel is required not only by the sustainable development of nuclear energy but also by the environmental protection. The separation-transmutation method is currently the most effective method of nuclear fuel treatment. The deuterium and tritium fusion in the fusion- fission hybrid reactor emits high energy neutrons with energy of 14.1MeV, and the high energy neutrons have a good transmutation effect on MA. Under the premise of satisfying the certain magnification and tritium multiplication ratio of the hybrid reactor, a molten salt blanket for MA transmutation was designed. The neutron characteristics and transmutation characteristics of the blanket were analyzed.

Keywords: *fusion-fission hybrid, molten salt, blanket, transmutation, Serpent*

***Corresponding author:** *zhao_jing@tsinghua.edu.cn*

Analysis of Water Activation in Fusion and Fission Nuclear Facilities

P2-129

Andrej Žohar *, Igor Lengar, Luka Snoj

Reactor Physics Department, Jožef Stefan Institute, Ljubljana, Slovenia

Water will be the cooling fluid in ITER and is under consideration to be the cooling fluid in future fusion power reactors. As the cooling water is exposed to high energy neutrons (~14 MeV) emitted from fusion of deuterium and tritium (DT) it gets activated. The decay of activated cooling water presents a significant radiation source around primary cooling system causing radiation damage to electrical components, increased doses to personnel and additional nuclear heating to superconducting coils. The major contributors to the activity of the cooling water are the activated oxygen nuclides. The most important activation reactions are on oxygen isotopes $^{16}O(n,p)^{16}N$, $^{17}O(n,p)^{17}N$ and $^{18}O(n,\gamma)^{19}O$. ^{16}N and ^{19}O are gamma emitters (6 and 7 MeV for ^{16}N and 0.2 and 1.4 MeV for ^{19}O) and ^{17}N decays by emitting neutrons (1 MeV). Activation of ^{16}O and ^{17}O are threshold reactions with thresholds at 10

and 8 MeV while ^{18}O can already be activated by thermal and epithermal neutrons. As currently no experiments for water activation in fusion reactors exist, computational methods are needed to determine water activation in fusion reactors and radiation damage to equipment and doses to personnel around cooling systems. An analysis of activation of all three isotopes of oxygen using Monte Carlo method will be presented in the paper. As the calculated activation of oxygen isotopes strongly depends on the cross-sections in evaluated nuclear data libraries, cross-sections from several up to date nuclear data libraries, such as FENDL-3.1d, ENDF/B-VIII.0, TENDL-2017 and JEFF-3.3 will be analyzed and compared with taking uncertainties in the cross-sections for water activation reactions into account. The analysis of water activation and activity in the cooling system with all studied nuclear data libraries will be performed for a fusion research reactor with 500 MW thermal power.

Calculated results of cooling water activation in fusion reactors are compared to calculated results for fission reactor as radiation and nuclear heating of activated water decay in fission reactors can used for extrapolation to conditions in fusion reactors.

Keywords: *Activated cooling water, Cross-sections, Monte Carlo, Research reactors*

Corresponding author: *andrej.zohar@ijs.si*

Towards Autonomous Robotic Systems in Nuclear Fusion Plant

P2-130

Pragna Das [a]*, Emily C. Collins [b], Robert Skilton [c]

[a] *RACE, UK Atomic Energy Authority, Culham, UK*

[b] *Department of Computer Science, University of Liverpool, Liverpool, UK & RACE, UK Atomic Energy Authority, Culham, UK*

[c] *RACE, UK Atomic Energy Authority, Culham, UK*

Remote handling and maintenance tasks within the Joint European Torus (JET) tokamak are currently conducted using the teleoperation MASCOT device. However, tele-operable robotics are not immune to problems arising out of human error and long execution time of tasks. This work proposes that an intelligent and autonomous decision-making control system could reduce these problems by introducing a mechanism of self-evaluation. This is a factor to measure the decision-making capabilities of robots without human intervention, termed as degree of autonomy. Once a measurement of degree of autonomy is established, a typical human-in-the-loop robotic teleoperation control-systems can be enriched by employing different levels of local autonomy in different areas. The first step towards such a system is to understand the areas where failures are occurring more and the reasons for long execution time. This aids to understand the areas where some level of autonomy could be introduced. This proposal represents a paradigm shift from tasks conducted with human-in-the-loop robotic teleoperation, towards autonomous control-systems. Unlike general task-based autonomous systems, the focus for autonomous fusion remote handling and maintenance is oriented on capability or cognition of the autonomous system, rather than on specific tasks. This is for the security and robustness needed in remote handling in fusion tokamaks. The decision-making scope of the robots will be varied from zero-autonomy to full-autonomy, while decreasing the operators' intervention from full-control to zero-control. Various degrees of autonomy could be quantified at different scopes based on the percentage of logical correctness, erroneous decision, execution time of task, occurrence of failure is measured. It paves the way for understanding the autonomy for dexterity, robustness, and immunity from failures in robotic remote handling tasks conducted in fusion tokamak such as JET.

Keywords: *Tele-operation, autonomous system, autonomy, robots, remote handling*

Corresponding author: *pragna.das@ukaea.uk*

Master Equation Study of HT Production in Tritium Breeding Blanket Purge Gas Flow

P3-001

Michiko Ahn Furudate [a]*, Denis Hagebaum-Reignier[b], and Seungyon Cho[c]

[a] *Chungnam National University, Daejeon, Korea*

[b] *Aix Marseille University, Marseille, France*

[c] *National Fusion Research Institute, Daejeon, Korea*

Some experiments have indicated that a certain portion of tritium is released in the form of HTO from breeder materials of solid breeder blankets. Because extracting tritium element from HTO is more complicated and expensive, it is desirable to recover HT from HTO as much as possible in the tritium breeding blanket. By adding a small amount of H_2 in the He purge gas, HT is expected to be produced by the isotope exchange reactions between HTO and H_2. Therefore, it is important to clarify the detailed chemical reactions processes between HTO and H_2, and to predict the rate of HT production by these reactions in the breeding blanket.

In the previous work of the authors, the chemical reaction processes to product HT molecules in the mixture of HTO and H_2 gas have been studied. In this study, the chemical reaction rate coefficients for the HT

production processes are derived, and the master equations for the chemical species concentration are solved to discuss the HT production rate in the Helium Cooled Ceramic Reflection (HCCR) Tritium Extraction System (TES).

Keywords: HT production, Chemical Kinetics, Purge Gas Flow, Tritium Breeding Blanket

**Corresponding author: furu@cnu.ac.kr*

Construction and Inactive Commissioning of a High Throughput Micro-Channel Reactor for Tritiated (Heavy) Water Production

P3-002

George Ana *[a], Gheorghe Pasca [b], Mirela Draghia [b], Anisia Bornea [b], Ciprian Bucur [a], Marius Zamfirache [a], Alina Niculescu [a]

[a] *National R&D Institute for Cryogenics and Isotopic Technologies, Rm. Valcea, Romania*
[b] *ISTECH, Timisoara, Romania*

A pilot plant for tritium and deuterium separation is under commissioning in Institute for Cryogenic and Isotopic Technologies (ICSI) from Rm. Valcea, Romania. During cleaning and purging of the plant systems will result tritiated gases that need to be processed before sending to the stack. The technology behind the system that will deal with the detritiation of the purge gases is based on catalytic hydrogen oxidation and tritiated (heavy) water production for safe storage. A tritium compatible Pt based micro-channel catalytic reactor with channels hydraulic diameter of 0.5 mm has been designed and manufactured to process a flow of tritiated purge gases up to 5 Nm^3/h with 20% hydrogen and produce up to 15 mL min^{-1} of tritiated (heavy) water. One of the operating procedure of the plant requires for the micro-reactor to be able to oxidize pure tritiated hydrogen and produce tritiated water with specific tritium activity of 1.11×10^{12} Bq kg^{-1}.

Prior to its integration in the plant systems, this equipment has been commissioned at different feed flow rates, gas compositions and temperatures. The results will demonstrate the performances of the micro-reactor in hydrogen oxidation and water production when fed with purge gases. Also preliminary data will be shown on the performances of the equipment when pure hydrogen is oxidized.

**Corresponding author: george.ana@icsi.ro*

Keywords: tritium, micro-channel reactor, tritiated water

Development of DT-fueling and Pumping Systems for DEMO-FNS Hybrid Facility

P3-003

Sergey S. Ananyev [a]*, Alexei Yu. Dnestrovskij [a], Andrei S. Kukushkin [a,b], Boris V. Ivanov [a], Alexander V. Spitsyn [a] and Boris V. Kuteev [a,b]

[a] *NRC Kurchatov Institute, Moscow, Russia*
[b] *NRNU MEPhI; Moscow, Russia*

The fueling and pumping systems of DEMO-FNS hybrid facility provide plasma feeding by DT-isotopes for maintaining stationary fusion and removal of excess particles from the divertor zones. Simulation of the hydrogen isotope fluxes in fueling systems of a tokamak-based fusion neutron source with the parameters R/a = 3.2 m/1 m, B = 5 T, I_{pl} = 4-5 MA, P_{NBI} = 30 MW and P_{ECR} = 6 MW is performed using "FC-FNS" fuel cycle (FC) model. FC systems pump gases from the top and bottom divertors, pre-purify and separate the impurity gases from the hydrogen isotopes, and purify the chemically-bounded hydrogen isotopes further. These systems store also the isotope reserves and provide fuel injection into the plasma ensuring the burning conditions, plasma heating and current drive (by a neutral beam injection system - NBIs) provides off-gases detritisation and processes waste. The FC model simulates the processes mentioned above together with tritium breeding in the blanket.

Since the fuel flows inside the vacuum chamber are distributed over the core and edge plasmas, a coupled modeling of the gas, solid and plasma flows of the fuel mixtures in these areas was performed for the first time. We have chosen a scenario for operation of the NBIs gas supply system and determined the tritium/deuterium ratio in the plasma composition that provides the highest fusion power. In the present work we consider the candidate technologies for hydrogen extraction, isotope separation, detritisation off-gases. Produced a preliminary analysis of the effectiveness of the most promising technologies. The principles of further optimization of the technologies are also under consideration. Various options of D_2/T_2/DT pellet injection from the HFS and LFS directions depending on the core plasma parameters have been considered. Injection from the LFS is considered the ELM control that should not affect fueling of the core significantly. Major tritium inventories will be in systems with a cyclic operation mode - these are the systems for extracting/separating/purifying the hydrogen isotopes. Optimization of these systems is an important task. Integral fuel flow through FC in the optimal system is $3\cdot10^{21}$ atom/s. The results obtained

have allowed us to choose the system configuration with the minimum tritium inventory (< 2 kg) and the lowest reprocessing time.

This work was partially supported by the Russian Science Foundation (grant № 18-72-10162).

Keywords: *hydrogen isotopes, fusion fuel cycle, fusion-fission reactor, D–T simulation, fusion neutron source, hybrid facility, simulation model, core and divertor plasma modeling*

**Corresponding author:* *Ananyev_SS@nrcki.ru*

HgLab Karlsruhe – An Infrastructure Facility to Support the Development of DEMO Vacuum Pumps with Mercury as Working Fluid

P3-005

Katharina Battes *, Thomas Giegerich, Christan Day

Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany

For the EU DEMO, a new vacuum pumping process for exhaust gases, the so-called KALPUREX® process (Karlsruhe liquid metal based pumping process for fusion reactor exhaust gases), is under development. This pumping system consists of three different types of pumps, namely a metal foil pump for gas separation, a linear diffusion pump as primary pump and a liquid ring pump as backing pump. The latter two pumps apply mercury as working fluid due to its perfect tritium compatibility.

Mercury (Hg) was previously used as a working medium for diffusion pumps, but for the liquid ring pumps it is even the first application with mercury, whose successful demonstration was shown by a proof-of-principle experiment in the THESEUS pump test facility at KIT in 2014.

As mercury is toxic, a dedicated working environment has been set up at KIT to examine the tested mercury ring pump and further analyze the properties of mercury. In the KIT mercury laboratory, called HgLab, working with Hg is possible at hazardous material work stations and under local exhaust system units. In addition, an inductively coupled plasma - mass spectrometer (ICP-MS) is available in the HgLab. With this highly sensitive analytical instrument for trace metal analysis, impurities in mercury can be detected.

The paper describes the layout of the KIT mercury laboratory HgLab under consideration of the necessary safety requirements and shows investigations of different mercury samples by ICP-MS. In addition, cleaning methods for mercury are established and their effect examined.

Keywords: *mercury, DEMO, liquid ring pump, diffusion pump, KALPUREX*

**Corresponding author:* *katharina.battes@kit.edu*

Design of the EU DEMO Tritium Extraction System Based on Permeation Against Vacuum Technology

P3-006

Roberto Bonifetto [a]*, Marco Utili [b], Laura Savoldi [a], Domenico Valerio [a]

[a] *NEMO group, Dipartimento Energia, Torino, Italy*

[b] *ENEA, Centro Ricerche Brasimone, Italy*

The recently updated European Roadmap towards electricity from fusion energy foresees the design and realization of the EU DEMO, the first demonstrator that will follow the experimental tokamak ITER. One of the DEMO targets will be the demonstration of tritium self-sufficiency, thanks to a Breeding Blanket (BB).

One of the proposed BB designs, namely the Water-Cooled Lithium-Lead (WCLL), adopts flowing PbLi as breeder material. To close the fuel cycle, the tritium contained in the PbLi mass flow rate exiting the BB must be extracted in the Tritium Extraction System (TES).

Different technologies are being proposed for the TES: in the Permeator Against Vacuum (PAV) the liquid PbLi with dissolved tritium flows through channels delimited by a membrane permeable to tritium, so that it can reach the secondary side, where vacuum is kept.

The design of the TES foreseen for the EU DEMO based on the PAV technology is reported here. It is based on the shell-and-tube geometry: the PbLi exiting the BB is forced to pass through a bundle of parallel tubes; the tritium in the PbLi diffuses to the cylinder (the shell) where the bundle is inserted and tritium is pumpied out.

The geometry and material of the pipe bundle and of the shell are defined in detail, as the result of an optimization process aimed at satisfying all the constraints (space occupation, pressure drop, extraction efficiency). The thermal-hydraulic and mechanical analysis of the TES is presented to assess the compliance with the specification and confirm the design, accounting also for safetyrelated issues such as the possibility to drain all the PbLi to avoid its freezing.

Finally, the piping and instrumentation diagram of the TES is shown, including also all the sensors needed for the system control and the safety diagnostics.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No. 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *EU DEMO, Tritium extraction system (TES), Permeator Against Vacuum (PAV), Breeding blanket*

**Corresponding author:* *roberto.bonifetto@polito.it*

Investigations Concerning the Influence of the Catalyst / Package Ratio and the Catalyst Distribution on the Separation Performances of a LPCE Process

P3-007

Anisia Bornea [a], George Ana [a], Ion Cristescu [b], Ciprian Bucur [a], Alina Niculescu [a], Iuliana Stefan [a], Marius Zamfirache [a], Stefan Welte [b]

[a] *National R&D Institute for Cryogenics and Isotopic Technologies-ICSI, Ramnicu Valcea, Romania*
[b] *Karlsruhe Institute for Technology, Institute for Technical Physics, Tritium Laboratory Karlsruhe, Eggenstein - Leopoldshafen, Germany*

Over the years, a number of researches have been carried out in order to develop the CECE or LPCE separation processes, in support of the developments concerning the water detritiation technologies for the fusion reactors or water detritiation facilities such as for the CANDU Nuclear Power Plants.

To achieve a high performance of the mixtures catalyst-packing, consisting of a hydrophobic Pt/C/PTFE type catalyst and a structured hydrophilic packing that are currently used to enhance the isotopic exchange process, various arrangements have been investigated.

The paper presents a comparative analysis for various ratio catalyst to packing and catalyst distribution within the mixture catalyst-packing as far as separation performances and column operability are concerned. During the experimental evaluation of the catalyst-packing performances the technological parameters such as pressure drop along the column and the liquid distribution over the mixtures have been thoroughly addressed. Most of the experiments have been carried out at the Tritium Laboratory from the Karlsruhe Institute of Technology and glass LPCE columns for flow visualization have been used, when possible. The functional analysis showed that the ratio catalyst to hydrophilic packing and the catalyst distribution have significant influence on the column hydrodynamic, including the pressure drop, and consequently the entire separation performances. Preferential flows along the LPCE columns in close association with the ratio catalyst/packing and catalyst distribution within the packing has been observed.

Keywords: *catalytic package, tritium, LPCE, detritiation*

Corresponding author: *anisia.bornea@icsi.ro*

Estimation of Tritium Transport Parameters in an Advanced Reduced Activation Alloy

P3-008

Woo Jun Byeon [a], Sang-Woon Yoon [b], H. J. Seo [a], H. S. Kim [a], and S. J. Noh [a, b*]

[a] *Department of Applied Physics, Dankook University, Yongin-si, Korea*
[b] *Department of Physics, Dankook University, Cheonan-si, Korea*

The transport parameters (permeability, diffusivity, solubility, trap site density, and trapping energy) of tritium in an advanced reduced activation alloy (ARAA), a reduced activation ferritic/martensitic (RAFM) steel under development at the Korea Atomic Energy Research Institute, were calculated from the quantum mechanical model based on a harmonic approximation using the measured values for hydrogen and deuterium in the temperature range of 250–600 °C. Appreciable trapping effects were observed at low temperatures (250–350 °C). The isotope effect ratios for the transport parameters were also estimated. Furthermore, our results for ARAA were compared with those for other RAFM steels previously reported by other authors.

This work was supported by the National Research Foundation of Korea (Project No. 2018R1A2B6002797)

Keywords: *Tritium transport, ARAA, RAFM, Quantum harmonic approximation*

Corresponding author: *sjnoh@dankook.ac.kr*

Study on Deuterium Permeation through Pure Iron after Exposed to LiPb

P3-009

Zhi Cao [a,b], Yongtao An [b*], Li Guo [b], Chang' an Chen [b]

[a] *Key Laboratory of Radiation physics and Technology, Ministry of Education, Institute of Nuclear Science and technology, Sichuan University, Chengdu, China*
[b] *Institute of Materials, China Academy of Engineering Physics, jiangyou, China*

Permeation Against Vacuum (PAV) has been identified as a promising technique for tritium extraction from eutectic lithium-lead in fusion power plants based on Dual Coolant Lithium-Lead (DCLL) breeding blankets[1]. The candidate permeable membrane materials were pure iron (Fe) because they have high hydrogen diffusivity and good chemical compatibility with LiPb[2-5].

The deuterium permeation through Fe membranes which exposed to LiPb at 823K for 200h has been studied on temperature dependence between 573 and 973K. The microstructure, mechanical properties and

deuterium permeation properties of the specimens were investigated. LiPb attacking could significantly weaken the deuterium permeation property of the Fe. With the increasing of temperature, the specimens enlarged the deuterium permeation rate and display the largest permeation rates while at 973K. After exposed to 823K in LiPb for 200h, it was found grain boundary sizes of the specimens enlarged after liquid LiPb corrosion and porous structure inhomogeneous distributed on the surface. It can be explained that Deuterium permeation and Lithium attack both were thermal diffusion process [6-7], as low-Z elements, the atom of Li permeated into the substrate and block up the path where the deuterium atom would pass through. In the meanwhile, the attack of LiPb would enlarge the grain boundaries and crystal defect [6-7], which formed the hydrogen trap, restrained the deuterium permeation property. Further study on the effect of deuterium permeation behavior as a function of LiPb corrosion temperature would be investigated.

[1] I. Ricapito, G. Benamati, Technologies of tritium extraction from Pb–17Li: review and comparison, ENEA Report LB- A-R-017, February 2003.

[2] E. Serra, G. Benamati, O.V. Ogorodnikova, J. Nucl. Mater. 255 (1998) 105.

[3] H. Feuerstein, J. Nucl. Mater. 233–237 (1996) 1383.

[4] H. Feuerstein, Compatibility of 31 Metals, Alloys and Coatings with Static Pb17Li Eutectic Mixture, Forschungs-zentrum Karlsruhe GmbH, Report FZKA 5596, 1995.

[5] G. Benamati, C. Fazio, I. Ricapito, J. Nucl. Mater. 307–311 (2002) 1391.

[6] Mieluch, J.; Smialowski, M. Corros. Sci. 1964, 4, 237–243.

[7] Joshi, A.; Stein, D. F. Corrosion 1972, 28(9), 321–330.

Keyword: *Permeation Against Vacuum; Iron; LiPb; deuterium permeation*

***Corresponding author**: anyt03@163.com, 11880333@qq.com*

Tritium Inventory Analysis for Compact Volumetric Neutron Source (CVNS) by Using Tritium Analysis Program for Fusion System (TAS)

P3-010

Dehong Chen [a], Xinyu Sun [a], Qiusun Zeng [a], **Zhibin Chen** [a]*, Jieqiong Jiang [a], Jie Yu [a], Yican Wu [a], Alexander Ivanov [b], Peter Bagryansky [b], Igor Kotelnikov [b], Vadim Prikhodko [b], Dmitry Yakovlev [b], FDS Team

[a] *Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Hefei, Anhui, China*

[b] *Budker Institute of Nuclear Physics, Siberian Branch, Russia Academy of Sciences, Russia*

Compact Volumetric Neutron Source (CVNS) based on Gas Dynamic Trap (GDT) was jointly proposed by Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences and Budker Institute of Nuclear Physics (BINP), Russia Academy of Sciences. It has the fusion power up to 3MW, 10^{18}n/s of neutron yield and 2MW/m^2 of neutron flux. In this case, tritium consumption will be no more than 200g/y so that tritium breeding and tritium self-sufficiency will not be the prerequisite for continuous operation of CVNS. While reducing the tritium inventory in the tritium cycle system of CVNS is an important topic for cost saving and nuclear safety of CVNS.

In this work, two optional tritium cycle modes were proposed, one is that the neutral beams inject the mixture of tritium and deuterium so that the tritium and deuterium do not need to be separated by isotope separation system (ISS) for saving tritium cycling time, but the tritium throughput will be larger due to fueling system for sustaining the background plasma also need fuel mixture of tritium and deuterium to keep the tritium and deuterium mixture ratio. Another mode is based on the ITER tritium cycle mode, the tritium and deuterium will be injected separately by neutral beams while the fueling system only inject the deuterium for reducing the tritium throughput but the ISS will be needed and the tritium cycling time will be longer. Based on the Tritium Analysis Program for Fusion System (TAS) developed by FDS Team, the tritium cycle simulation model was updated with the two optional tritium cycle modes for the tritium inventory analysis of CVNS. An optimized tritium cycle scheme was proposed for minimized the tritium inventory of CVNS.

Keywords: *Gas dynamic trap, fusion volumetric neutron source, tritium inventory, tritium cycle*

***Corresponding author:** zhibin.chen@fds.org.cn*

Evaluation of Thermal Profile in Catalytic Reactor by Exothermic Hydrocarbon Feed into Detritiation System

P3-011

Yuki Edao *, Yasunori Iwai

National Institutes for Quantum and Radiological Science and Technology, Tokai, Ibaraki, Japan

Detritiation system (DS) for fusion reactor should be designed to ensure the removal of tritium is even when an extraordinary event such as fire occurs. In the event of fire, the generation of high concentrations of hydrocarbons, mainly ethylene, is assumed to be caused by combustion of polymer materials such as cables. When the hydrocarbons are fed to a catalytic reactor elevated temperature for tritium oxidation of DS, it is concerned that the excessive rise in temperature of the catalytic reactor due to the heat of combustion becomes a risk. In this study, in order to evaluate the excessive rise in temperature of the catalytic reactor, the rate constant of the reaction of hydrocarbons combustion and thermophysical property of catalyst is

evaluated experimentally, and thermal profile in a catalytic reactor is evaluated by a numerical analysis using the thermophysical properties.

The numerical analysis indicated that the temperature of a catalytic reactor rose to 900 °C due to heat of combustion of ethylene when air containing a hydrogen concentration of 1% and an ethylene concentration of 1% was processed in the reactor packed with a hydrophobic platinum catalyst at 200°C. At 900°C, there is a concern about the influence on the soundness of the wall material of a reactor, the influence on the equipment on the downstream side of the catalytic reactor by the high temperature gas and the increase in tritium permeation. Hence, it is necessary to take measures such as a mechanism to remove heat for the design of a catalytic reactor.

Keywords: *catalytic reactor, thermal profile, oxidation, tritium, hydrogen, hydrocarbons, detritiation system*

**Corresponding author:* *edao.yuki@qst.go.jp*

Testing of Ceramic Porous Membranes for Separation of Plasma Enhancement Gases

P3-012

Luca Farina *, Marco Incelli, Mirko Sansovini, Alessia Santucci, Silvano Tosti

ENEA, FSN Department, Frascati, Italy

Plasma Enhancement Gas (PEG), namely inert gases (nitrogen, neon, argon, etc.), are used in order to mitigate the heat load over plasma facing components. In particular, PEG separation is a part of the Plasma Exhaust Processing System of DEMO.

Previous studies have been dedicated to study the PEG separation via ceramic porous membranes by modelling the mass transfer mechanisms and performing preliminary tests with pure gases.

This manuscript reports the results of gas permeability tests carried out with binary mixtures of Ar, N_2, He, H_2 and D_2. The gas mass transfer model is validated through the comparison with the experimental results expressed in terms of gas permeance and selectivity.

In particular, test with D_2 are aimed at investigating the isotopic effect of gas transport through porous membranes and finally assessing the permeance of DT.

According to the process requirements of the PEG separation of DEMO, design and performance evaluation of a ceramic membrane cascade is also presented.

Keywords: *plasma enhancement gas, membrane separation, gas permeation, plasma exhaust processing.*

**Corresponding author:* *luca.farina@enea.it*

Work in Progress on a Novel Approach for Core Fuelling of DEMO by Injection of High-Speed Pellets from the High-Field Side

P3-013

Antonio Frattolillo [a]*, Larry R. Baylor [b], Francesca Bombarda [a], Fabio Cismondi [c], Andrea Colangeli [a], Christian Day [d], Gerardo D'Elia [a], Trey E. Gebhart [b], Francesco Iannone [a], Peter T. Lang [e], Steven J. Meitner [b], Silvio Migliori [a], Fabio Moro [a], Bernard Pégourié [f], Bernhard Ploeckl [e], Fabrizio Poggi [a]

[a] *ENEA C.R. Frascati, Rome, Italy*
[b] *Oak Ridge National Laboratory, Oak Ridge (TN), U.S.A.*
[c] *EUROfusion Consortium, Programme Management Unit, Garching, Germany*
[d] *Karlsruhe Institute of Technology, Karlsruhe, Germany*
[e] *Max Planck Institute for Plasma Physics, Garching, Germany*
[f] *CEA-IRFM, Saint-Paul-lez-Durance, France*

Extensive investigations, within the EUROFusion Work Package "Tritium, Fuelling and Vacuum", indicate that sufficiently deep fuel deposition inside H-mode plasmas of the EU-DEMO tokamak requires injection of fuel pellets from the High Field Side (HFS) at speeds $\gtrsim$ 1 km/s. To implement this, two different approaches are being pursued: one makes use of guide tubes featuring large bend radii ($\gtrsim$ 6 m), to transport 1 km/s pellets to the HFS while preserving their mass and integrity; the other explores the feasibility of injecting high-speed ($\gtrsim$ 2 km/s) pellets from the HFS, along either vertical or oblique "Direct-Line-of-Sight" (DLS) paths. This paper focuses on the latter approach. Recent tests with an existing ENEA-ORNL high-speed injector have confirmed that the trajectories of free-flight pellets, travelling under vacuum at speeds up to 2.4 km/s, spread within an angle $\lesssim$ 0.7°. Despite their small scatter cone, free-flight pellets may require too much cut off volume of the Breeding Blanket (BB), due to the large distance between the injector and the plasma. The introduction of a straight guiding tube could transport high-speed pellets to the plasma without significant loss of BB material. Changes to the the existing injector, to test such a straight guiding tube, are examined. Recently, the EU-DEMO configuration has changed from 18 back to 16 TF coils. A CAD investigation has been launched to assess whether this change will open new options for DLS injection to the HFS. The neutron flux across a DLS aperture is being currently addressed; preliminary results are shortly reported (details are given in a dedicated paper in this conference).

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *EU-DEMO tokamak, High Field Side high-speed pellet injection, straight guide tubes France*

**Corresponding author : antonio.frattolillo@enea.it*

Progress of the R&D Programme to Develop a Metal Foil Pump for DEMO

P3-014

Stefan Hanke [a]*, Christian Day [a], Thomas Giegerich [a], Juri Igitkhanov [a], Yannick Kathage [a], Xueli Luo [a], Stylianos Varoutis [a], Alejandro Vazquez Cortes [a], Thomas Härtl [b], Andrei Busniuk [c], Alexander Livshits [c], Stefan Merli [d], Andreas Schulz [d], Matthias Walker [d], Klaus Baumgärtner [e], Jens Hofmann [e], Monika Balk [e]

[a] *Karlsruhe Institute of Technology (KIT), Institute for Technical Physics (ITEP), Karlsruhe, Germany*
[b] *EUROfusion Consortium, PPPT Department, Garching, Germany*
[c] *Bonch-Bruevich Saint-Petersburg State University of Telecommunications, St. Petersburg, Russia*
[d] *Institute of Interfacial Process Engineering and Plasma Technology, Stuttgart, Germany*
[e] *Mügge GmbH, Reichelsheim, Germany*

DEMO requires a novel fuel cycle architecture in order to drastically minimize the tritium inventory in the fuel cycle. This is achieved by Direct Internal Recycling, a concept which introduces a short-cut to the classical fuel cycle architecture. The central element of it is the separation (and compression) of hydrogenic gas from the exhaust gas stream, close to the divertor, which can immediately be recycled to feed the matter injection systems. Due to the fact that the separated gas will be pure hydrogen, it does not have to be routed through the tritium plant.

The separation shall be achieved by a novel pump type which relies on the effect of superpermeation. To achieve a convincing performance, each hydrogen particle has to undergo successfully a series of processes before it is pumped and therefore separated. Firstly, suprathermal hydrogen particles have to be generated at large amounts, which is done in a collisional plasma. After transport to the metal foil membrane they have to be absorbed and then be transported through the bulk. Finally, they have to leave the foil membrane at the rear side -rather than being reflected back into the bulk- and will recombine to neutral diatomic hydrogen. R&D is currently being focused on (i) the plasma source, (ii) the transport of the suprathermal hydrogen, and (iii) the material and surface aspects of the foil. Examples of recent results will be presented, a first quantitative prediction on the performance will be derived, and a first design suggestion of a technical scale pump will be presented.

Keywords: *Metal Foil Pump, Direct Internal Recycling, Superpermeation, Hydrogen Separation, Fuel Cycle*

**Corresponding author: stefan.hanke@kit.edu*

Hydrogen Storage System for Hydrogen Isotope Separation Using Gas Chromatograph

P3-015

Norihiro Ikemoto [a, b]*, Tomohiko Kawakami [a], Kazuo Yonehara [a], Yuri Natori [a], Katsuyoshi Tatenuma [a], Masanori Hara [b]

[a] *Kaken Inc., Mito-shi, Ibaraki, Japan*
[b] *Univesity of Toyama, Toyama-shi, Toyama, Japan*

In the fuel cycle of nuclear fusion reactor, the hydrogen isotope separation is a key process. The gas chromatograph (GC) is a candidate method for the hydrogen isotope separation. Since GC system should be operated under a batchwise mode, a hydrogen feeding system into GC is indispensable. The feeding system need to store a mixture gas of hydrogen isotopes that is already purified, and the mixture stored is supplied to a gas feeding volume in GC at a given pressure.

Molecular sieves (MS) readily adsorb hydrogen isotopes at 77 K (liquid nitrogen temperature). Before using MS as an adsorbent of hydrogen isotopes, MS should be heated in vacuum to remove water molecules on the surface of MS.

On this study, the interference of residual water molecules on MS surface was considered. MS was loaded into a stainless steel vessel up to a given amount. MS was heated at various temperature from 473 K to 673 K under 1000 Pa. After the heat treatment, hydrogen adsorption isotherms of MS at 77 K were measured by a volumetric method. Isotherms were changed with increasing the temperature of heat treatment. At low hydrogen pressure in isotherms, the shape of isotherm was unchanged. However, the amount of adsorbed hydrogen in high pressure region increased with increasing the temperature of heat treatment. Because the residual adsorbed water on MS surface decreases with increasing the temperature. Therefore, the temperature of heat treatment of MS is required more than 673 K. At conference, deuterium isotherms will be also shown.

Keywords: *fuel purification system, molecular sieve, hydrogen isotope separation, gas chromatograph, cryogenic adsorption, adsorption isotherm*

**Corresponding author:* *n.ikemoto@kakenlabo.co.jp*

Experimental Results of a Medium-Scale Pd-Ag Permeator for the Tritium Extraction and Removal System

P3-016

Marco Incelli*, Alessia Santucci, Silvano Tosti

ENEA Department of Fusion and Technology, Frascati, RM, Italy

A systematic approach has been developed in order to identify the most suitable technologies for tritium extraction in the blanket. This assessment has demonstrated the maturity of the cryogenic technologies in tritium removal from He purge gas. Hence, they have been selected as the baseline technology for the TER system, while a back-up solution, consisting in a combination of zeolite and Pd-Ag membrane technologies, is still at R&D level although it could potentially perform continuous operation, reduce the tritium inventory and the plant layout complexity.

For these reasons, in ENEA laboratories, R&D activities are currently focused on the process and design engineering of the Pd-Ag membrane technologies. A 1-D simulation code has been already developed in collaboration with KIT and successfully validated with experiments in order to investigate the Q_2 permeation efficiency in different operating condition and different geometrical parameters. Moreover, a single-tube permeator has been deeply tested and a multi-tube permeator consisting of 10 Pd-Ag membrane tubes has been designed and commissioned in order to test Pd-Ag membrane technologies at medium scale.

In order to fulfil the requirements for the TER system, in this work, the experimental results of the hydrogen separation performances of the multi-tube permeator will be presented. In particular, the permeation efficiency will be evaluated in different He/H_2 feed flow ratios, pressures (1 to 5 bar), temperatures (300 °C to 450 °C) and the flow-rate influence will be assessed, moving closer to DEMOrelevant conditions.

Keywords: *tritium extraction and removal system, Pd-Ag membranes, DEMO HCPB blanket*

**Corresponding author:* *marco.incelli@enea.it*

Development of Low Specific Activity Analyzer with Proportional Counter

P3-017

Xiaomei Ji*, Zhizhen Liu, Xingchen Guo, Peixu Zhang, Weiying Li, Qin Zhan

China Institute of Atomic Energy, Beijing, China

Based on the self-designed low specific activity analyzer with proportional counter, an on-line system for specific activity measurement of tritium was established. The low specific activity analyzer with proportional counter is composed of proportional counter, preamplifier, tritium activity measurement analyzer and computer application analysis software. Through this system, the plateau curve and background counting of proportional counter are measured, and the methods of dead time correction, threshold correction, end effect correction and wall effect correction are studied. The above methods reduce the minimum detection limit of the measurement system and improve the detection efficiency. The P10 gas and low pressure gas sampling method are used to calibrate the low specific activity analyzer with proportional specific. The measurement range of the tritium specific activity of the system is 370Bq/L-3.7E+5Bq/L.

Keywords: *low specific activity analyzer, tritium, on-line activity measurement, correction method*

**Corresponding author:* *jixiaomei_2000@163.com*

A study on the Recovery Process of Hydrogen Isotopes

P3-018

Woo-Chan Jung [a]*, Sang-Woo Han [a], Young-Min Kim [a], Hung Man Moon [a], Min-Ho Chang [b], Hyeon-Gon Lee [b]

[a] *Daesung Cryogenic Research Institute, Daesung Industrial Gases, Ansan, Korea*

[b] *ITER Korea, National Fusion Research Institute, Daejeon, Korea*

There are various gas components in the exhaust gas of the D-T fusion reaction. All of the hydrogen isotopes are recovered and reused as fuel, and the remaining components are released to the environment. It is not good situation if hydrogen isotopes are released without being recovered. Therefore, the process must be designed and managed so that hydrogen isotopes are controlled below the allowable concentration.

In this study, experiments using hydrogen and deuterium were conducted to confirm the possibility of the recovery of hydrogen isotopes in the fusion exhaust gas. In the feed gas containing H_2 and D_2, impurities were removed in the Pd membrane process, and only pure H_2 and D_2 were recovered. Remaining gas including trace amounts of H_2 and D_2 were all converted to H_2O through the catalytic oxidation reaction. And the converted H_2O was recovered by adsorption process.

Keywords: *hydrogen isotopes, recovery process, Pd membrane, oxidation, adsorption*

**Corresponding author:* *chan@gastopia.co.kr*

Hydrogen Removal Efficiency Using Storage-Bed Circulation Process for Helium-3 Collection System

P3-019

Pil Kap Jung *, Min Ho Chang, Hyun-Goo Kang, Dong-You Chung , Jea-Uk Lee

National Fusion Research Institute, Yuseong-gu, Daejeon, Korea

When tritium is stored in the storage bed for a long, helium-3 produced by beta decay is concentrated. It should be collected periodically as an obstacle to the tritium storage and supply. The circulation process using the storage bed can be applied as the helium-3 collection system. First, the storage bed is heated to desorb helium-3 and tritium. Then, when the desorbed mixed gas is supplied to the storage bed of circulation process at room temperature, tritium is adsorbed and helium-3 is enriched.

This study was conducted to investigate the amount of residual tritium in helium-3 concentrated in the storage bed circulation process. The hydrogen and helium-4 mixed gas was supplied into the circulation process consisting of the getter bed, the pump and the buffer tank, and the hydrogen concentration in the residual gas was measured after sufficient circulation. In addition, the hydrogen concentration was observed by varying bed temperature, initial hydrogen concentration and fraction to understand the influence factors on the residual amount of hydrogen. The experimental results can be used to predict the tritium residual amount in the helium-3 collection process, and will be useful in determining the need for subsequent treatment processes and in selecting suitable processes.

Keywords: *tritium, storage bed, helium-3 collection, circulation system*

**Corresponding author:* *jpk75@nfri.re.kr*

Influence of Lithium Mass Transfer on Tritium Behavior in Pebbles of Li_2TiO_3 with Excess Lithium

P3-020

Kazunari Katayama [a]*, Akito Ipponsugi [a] , Tsuyoshi Hoshino [b]

[a] *Department of Advanced Energy Science, Kyushyu University, Kasuga, Fukuoka, Japan*

[b] *Department of Blanket Systems Research, National Institutes for Quantum and Radiological Science and Technology (QST), Rokkasho-mura, Kamikita-gun, Aomori, Japan*

It is important to understand tritium behavior in a tritium breeding blanket for safety control of tritium and establishment of optimized fuel cycle of a fusion DEMO reactor. Because Li ceramic pebbles for tritium breeding are used in high temperature conditions for a long time in the blanket, a certain amount of Li or Li compounds will be evaporated and transported from the pebbles. One concern is the change of tritium behavior due to Li mass transfer during operation of the reactor. However, study on tritium behavior in Li ceramic pebbles heated for a long period is insufficient. In this work, influence of Li mass transfer on tritium sorption behavior was investigated.

The pebbles of Li_2TiO_3 with excess Li, which was developed by National Institutes for Quantum and Radiological Science and Technology as an advanced tritium breeding material, were heated in 1000 Pa H_2/Ar flow at 900 °C for 30 days. Due to this heating, the Li mass loss of 0.7 wt% occurred and specific surface area decreased from 0.7 to 0.2 m^2/g. From SEM observation, averaged grain size was increased from 1.8 μm to 3.1 μm. Tritiated water vapor was introduced to the packed bed of long time heated and unheated pebbles at room temperature, 300 °C, 600 °C and 900 °C and tritium sorption capacity obtained from tritium recovery process was compared. The sorption capacity at room temperature and 300 °C on the heated pebbles was two orders of magnitude smaller than that of the unheated sample. This indicates that by preheating the packed bed up to 300 °C the surface tritium inventory of the pebbles can be significantly reduced. The sorption capacity at 600 °C and 900 °C on both heated and unheated pebbles was almost the same at 3.6×10^{-5} mol-T_2/g. The pebbles have tritium sorption capacity that is not lost even at long heating time at 900 °C. 0.7 wt% Li mass loss and 1.7 times grain growth did not affect the sorption capacity.

Keywords: *Li_2TiO_3, Tritium, Li evaporation, grain growth*

**Corresponding author:* *kadzu@nucl.kyushu-u.ac.jp*

Dynamic Modeling and Simulation of Pellet Injection System in ITER

P3-021

Min-Kyung Lee [a], Jae-Uk Lee [b], Min Ho Chang [b], Jin-Kuk Ha [c], In-Beum Lee [a], and Euy Soo Lee [c,*]

[a] *Department of Chemical Engineering, Pohang University of Science and Technology, Pohang, Korea*
[b] *National Fusion Research Institute, Yusong-gu, Daejon, Korea*
[c] *Department of Chemical & Biochemical Engineering, Dongguk University, Seoul, Korea*

This paper proposes a dynamic model for simulation of pellet injection system (PIS) in fusion power plant. PIS, consisting of several devices such as valves, vessels, pipelines, extruders and heat exchangers, is a part of fueling system between storage delivery system (SDS) and Tokamak. From a safety perspective, the PIS design is dictated by a limit on the tritium inventory to 70 g within a PIS cask. Dynamic simulation for investigating transient behavior of tritium inventory in PIS is crucial, with respect to the safe design and reliable operation. The proposed model provides an overall insight into time-varying tritium inventory under a wide range of processing parameters while satisfying Tokamak fuel demand. The dynamic simulation model allows to assess in real-time the impact of changing process conditions on system behavior. Sensitivity analysis to determine optimal processing parameters for operation with minimum tritium inventory was conducted.

Keywords: *Dynamic simulation, Pellet injection system, Tritium inventory, Tokamak fuel demand*

**Corresponding author:* *eslee@dongguk.edu*

Simulation and Analysis of Fuel Storage System in Fusion Fuel Cycle Considering Offnormal Heat Loads

P3-022

Jae-Uk Lee *, Min Ho Chang, Hyun-goo Kang, Dongyou Chung

National Fusion Research Institute, Yusong-gu, Daejon, Korea

Off-normal heat load scenarios of fuel storage system in fusion fuel cycle have been studied by process simulations. Fuel storage system consists of vessels and hydride beds as main fuel storage units. Fuels exist in gas phase in both units to be transferred to other systems. One of main fuels, tritium, is strongly prohibited to be released to environment due to its radioactivity and explosiveness. In this study, dynamic model has been developed to analyze effects of off-normal heat loads on vessels and hydride beds to overall system. External fire on vessels and control fail of heater in beds during normal operation of the Tokamak are considered base off-normal scenarios. The model can estimate transient pressure profile through entire system and provide required size of relief system

Keywords: *fusion fuel cycle, fuel storage system, tritium release, dynamic simulation, off-normal events*

**Corresponding author:* *jaeuklee@nfri.re.kr*

Process Design of the Water Detritiation System for China Fusion Engineering Test Reactor (CFETR)

P3-023

Peilong Li, Wenhua Luo, Jinguang Cai, Zhi Zhang, Yan Shi, Jiangfeng Song, Xiaojing Qian, Deli Luo

Institute of Materials, China Academy of Engineering Physics, Mianyang, China

According to the characteristics of operation, maintenance and emergency condition of China Fusion Engineering Test Reactor (CFETR), specific design requirements for Water Detritiation System (WDS) are systematically analyzed. The overall process and each subsystems are detailedly introduced in this present work. The process design in which mature and well-demonstrated technologies are adopted conforms to the latest input requirements of CFETR, and is also optimized based on the latest research and practice experiences. Water Distillation (WD) process and Combined Electrolysis Catalytic Exchange (CECE) are designed for tritiumed water with different tritium concentration. Higher operation efficiency and reduced overall energy consumption can be achieved in this WDS process.

Keywords: *China Fusion Engineering Test Reactor (CFETR); Water Detritiation System (WDS); technical process; design*

**Corresponding author:* *lipeilong2018@foxmail.com*

D/T Separation from He with Superpermeable Membranes: FCC V-Alloys as New Membrane Materials

P3-024

Vasilii Alimov [a], Andrei Busniuk [b], **Christian Day** [b], Stefan Hanke [b], Evgenii Peredistov [a], Alexander Livshits [a]*

[a] *Bonch-Bruevich Saint-Petersburg State University of Telecommunications, St. Petersburg, Russia*
[b] *Karlsruhe Institute of Technology (KIT), Institute for Technical Physics (ITEP), Karlsruhe, Germany*

Direct Internal Recycling concept has been proposed to reduce the tritium inventory of DEMO, the demonstration nuclear fusion power plant. In the exhaust gas stream of the machine the major portion of D/T mixture is supposed to be separated from other gases in the immediate vicinity of the divertor output, compressed there and directed back into the plasma by the shortest way without passing the tritium plant. The superpermeable membranes (SPM), which can effectively pump suprathermal hydrogenic particles, can be applied for the implementation of such separation function. The combination of SPM with the plasma generation of suprathermal hydrogen appears to be the most suitable technical solution. Vanadium was a candidate membrane material suggested previously. To reduce the tritium inventory in the membrane material it seems worthwhile to investigate the option to use a vanadium based alloy with reduced solubility of hydrogen. In this case the tritium inventory is determined not only by the membrane temperature but also by the concentration of the alloying element that can be varied providing an additional degree of freedom for development of a self-consistent plasma-SPM system. A systematic study of hydrogen solubility in- and hydrogen transport through FCC V-Pd alloys was carried out in the range of Pd concentration from 0 to 19 at%. Alloying V with Pd appeared to be a rather effective means to reduce the hydrogen solubility. This indicates a promising way to controllably reduce the tritium inventory in the membranes by more than an order of magnitude.

Keywords: *Metal Foil Pump, Direct Internal Recycling, Superpermeation, Hydrogen Separation, Fuel Cycle*

****Corresponding author:*** *livshits@sut.ru*

Effect of Ni and Ti Doping on Stability and Hydrogen Absorption Properties of Zr_2Fe Alloy

P3-025

Jiangfeng Song [a,b]*, Xushan Zhao [c], Xianggang Kong [d], Jingchuan Wang [a], Zhenghe Zhu [b], Daqiao Meng [a,b]

[a] *Institute of Materials, China Academy of Engineering Physics, China*
[b] *Institute of Atomic and Molecular Physics, Sichuan University, China*
[c] *Computer Network Information Center, Chinese Academy of Sciences, China*
[d] *Chengdu University of Information Technology, China*

Based on MatCloud High-throughput Material Integrated Computing Platform (matcloud.cnic.cn), an automated process algorithm was established to solve the disproportionation reaction of Zr_2Fe. The structural stability of modified Zr_2Fe alloys with different doping elements and doping concentrations are systematically studied. The results indicate that Ni doping is beneficial to the enhancement of hydrogen absorption performance and the maximum doping concentration of Ni-doped Zr_2Fe is 33.at%. While Ti doping Zr_2Fe will form new phase and is beneficial to the diffusion of hydrogen atoms, the overall hydrogen absorption capacity decreases with the deceasing of the phase content of Zr_2Fe in the main phase.

Keywords: *Nuclear materials; Metals and alloys; doping; Simulation; hydrogen absorption*

****Corresponding author:*** *iterchina@163.com*

Rapid Purification of Hydrogen Isotopes by Helical Tubular Pd-8%Y Alloy Membrane

P3-026

Xiong Yifu, Shi Yan, Jing Wenyong, He Mingmin, Ba Jingwen, Song Jiangfeng *

Institute of Materials, China Academy of Engineering Physics, China

The palladium-yttrium (Pd-Y) helical tube was developed for the fast purification of hydrogen-isotope gas. The result indicated that under different temperatures and pressures, the overall leakage rate was down to less than $1.5\times10^{-9}Pa.m^3.s^{-1}$, the recovery rates for low-content hydrogen isotopes were up to more than 99%, and the daily processing capacity was approximately a tenfold increase to 20ml compared to the conventional straight tube. The fundamental solution was achieved on

the rapid removal of tiny amount of ^3He gas in a large batch of hydrogen-isotope gas thus the significant increase was also acquired on the purification capacity for hydrogen isotopes.

Keywords: *Palladium-yttrium helical tube; Hydrogen isotopes; purification*

**Corresponding author:* *iterchina@163.com*

Density-Functional Calculations of Hydrogen Diffusion in Rutile TiO_2 (110)

P3-027

Wei Mao [a]*, Wilde Markus [b], Shohei Ogura [b], Takumi Chikada [c], Katsuyuki Fukutani [b], Hiroyuki Matsuzaki [d,e], Takayuki Terai [a,e]

[a] *Institute of Engineering Innovation, School of Engineering, The University of Tokyo, Bunkyo-ku, Tokyo, Japan*
[b] *Institute of Industrial Science, The University of Tokyo, Meguro-ku, Tokyo, Japan*
[c] *Department of Chemistry, Graduate School of Science, Shizuoka University, Suruga-ku, Shizuoka, Japan*
[d] *The University Museum, The University of Tokyo, Bunkyo-ku, Tokyo, Japan*
[e] *Department of Nuclear Engineering and Management, School of Engineering, The University of Tokyo, Bunkyo-ku, Tokyo, Japan*

TiO_2 is regarded as one of candidate materials for tritium permeation barriers (TPB) because of its low diffusivity, its high porosity that allows maximizing the hydrogen flux, and its good inert properties towards both the metals and the gas mixtures. At present, rutile TiO_2 (110) has become the most studied oxide surface, and is therefore used to study H diffusion behaviors.

Density functional theory (DFT) was used to calculate the energtics of hydrogen diffusion on and from the surface, and in the bulk of TiO_2 (110). All calculations were done using GGA-PBE functional and Vienna Ab initio Simulation Package (VASP) code. A k-points of 2×2×1 is used for the slab of 2×4×5 (80 Ti atoms and 40 O atoms), and the vacuum region is set to 14 Å.

The diffusion measurement shows that H diffuses from the surface to the subsurface and into the bulk of TiO_2 (110) was elucidated using DFT method. The diffusion coefficient is $D = (7.57\times10^{-7} cm^2/s) \exp(-1.09\ eV/RT)$, and H diffusion is in the direction of a axis. Our results provide a fundamental understanding of H diffusion in bulk TiO_2 for TPB coatings.

Keywords: *TiO_2(110), density functional theory, diffusion coefficient, tritium permeation barriers*

**Corresponding author:* *mao@nuclear.jp*

Fuel Flow and Stock during Deuterium-Deuterium Start-up of Fusion Reactor

P3-028

Kento Miyamae [a]*, H. Yamada [b,a], R. Kasada [c], S.Konishi [d], R. Sakamoto [b,e]

[a] *Graduate School of Frontier Sciences, The University of Tokyo, Kashiwa, Japan*
[b] *National Institute for Fusion Science/NINS, Toki, Japan*
[c] *Institute for Material Research, Tohoku University, Sendai, Japan*
[d] *Institute of Advanced Energy, Kyoto University, Uji, Japan*
[e] *SOKENDAI, Toki, Japan*

Flow and stock of fusion fuels in a fusion reactor has been investigated from the aspect of systemdynamics. In particular, operational scenario of the start-up only by deuterium (D) has been discussed including effects of dilution due to helium (He) ash and fuel recycling in a plasma vacuum vessel. Initial loading of tritium (T) in a fusion reactor is a critical issue because of availability of T. Thus, the start-up only from D has been attracting interests. The present model of burning plasma has been much improved compared with the previous research which showed technical feasibility of this D-D startup scenario [1]. The temporal evolution of plasma temperature has been solved in an integrated manner of temperature and density profiles, an empirical scaling of energy confinement time, isotope effect of confinement, slowing down and velocity distribution function of energetic particles, dilution due to He ash, radiation losses, and recycling of fuels. Operational parameters are based upon the recent tokamak fusion DEMO design by the Joint Special Design Team in Japan [2] with nominal fusion power of 1.4GW. In the case without He dilution and recycling, it takes 105 days to reach the DT fuel equilibrium in a burning plasma and 481 days to save T storage of 5 kg which is required for initial loading of a DT fusion reactor. Here tritium breeding ratio (TBR) is assumed to be 1.1. When TBR is degraded to 1.05, T storage is only 1.6 kg after 481-days operation. Impact of fuel dilution due to He ash in this scenario is serious. When equivalent particle confinement is assumed for He and T, fusion power is degraded to 0.74GW and T storage is 2.4 kg after 481-days operation. Recycling of fuels also deteriorates the performance. Discussion based upon detailed parameter survey will be presented.

[1] *R.Kasada et al., Fusion Eng. Design 98-99 (2015) 1804.*

[2] *Y.Sakamoto et al., Fusion Eng. Design 89 (2014) 2440.*

Keywords: *Tritium inventory, System dynamics, Dilution of fuels, Particle recycling, D-D start-up*

**Corresponding author:* *miyamae.kento18@ae.k.u-tokyo.ac.jp*

New Sphere Granulation and Sintering Method of Tritium Breeding Solid-Ceramics

P3-029

Yuri Natori*, Norihiro Ikemoto, Kazuo Yonehara, Katsuyoshi Tatenuma

Kaken Inc., Hori, Mito, Ibaraki, Japan

Tritium breeding solid-ceramics loaded on the blanket system of fusion reactor needs to have a high tritium-breeding characteristics and recovery efficiency. Therefore various investigations have been performed and reported for forming spheres using lithium ceramic as initial powder. For forming sphere of lithium ceramic, several wet or dry granulation methods have been established. However, in either method, a plurality with complicated process of inclusions for adjusting aggregation, adhesion, flow, dispersion, and the like among particles are necessary, moreover it is difficult to perform reproducible homogeneous granulation. Therefore, the problem which develops the mass production technique is left.

By the reason, we developed a new granulation method for agglomerating particles by static electricity without inclusions such as binder and dispersant at all. Compared to the conventional granulation method, this method realized high spherical property, more efficient, more mass-producible. Therefore, it is also a high advantage that it can be applied to chemically active lithium compounds.

In calcination, we established a method to enable high hardness while promoting contraction and densification, and furthermore aimed at systemization of the granulation and calcination steps. A process of high-temperature heating for decomposing the inclusions and additives, e.g. binder, is not necessary, and in a shorter time of calcination, it was able to realize mass production of high-quality and quantity lithium compound spheres.

Keywords: *blanket, tritium breeding, lithium, solid-ceramics, electro static, sphere granulation, sintering*

Corresponding author: *y.natori@kakenlabo.co.jp*

Development of Specific Software for Hydrogen Isotopes Separation by Cryogenic Distillation of ICSI Pilot Plant

P3-030

Alina Elena Niculescu, Anisia Bornea, George Ana, Marius Zamfirache

Institute for Cryogenic and Isotopic Technologies, Rm. Valcea, Romania

Cryogenic distillation is a widely applied process for separating the components of a mixture of hydrogen isotopes, which may contain up to all six isotopologues. ITER ISS (Isotopic Separation System) is based on cryogenic distillation process and is one of the main systems within the ITER Fuel Cycle. Similar to ITER ISS, one of the key systems of a detritiation plant for low level tritiated light or heavy water is the cryogenic distillation. Provisions are being made in ICSI to design and build an experimental facility based on CECE process coupled to a cryogenic distillation system able to process these kinds of waters.

The aim of this work is to present the results of the design and simulation activities in support to the development of a cryogenic distillation system capable to achieve the required decontamination factor and tritium enrichment better than 90%. This is done by using a software developed "in house", based on heat and mass transfer model. Based on the input data like, flow rates, composition of the feeding gas, pressure drop, the simulation provides the distribution of all the molecular species involved, temperature profile and also the liquid and vapor enthalpies along the column.

The results may provide useful information for the ITER ISS decontamination factor during dynamic operation due to the ISS transient feed flow rate coming from the WDS (Water Detritiation System).

Keywords: *hydrogen isotopes, cryogenic distillation, design and simulation of a cryogenic distillation column*

Corresponding author: *alina.niculescu@icssi.ro*

Hydrogen Isotope Exchange in Tungsten During Baking in Hydrogen Isotope Atmosphere

P3-031

Yuji Nobuta [a]*, Yuji Hatano [b], Sun Eui Lee [c], Masato Nakayama [b]

[a] *Graduate school of Engineering, Hokkaido University, Sapporo, Japan*
[b] *Hydrogen Isotope Research Center, University of Toyama, Toyama, Japan*
[c] *Graduate School of Science and Engineering, University of Toyama, Toyama, Japan*

A removal technique for tritium in tungsten (W) needs to be established for tritium recovery in fusion devices. An isothermal heating of plasma-facing material, baking, is widely used for hydrogen removal and will be applied in ITER. In order to shorten the baking period, more efficient technique must be developed. In the present study, polycrystalline W containing tritium or deuterium was respectively exposed to deuterium or tritium gas at elevated temperature to investigate the isotope exchange behavior in W during baking process.

Samples of W were irradiated with 5 keV helium ions to create trap sites for hydrogen isotope. The irradiated samples were first exposed deuterium-diluted tritium (7.1 %) gas to load tritium into W. After non-destructive measurements of tritium using imaging plate technique and β-ray induced x-ray spectrometry, the samples were exposed to deuterium gas to extract tritium via isotope exchange process. The sample temperature and duration of each exposure were 573 K and 3 hs, respectively. The total gas pressure during exposure was 1.2 kPa. Exposures in reversed sequence (deuterium gas exposure followed by tritium gas exposure) was also performed to examine hydrogen isotope penetration via isotope exchange process.

The deuterium exposure that followed tritium exposure resulted in removal of 96 % of tritium retained in W. This extent of removal was larger than that observed after heating in vacuum. The tritium retention after tritium exposure that followed deuterium exposure (reversed exposure) was comparable with that after tritium exposure in the normal sequence, suggesting isotope exchange was almost completed under the present conditions. These observations indicate that isotope exchange during baking could enhance tritium removal in fusion devices.

Keywords: *Hydrogen isotope exchange, tritium removal, baking, tungsten, helium irradiation*

**Corresponding author: y-nobuta@eng.hokudai.ac.jp*

Tritium Permeation Modeling in DEMO WCLL Cooling System

P3-032

Leonardo Noschese [a]*, Alessia Santucci [b], Marco Incelli [b], Silvano Tosti [b], Francesco Romanelli[a]

[a] *University of Rome "Tor Vergata", Rome, Italy*
[b] *ENEA-Frascati, Frascati (RM), Italy*

Tritium management is one of the most critical aspects in DEMO Design. Knowledge of tritium permeation is a priority both in terms of safety and plant performance, as it may represent a radiological risk for the employed personnel and for the external environment.

The aim of the present contribution is the modeling of tritium permeation mechanism and the calculation of permeation flows inside the Steam Generators of DEMO WCLL (Water Cooled Lithium-Lead) concept.

Starting from the tritium flowrate permeating from the breeding blanket region, the model assesses under thermodynamic equilibrium conditions the concentrations and partial pressures of HT and HTO in primary and secondary coolant loops. Generally, high levels of pressure and temperature promote the permeation mechanism, which is mainly driven by the resulting HT partial pressure in gas phase and depends on system geometry, operative conditions and permeability of materials.

By taking in account the current DEMO WCLL system design as input data, a parametrical analysis has been carried out by varying the Permeation Reduction Factor (PRF) in the Steam Generator 1 to 1000, the hydrogen concentration in cooling water 0.4 to 40 mol/m^3 and the size of the coolant purification system.

Results show tritium permeation from primary to secondary coolant loop of the order of $10^{-3} \div 1$ g/y, with a strong dependence on the above described parameters. In order to validate qualitatively the model, results have been put in comparison with tritium permeation in CANDU reactors.

Keywords: *Tritium, Permeation, DEMO, Safety*

**Corresponding author: leonardo.noschese@outlook.com*

Commissioning of the LPCE and Purification Systems as Front-End of the Experimental Pilot Plant for D-T Separation

P3-033

Gheorghe Popescu*, **George Ana**, Liviu Stefan, Anisia Bornea, Ciprian Bucur, Ovidiu Balteanu, Iulia Stefan, Alina Niculescu

National R&D Institute for Cryogenics and Isotopic Technologies – ICSI Rm. Vâlcea, Romania

Liquid Phase Catalytic Exchange (LPCE) as front-end process and Cryogenic Distillation (CD) as back-end process are the processes

employed for heavy water detritiation used in Cernavoda CANDU-type Nuclear Power Plant (NPP) from Romania as neutron moderator and in the primary heat transfer system. The main purpose of the detritiation plant is to reduce the emissions to the environment, to reduce the personel doses and to reduce the heavy water consumption.

The experimental Pilot Plant from ICSI Rm. Valcea has the aim to research and obtain technological data and functional characteristics of specific equipment used in LPCE and CD processes in order to design the Detritiation Facility for Cernavoda NPP.

The research approach for the technology had two stages: the design and construction of the pilot plant as chemical facility for protium-deuterium isotope separation and the second stage consisting in changing all components for fully tritium compatibility and adding a state of the art control system for process automation and for safety.

This paper will give a detailed overview of the current status for the front-end of the pilot plant, LPCE and purification systems, including all process components. We will particularly highlight the main strengths of the PESTD plant, ie technological improvements and exploitation safety, as well as the underlying reasons. Additionally, the paper will present the results of the functional test of the LPCE using protium and deuterium.

Keywords: *tritium; LPCE, purification, TRF, safety.*

Corresponding author: *gheorghe.popescu@icsi.ro*

Effect of the Oxidation of Polysilane Scintillator on Its Fluorescence Properties

P3-034

Wenjing Pu [a], Shi Bo [a], Xiaodong Li [c], Weihua Wang [b], Chun Zhang [a], Wei Lu *[b]

[a] *Teaching and Research Department of Physics, Army Academy of Artillery and Air Defense, Heifei, China*
[b] *Institute of Physical Science and Information Technology, Anhui University, Heifei; College of Science, China*
[c] *National University of Defense Technology, Changsha, China*

Tritium is radioactive hydrogen which can be easily absorbed by organisms. If there is Tritium leaks into environment, it would be harmful to human health. Scintillator is the core material of tritium detector. The traditional plastic scintillator is based on polystyrene, which will produce peroxide under high temperature and irradiation, and will absorb the photons emitted by the scintillator in the process. The chemical and mechanical degradation of the plastic scintillator not only causes cracks but also damages the optical transmission properties of the material. Polysilane has unique optical properties for the σ conjugate structure, which perform both crystal scintillator and plastic scintillator advantages. So, polysilane is expected to be a highly efficient tritium-releasing scintillator material by modifying the side chains. In this paper, preliminary studies about the modified polymethylsilane (PMS) is carried out on the relationship between oxidation resistance and electron excitation spectrum/ultraviolet absorption spectra/fluorescence emission spectrum. At the same time, the mechanism of the optical performance degradation with the oxidation of modified PMS is also be discussed. Results show that mild oxidation of substituent will only cause the molecular chain into crosslinking, but won't lead to Si - Si main chain degradation and fluorescence loss, which is different from the traditional plastic scintillator. Therefore, stable and effective substituents are expected to improve PMS fluorescence properties and oxidation resistance at the same time, which is the pressing needs for the scintillator material with high temperature resistance and irradiation resistance.

Keywords: *scintillator, polysilane, tritium beta decay, oxidation, fluorescence*

**Corresponding author:* *taigold3rd@163.com*

Optimization of Tritium Breeding Ratio in ARC Reactor

P3-035

Stefano Segantin [a]*, Raffaella Testoni [a], Zachary Hartwig [b], Dennis Whyte [b], Massimo Zucchetti [a]

[a] *DENERG, Politecnico di Torino, Italy*
[b] *Plasma Science and Fusion Center, MIT, Cambridge (MA), USA*

Affordable Robust Compact reactor is a conceptual design for a Tokamak conceived by Massachusetts Institute of Technology (MIT) researchers. It represents a new generation of fusion reactors, which have the final goal of tracing a new path to a clean, fast and cheap fusion energy connectable to the power grid. The design of this tokamak is under development and update. One of the key parameters for fusion reactor power plants is the tritium breeding ratio (TBR), which has to guarantee the tritium self-sufficiency (therefore this ratio has to be at least 1).

From a safety viewpoint, the tritium inventory circulating in a fusion power plant must be minimized. In the meantime, to enhance plant's economics, the amount of tritium generated and stored for should be maximized, since it would be used to startup new reactors. Both of the aforementioned trends meet their best in a TBR as high as possible. In this work, ARC tritium breeding ratio is studied and optimized.

ARC's vacuum vessel has been designed as a double walled chamber with several layers of materials for structure, first wall and neutron multiplier. It is immersed in a liquid blanket tank and the fluid, that is FLiBe (LiF-BeF_2), also flows between the two vessel walls, in order to cool them down and breed as much tritium as possible.

Taking advantage of Monte Carlo codes, several configurations of ARC's blanket and vacuum vessel have been analyzed in order to find the

most effective one for a high TBR. The study takes into account different materials for the first wall and structure. Moreover, it scans different thicknesses of each material layer and studies the position and width of FLiBe channels, looking for the best configuration that allows the highest possible tritium breeding ratio. A conclusive analysis on the total radioactive inventory due to tritium and activated structure is carried out. Results of the study are presented in the paper.

Keywords: *ARC, Tokamak, TBR, Blanket, Monte Carlo*

Corresponding author: *stefano.segantin@polito.it*

Status Tritium Laboratory of Complex TSP JSC "SRC RF TRINITI" and Tasks of Development Tritium Fuel Cycle Under the Ignitor Project Requirements

P3-036

Mikhail Subbotin [a]*, Alexander Gostev [b], Mikhail Rozenkevich [c], Alexander Perevezentsev [c]

[a] *NRC "Kurchatov Institute", Moscow, Russia*
[b] *JST "SRC RF TRINITI" (State Corporation "Rosatom"), Moscow, Troitsk, Russia*
[c] *D. Mendeleev University of Chemical Technology of Russia, Moscow, Russia*

The project of tokamak Ignitor is one of the main themes of long-term scientific cooperation between the Russian Federation and the Italian Republic. Currently, negotiations on the development of technical design tokamak Ignitor with placement on the site of complex "Tokamak with Strong Field" (TSP) JSC "SRC RF TRINITI" (complex TSP TRINITI), Moscow, Troitsk, Russia. Project Ignitor differs significantly from that currently under consideration of the projects of fusion reactors based on the tokamak. Tokamak Ignitor has a super strong magnetic field (13 T), in which the pulse discharge (about 10 sec) flows a powerful discharge current (11MA). The Ohmic heating is the main mechanism of ignition of the thermonuclear fusion reaction. It will operate with short pulses (of approximately 10 s in length) and will not have a tritium breeding blanket. The requirements of the tritium fuel cycle are high because three tritium pulses per day are foreseen with a total amount of approximately 10 grams to be processed daily. Fuelling will be based on gas injection.

The main purpose of this stage of research was to determine the current state of Tritium laboratory infrastructure of the complex TSP TRINITI and the preparation of technical proposals for development of the full tritium processing cycle for the task of the Ignitor Project implementing. It necessary to provide a full tritium processing cycle including the storage and supply of the tritium, plasma exhaust purification, separation of the hydrogen isotopes, detritiation of gaseous and water streams, and tritium recovery from the plasma facing components. Creation of the tritium fuel cycle is an important task in a modification of the facility already existing at TRINITI.

Keywords: *fusion reactor, ignition, fuel cycle, TSP TRINITI, tritium complex, Ignitor*

Corresponding author: *Subbotin_ML@nrcki.ru*

Initial Operation Results of Exhaust Detritiation System Using a Polymer Membrane

P3-037

Masahiro Tanaka [a, b]*, Naoyuki Suzuki [a], Hiromi Kato [a], Minoru Yokosawa [c]

[a] *National Institute for Fusion Science, Gifu, Japan*
[b] *SOKENDAI (The Graduate University for Advanced Studies), Gifu, Japan*
[c] *JGC CORPORATION, Yokohama, Kanagawa, Japan*

The deuterium plasma using large fusion test device produces tritium in the vacuum vessel by D-D reaction. The produced tritium is exhausted with deuterium gas via the vacuum pumping system and a part of tritium is retained in the plasma facing components. After the plasma experimental campaign, the vacuum vessel is opened for the maintenance activity and the workers enter the vacuum vessel. To decrease the internal exposure of workers by tritium and prevent a shortage of oxygen, the vacuum vessel is ventilated by room air. The ventilated air contains tritium because the retained tritium is released from the plasma facing components. Then tritium in the air has to be removed by exhaust detritiation system [EDS] from the viewpoints of radiation safety and public acceptance. In National Institute for Fusion Science, EDS for Large Helical Device [LHD] has been installed and operated since 2017. The EDS consists of two systems; one is molecular sieves [MS] type for plasma exhaust gas, other is polymer membrane [PM] type for maintenance activity [1]. The tritium recovery by the polymer membrane has the following advantages: a compact system, reduced energy consumption, continuous operation and ease of maintenance. During the LHD maintenance activity, ventilated air was treated by PM type system and then tritium could be successfully removed from the ventilated air. The tritium removal performance during the maintenance activity and the detailed operation results of the PM system at the initial phase will be presented at the symposium.

[1] M. Tanaka, et al., Fusion Eng. Des., 129, 259–262, (2018).

Keywords: *fusion test device, tritium removal, polymer membrane, exhaust gas*

Corresponding author: *tanaka.masahiro@nifs.ac.jp*

Membrane Gas-Liquid Contactor for Tritium Extraction from LiPb

P3-038

Silvano Tosti [a]*, Alfonso Pozio [b], Luca Farina [a], Marco Incelli [a], Alessia Santucci [a]

[a] *ENEA-Frascati, Frascati (RM), Italy*
[b] *ENEA-Casaccia, S. Maria di Galeria (RM), Italy*

Gas-Liquid Contactors (GLC, namely consisting of bubble columns, packed columns and spray columns) have been studied for the extraction of tritium from LiPb. More recent researches have introduced the Permeator-Against-Vacuum (PAV) concept that uses metal membranes (V, Nb, Ta) immersed into flowing LiPb where the hydrogen isotopes pass through a metal dense wall and are collected in the permeate side via vacuum pumping.

In a new approach, a porous membrane is proposed for the extraction of the hydrogen isotopes from liquid LiPb blankets. In such a device, the liquid LiPb penetrates the pores of the porous membrane without passing through them, then realizing a gas-liquid interface through which the hydrogen isotopes mass transfer takes place. The pore size of this membrane, defined a Membrane Gas-Liquid Contactor (MGLC), has been selected according to the Washburn equation so that the LiPb behaves as a "non-wetting liquid" and does not enter the vacuum phase where leaks of liquid metal are not allowed.

The paper describes the design of a MGLC made up of a porous metal membrane and its experimental characterization at 300-450 °C and 100-500 kPa. A mass transfer model of the MGLC is also presented and verified through the comparison of the measured permeance with the values theoretically calculated.

Keywords: *membrane gas-liquid contactor, tritium, LiPb, liquid breeding blanket*

Corresponding author: *silvano.tosti@enea.it*

Characterisation of Tritium Extraction Unit from Liquid Pb-16Li Alloy of WCLL-TBM in TRIEX-II Facility

P3-039

Marco Utili [a]*, Luigi Candido [b], Alessandro Venturini [c], Mariano Tarantino [a]

[a] *ENEA Brasimone, FSN-ING, Bologna, Italy*
[b] *Dipartimento Energia, Politecnico di Torino, Torino, Italy*
[c] *University of Pisa, Dipartimento di Ingegneria Civile e Industriale, Pisa, Italy*

The experimental qualification of the Tritium Extraction System (TES) from the eutectic Pb-16Li alloy, the breeder material of the Water Cooled Lead-Lithium (WCLL) Breeding Blanket, is one of the fundamental item for the demonstration of tritium balance sustainability for the fusion DEMO reactor. Several technologies have been proposed as TES for DEMO reactor but the selection of the reference technologies can be carried out only after the experimental measurement of the tritium extraction efficiency. For this purpose, a dedicated facility, called TRIEX-II, was designed and installed at C.R. ENEA Brasimone; the facility will be able to qualify Gas Liquid Contactor (GLC), Permeator Against Vacuum (PAV) and Vacuum Droplets Tower technologies at different temperatures, PbLi mass flow rates and hydrogen isotopes concentrations. In TRIEX-II the hydrogen or deuterium, used to simulate tritium, are solubilised inside Pb-16Li with a dedicated saturator and are then extracted from Pb-16Li in the GLC mock-up, which uses helium as stripping gas and works in the temperature range between 300 and 500°C.

This work presents the characterisation of the Gas Liquid Contactor, the reference technology for WCLL TBM of ITER, in TRIEX-II loop, paying particular attention to the accuracy and reliability of the hydrogen isotopes balance in the facility. For this purpose, the hydrogen and deuterium concentration in Pb-16Li has been measured in liquid and in gas phases at the inlet and outlet of the saturator and extractor units, with customised hydrogen permeation sensors and a mass spectrometer. Moreover, to demonstrate that the efficiency of hydrogen isotopes extraction improves when hydrogen is added in the stripping gas, a dedicated experimental campaign is performed with deuterium solubilised in Pb-16Li to simulate tritium and helium plus hydrogen as stripping gas.

Keywords: *Lead-Lithium Eutectic, Tritium Extraction System, Gas Liquid Contactor, TRIEX-II.*

Corresponding author: *marco.utili@enea.it*

Uncertainty Analysis of the Computed Pump Throughputs of the ITER Divertor Gas Exhaust System

P3-040

Nikos Vasileiadis *, Giorgos Tatsios, Dimitris Valougeorgis

Department of Mechanical Engineering, University of Thessaly, Pedion Areos, Volos, Greece

Recently, an integrated software algorithm for modeling complex gas distribution systems operating under vacuum conditions has been developed and successfully implemented to model the ITER divertor pumping system providing the pumped throughputs for several pumping scenarios of the burn and dwell phases [1]. The computed output quantities are subject to the input data, which include the pipe network geometry, approximating the real geometry of the divertor exhaust system and the operating data, namely, the vacuum vessel pressure distribution and the cryopump limiting pumping speed. In the present work the effect of the uncertainties of the input parameters to the pumped throughput, is investigated by performing an uncertainty propagation analysis via the Monte Carlo method. Carrying out the required number of trials for the values of each input uncertainty, accordingly sampled from its respective distribution, the distribution function of the output quantity and its associated uncertainty is obtained. Documenting the effect of uncertainty of the pumped throughputs with regard to the uncertainty of each input parameter is certainly beneficial in judging the expected validity of the modeling and simulation results, as well as, in establishing the optimum operating conditions of the ITER divertor primary gas exhaust system.

Acknowledgment: This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission

[1] N. Vasileiadis et al., Fusion Eng. Des., 103, 125 (2016

Keywords: *Gas distribution systems, kinetic modeling, Monte Carlo, uncertainty propagation.*

***Corresponding author:** *nikovasi@uth.gr*

Experimental Determination of Hydrogen Transport Parameters of 316L Steel in the Two-Side Purge Permeation Setup Q-PETE

P3-041

Axel von der Weth [a], F. Arbeiter [a], D. Klimenko [a], K. Nagatou [b], V. Pasler [a], G. Schlindwein[a]

[a] *Karlsruhe Institute of Technology (KIT), Institute of Nuclear Research and Reactor technology (INR-MET), Eggenstein-Leopoldshafen, Germany*

[b] *Institut für Analysis, Karlsruhe Institute of Technology (KIT), Germany*

Quantitative analyses of hydrogen permeation in the breeder zone and other components of the fuel cycle (ferritic martensitic and austenitic steels) are relevant for safety and tritium self-sufficiency of future fusion power plants. The Q-PETE/D2 experimental setup has been planned and taken into service, with the aim to validate hydrogen transport models and to determine material data. The temperature-controlled permeator setup consists of two gas-purged volumes separated by a metal membrane. Hydrogen permeating through the membrane mixes into the purge gas flow and is measured by a mass spectrometer.

In the first series of experiments, a stainless steel (DIN 1.4404, AISI 316L) membrane of 1.14mm thickness was used. The hydrogen content in the purge gas on the feed side was at 3000 ppm. The investigated temperature range was 300 – 500 °C.

A solver based on the finite differences method allows the simulation of the expected permeation fluxes of the experiment, based on input of the temperature dependent Diffusion constant and Sieverts´ constant and experimental boundary conditions. An iterative optimization routine based on a Branch and Bound algorithm was used to extract the effective Diffusion and Sieverts´ constants as inverse problem of the conducted experimental permeation data.

In this paper, the effective Diffusion and Sieverts constants for the used membrane are presented. Sensitivity on analysis parameters is discussed. The obtained values are compared to literature data of the same steel grade.

***Corresponding author:** *axel.vonderweth@kit.edu*

Diffusion Bonding Experiments of 1.4404 Steels in a Gleeble 3800 Thermomechanical Simulator for Investigation of Non-Destructive Inspection Methods

P3-042

Tétény Baross [a]*, Péter Bereczki [b], László Jánosi [c], Miklós Palánkai [a], Botond Sánta [d] and Gábor Veres [a]

[a] *MTA Wigner FK RMI, Budapest, Hungary*
[b] *Department of Material Science, University of Dunaújváros, Dunaújváros, Hungary*
[c] *Department of Mechatronics, Szent István University, Gödöllő, Hungary*
[d] *Department of Physics, Budapest University of Technology and Economics, Budapest, Hungary*

Diffusion bonding methods will be the most significant welding methods for Plasma Facing Components in fusion reactors like ITER or in later decades the DEMO reactor.

For diffusion bonding physical simulation the Gleeble 3800 thermomechanical simulator provides a non-conventional heating method. Instead of having a furnace with uniform radiation heating of the bulk material in a vacuum chamber, Gleeble applies direct resistance heating with 50 Hz alternating current passed through the specimens up to 10,000°C/second. Specimens are gripped by jaws that lead to a temperature distribution along the specimens in vacuum chamber, were the system provides a flexible control also for axial static pressure up to 20 ton.

Analytical and discretized heat conduction model was built up investigating the mating surface thermal properties in FE modeling. The joined surface with different percent of bonded area has different thermal and electrical conductance that have to be measured by standard, but sensitive methods. Since the diffusion welding of 1.4404 are well known - development of non-destructive inspection methods are envisaged for steels diffusion welding. The authors intend to summarize the results of the specimens welding compared to the non-destructive inspection methods.

Further goal is to prepare the Gleeble physical simulations for the newly developed fusion material as ODS steels, which weldability is a key issue at fusion reactor first walls.

***Corresponding author:** *baross.teteny@wigner.mta.hu*

Development, Characterization and R&D Activities on Lithium Ceramic Breeder Materials in India

P3-043

Paritosh Chaudhuri [a,b]*, Maulick Panchal[a], Aroh Shrivastava[a]

[a] *Institute for Plasma Research, Bhat, Gandhinagar, Gujarat, India*
[b] *Homi Bhabha National Institute, Anushaktinagar, Mumbai, India*

Lithium meta-titanate (Li_2TiO_3) and Lithium ortho-silicate (Li_4SiO_4) are considered to be the suitable candidate materials for tritium breeders. India has developed and prepared Li_2TiO_3 as the tritium breeder materials for fusion blankets. Li_2TiO_3 power was prepared by solid state reaction using $LiCO_3$ and TiO_2 followed by ball-milling and calcination. Li_2TiO_3 pellets and pebbles are prepared from this powder followed by high temperature sintering. Effect of sintering time and temperature on the properties of pebbles has been studied. At every stage of preparation, extensive characterizations are being carried out to meet the desired properties of these materials. For a robust design of blankets requires a thorough understanding of the thermo-mechanical response of the breeder materials at different loading conditions. In this context, the material characterization plays a vital role in determining the breeder response. It is essential to measure the mechanical and thermo-mechanical properties of pebble bed. Experimental set ups have been built indigenously at IPR for the measurement of effective thermal conductivity of pebble bed using steady state-axial heat flow and transient hot wire methods. Results obtained from these experiments and also the future scope will be discussed in this paper. Details of lithium ceramic breeder material development, their characterizations, material database preparation and related R&D activities will be discussed in this paper.

***Corresponding author:** *paritosh@ipr.res.in*

Direct Observation of Hydrogen Permeation Through Grain Boundaries in Tungsten

P3-044

Pablo Díaz-Rodríguez [a*], M. Panizo [a], R. González-Arrabal [a,b], O. Peña-Rodríguez [a,b], I. Martín-Bragado [c], C. González [d], R. Iglesias [e], J.M. Perlado [a,b], A. Rivera [a,b]

[a] *Instituto de Fusión Nuclear Guillermo Velarde, Universidad Politécnica de Madrid, Madrid, Spain*
[b] *Departamento de Ingeniería Energética, ETSI Industriales, Universidad Politécnica de Madrid, Spain*
[c] *UCAM, Universidad Católica de Murcia, Murcia, Spain*
[d] *Department of Theoretical Condensed Matter Physics & Condensed Matter Physics Center (IFIMAC), Universidad Autóonoma de Madrid, Madrid, Spain*
[e] *Department of Physics, University of Oviedo, Oviedo, Spain*

The influence of grain boundaries (GB) on hydrogen permeation in tungsten has been studied both experimentally and by computer simulations, based on Monte Carlo methodology. For this purpose, hydrogen permeation experiments were performed with two different sets of nanostructured tungsten (NW) samples, namely self-damaged NW and non-damaged NW. These studies were carried out at different temperatures (473 to 723K) and hydrogen pressures (100 to 8000 mbar). To attain a better understanding of the processes involved, Object Kinetic Monte Carlo simulations parametrized with Density Functional Theory (DFT) data were performed. Both experimental and simulation results show clearly: a sudden rise in the permeation flux until a plateau is attained (stationary regime) associated to fast hydrogen migration through the GBs, followed by a second, low slope raise until reaching a second plateau, due to the slower hydrogen permeation through the tungsten bulk, which it is only appreciable once the GBs have been saturated with hydrogen atoms. Hydrogen permeation though the bulk is delayed in the case of the self-damaged NW samples, due to the presence of traps, which makes the permeation route through the GBs even more dominant. GBs were considered preferential paths for hydrogen atoms through indirect experiments combined with modelling. Now, this approach allows us to obtain a direct evidence of the dominant role of GBs on hydrogen permeation. This has important implications for the design of materials with superior capabilities to release hydrogen.

Keywords: *Tungsten, Hydrogen, grain boundary, permeability*

Corresponding author: *p.diazr@upm.es*

Deuterium Retention in Reduced Activation Ferritic/Martencitic Steels (RAFMS) at ELM-like Pulse Plasma Heat Loads

P3-045

Anna V. Golubeva [a*], Dmitry V. Kovalenko [b], Vitaly S. Efimov [c], Yury M. Gasparyan [c], Vladimir A. Barsuk [b], Viacheslav M. Chernov [d], Dmitry A. Terentyev [e], Nikolay S. Klimov [b], Artem A. Mednikov [a], Nikolay P. Bobyr [a], Dmitry A. Kozlov [a]

[a] *NRC "Kurchatov Institute", Moscow, Russia*
[b] *SRC RF TRINITI, Moscow, Russian Federation*
[c] *National Research Nuclear University MEPhI (Moscow Engineering Physics Institute), Moscow, Russia*
[d] *JSC "A.A. Bochvar High-Technology Research Institute of Inorganic Materials", Moscow, Russia*
[e] *Structural Materials Group, Institute of Nuclear Materials Science, SCK•CEN, Mol, Belgium*

The reduced activation ferritic-martencitic steels (RAFMS) are promising structural materials for fusion. Some authors also propose to use RAFMS as plasma-facing materials in areas with relatively low plasma and heat loadings. In the present work the 2 mm thick samples of Rusfer (EK-181) and Eurofer RAFMS were irradiated at QSPA-T facility by ELM-like pulse deuterium plasma. Two levels of heat loads on the samples were selected: 0.3 MJ/m^2 and 0.6 MJ/m^2. The pulse duration was ~1 ms, number of pulses was in a range of 1÷50. At the plasma heat loading of 0.3 MJ/m^2 surface layers of RAFMS was not melted but cracks appeared at surfaces of both materials. The density of cracks at Eurofer surface is about twice lower. After loading with 0.6 MJ/m^2 the surface of Rusfer samples is waved. The recrystallization and the cracking of Rusfer surface is more pronounced than of Eurofer surface. Deuterium retention in RAFMS after plasma irradiation was investigated by thermodesorption measurements. If surface layer melts (at loading with 0.6 MJ/m^2) deuterium retention in RAFMS is several times higher than at loading below melting threshold (0.3 MJ/m^2). The maximum amount of deuterium retained in RAFMS samples was 10^{21} D/m^2.

This work was supported by RFBR grant # 18-58-53026 GFEN_a

Keywords: *RAFMS, plasma heat load, deuterium, retention*

Corresponding author: *golubeva_av@nrcki.ru*

Validation Study of Turbulence Models for Thermal-Hydraulic Simulation of Helium Cooled DONES High Flux Test Module

P3-046

Sergej Gordeev *, Frederik Arbeiter

Karlsruhe Institute of Technology, Karlsruhe, Germany

Helium flow at low pressure (0.3 MPa) is used to cool the specimen capsules and the structure of the neutron irradiated High Flux Test Module (HFTM) of the DEMO-Oriented Neutron Source (DONES). The flow path includes mini-channels with gap-widths less than 1 mm where a high velocity low Reynolds number helium flow is used as cooling medium. The large span of Reynolds numbers from laminar to fully turbulent is a significant challenge for the simulation of the complete HFTM. Additionally such effects as acceleration of the heated gas flow, compressibility and secondary flows cannot be neglected in the heat transfer analysis of this gaseous mini-channel flow.

Four turbulence models offered in the commercial CFD code StarCCM+, the Reynolds Stress transport (RSM); the Realizable k-ε (RKE) model; the k-ω Shear-Stress-Transport (SST) model and the V2F model were tested using experimental results obtained in the ITHEX (IFMIF Thermal-Hydraulic Experiment) and HFTM-SR (HFTM Single Rig) experiments conducted in in Karlsruhe Institute of Technology (KIT). Numerical simulations have been performed for low Reynolds number (Re=6000-10000) helium flows in a heated mini-channels. The RSM model provides the best heat transfer prediction for the full range of Re numbers. In the SST model the turbulent time scale is calculated using Durbin's realizability constraint, implemented into the eddy viscosity formulation. This realizability constrained was systematically varied, and best predictions have been achieved with the variable constraint with continuous decreasing from the standard value of 0.66 for "turbulent" case (Re=10000) down to 0.31 for the "laminar" case (Re=4500). The RSM as well as the SST (non-linear formulation) models are able to reproduce the secondary flows in the channels with rectangular cross sections. Both turbulence models can be considered as appropriate numerical tools for the heat transfer analysis of HFTM. This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission, Fusion for Energy, or of the authors' home institutions or research funders.

Keywords: *DONES, HFTM, CFD, heat transfer, turbulence*

****Corresponding author:*** *sergej.gordeev@kit.edu*

Application of Friction Stir Processing on CuCrZr to Improve Material's Property

P3-047

Dai Hamaguchi [a]*, Yoshiaki Morisada [b], Hidetoshi Fujii[b], Hiroyasu Tanigawa [a]

[a] *Rokkasho Fusion Institute, National Institute for Quantum and Radiological Science and Technology, Rokkasho, Aomori, Japan*

[b] *Joining and Welding Research Institute, Osaka University, Ibaraki, Osaka, Japan*

Copper alloy is the candidate material for cooling components of DEMO divertor. CuCrZr is a first choice among others but issues related to quality control during manufacturing and also among fabrication of the divertor components still remains. CuCrZr also exhibit some weakness against neutron irradiations. One of the keys to deal with these issues is a grain-refinement.

Friction Stir Processing (FSP) is a solid-state process where a rotational tool is plunged into the work piece to produce local friction heating inducing complex material flow and intense plastic deformation that leads to grain-refinement, which also may improve irradiation resistivity. The purpose of this study is to examine the applicability of FSP on CuCrZr to improve material's performances.

We first examined the applicability of FSP on Cu materials using OFC and found that the most effective grain refinement and hardness increase can be achieved at the rotation speed of 200 to 300rpm with a vertical force of 1.5ton. However, when increasing the rotation speed to 500rpm, the grain-refinement was not effective enough and hardness did not improve. Based on the results, the examination on CuCrZr alloy was conducted. The material used in this study were newly produced CuCrZr plate that matches the ITER-Gr criteria. FSP tests were performed with the rotation speed of 200 to 700rpm with vertical force of 1.5, 2.0, and 2.5tons. As a result, grain-refinement and hardness increase were achieved in all conditions, even at 500rpm and above. However, introduction of cavities and local cracks were seen below 500rpm. Furthermore, these defects were introduced even at 600rpm when the vertical force was set to 1.5ton. The result indicates that the rotation speed and vertical force can be much higher for CuCrZr compared to pure Cu. More details will be reported on the presentation.

Keywords: *CuCrZr, FSP, grain-refinement, irradiation resistivity*

****Corresponding author:*** *hamaguchi.dai@qst.go.jp*

Temperature Parallel Simulated Annealing with Self-Generated Basins for Searching the Stable State of Microstructures in Materials

P3-048

Sho Hayakawa [a*], Kyosuke Shibasaki [a], Taira Okita [b]

[a] *School of Engineering, University of Tokyo, Tokyo, Japan*

[b] *Research into Artifacts, Center for Engineering, University of Tokyo, Kashiwa, Japan*

Ensuring the reliability of fusion structural materials is essential to building fusion reactors. A fusion reactor environment is characterized by 14-MeV neutron irradiation, which generates numerous irradiation-induced defects in materials, leading to considerable degradations in material properties. Considering the current situation in which testing fusion reactors are in the development phase, modeling approaches play a major role in predicting the behavior of materials. Molecular dynamics (MD) studies have been extensively conducted to clarify the processes involved in the defects formation because of the atomic-scale fidelity of MD, and the insights extracted from these studies have contributed considerably to a comprehensive understanding of the microstructural evolution of irradiated materials. However, certain processes subsequent to the defects formation occur far beyond the MD timescale, suggesting that developing schemes for bridging the large gap in timescale is essential. In this study, we developed temperature parallel simulated annealing with self-generated basins (TPSA/SGB) to enable searching the energetically stable state of microstructures in materials beyond the MD timescale. TPSA/SGB is based on simulated annealing (SA), but it overcomes the disadvantages of SA through parallel computing. Furthermore, neighboring points for the SA algorithm are searched by considering the potential energy surface of the system, enabling the accurate and effective sampling of events that the system experiences. We applied TPSA/SGB to defect evolution processes in body-centered cubic iron and benchmarked its performance. A detailed explanation of the method and comparison with MD are given in the presentation. This study contributes considerably to the multi-scale modeling of materials behavior under irradiation by providing input parameters to higher-level modeling techniques such as phase-field models and the finite element method.

Keywords: *mesoscale simulation, lattice defect, BCC-Fe*

Corresponding author: *hayakawa@race.u-tokyo.ac.jp*

Study of Extremely Low Nitrogen Concentration Lithium by Fe-5Ti Alloy Hot Trap

P3-049

Takuma Higashi [a*], Taichi Koyama [a], Juro Yagi [b], Takayuki Terai[a]

[a] *School of Engineering, The University of Tokyo, Yayoi, Tokyo, Japan*

[b] *Institute of Advanced Energy, Kyoto University, Uji, Kyoto, Japan*

In the lithium loop of IFMIF (International Fusion Materials Irradiation Facility), the target value of the nitrogen concentration in lithium flow is designed to be <10 wt.ppm from the viewpoint of corrosion prevention. However, the optimum method for obtaining this extremely low nitrogen concentration lithium (<10 wt.ppm-N) has not yet been established. In this study, therefore, we aimed to investigate the hot trap method using Fe-5Ti alloy grains and the nitrogen concentration measurement method for the extremely low nitrogen concentration in lithium. When lithium was handled in an argon atmosphere glove box (>99.9999%), it was contaminated by nitrogen as an impurity of the cover gas. Therefore, the hot trap experimental pot sealed was placed in a vacuum exhaust glove box. In order to minimize nitrogen contamination, lithium sampling for nitrogen concentration measurement, was carried out by scraping out lithium in a solid state. Then, nitrogen concentration was determined by the ammonia conversion method. As a result of the hot trap experiment under the above conditions, the nitrogen concentration in lithium was reduced from 43 to 10 wt.ppm by immersing Fe-5Ti grains (~160 μm) at 550°C after 192 hours, and from 65 to 9.9 wt.ppm at 600°C after 144 hours.

Keywords: *FMIF, lithium, nitrogen concentration, hot trap, Fe-5Ti alloy*

Corresponding author: *takuma.higashi0728@gmail.com*

Evaluation of Fatigue Properties of Reduced Activation Ferritic/Martensitic Steel, F82H for Development of Design Criteria

P3-050

Takanori Hirose*, Taichiro Kato, Hideo Sakasegawa, Hiroyasu Tanigawa

National Institutes for Quantum and Radiological Science and Technology, Rokkasho, Aomori, Japan

Reduced activation ferritic/martensitic (RAFM) steels are promising candidate for structural material of tritium breeding blanket in a fusion reactor. Database accumulation and definition of design criteria for RAFM have been intensively studied together with the progress of blanket design activities. As a part of database accumulation for fusion

blanket, a RAFM steel, F82H was fatigue-tested at 673 and 723 K in the air. Axial strain-controlled fatigue tests were carried out with a cylindrical specimen with 8 mm of diameter with -1 of strain ratio condition in accordance with Japanese Industrial Standard, JIS Z 2279, "Method of high-temperature low cycle fatigue testing for metallic materials." For high cycle tests, the maximum test cycles exceeded 10^6 cycles.

Fatigue lifetime can be estimated from tensile properties using an empirical equation like universal slope method. The lifetime at 673 K was found to fall into the scatter band of a lifetime at temperature ranging ambient temperature to 673 K as the temperature dependence of tensile properties demonstrated. It was studied using experimental results that fatigue-related design limit based on RCC-MRx, such as fatigue curves, half-life cyclic curves, and related coefficients.

Keywords: *Reduced activation ferritic/martensitic steel, F82H, Fatigue, Failure, elevated temperature,*

Corresponding author: *hirose.takanori@qst.go.jp*

Interaction Between Surface Behavior and Inner Flow Pattern of Liquid Li Jet for Fusion Neutron Sources

P3-051

Eiji Hoashi [a]*, Yasuhiro Fujii [a], Shunya Yamada [a], Takafumi Okita [a], Hiroo Kondo [b,c], Chang Ho Park [b], Makoto M. Nakamura [b], Kentaro Ochiai [b], Atsushi Kasugai[a]

[a] *Graduate School of Engineering, Osaka University, Osaka, Japan*
[b] *National Institutes for Quantum and Radiological Science and Technology, Kamikita-gun Aomori, Japan*
[c] *Currently: Mitsubishi Heavy Industries, Ltd., Kobe Hyogo, Japan*

Liquid lithium (Li) jet is planned as a beam target in fusion neutron sources (FNSs), such as the international fusion materials irradiation facility (IFMIF) in Japan and EU, advanced fusion neutron source (A-FNS) in Japan and IFMIF-DEMO-oriented neutron source (IFMIF-DONES) in EU. For the safety and the efficiency of such FNSs, it is desirable to keep the high-speed Li jet stable. In addition to many experimental researches using Li loops at QST (former JAEA) and Osaka University, numerical approaches using computational fluid dynamics (CFD) simulation have been also required in order to clarify the mechanism of the surface fluctuation of the Li jet because Li is an opaque metal. In our previous simulation, it was confirmed that the vortex structure under the free surface of the Li jet had the strong influence on the surface fluctuation using Large Eddy Simulation (LES) as a turbulent model and the vortices were generated in the boundary layer inside the two-staged contraction nozzle. In this study, in order to clarify the interaction between the surface behavior and the inner flow pattern of the Li jet, two simulations are conducted: one is the LES simulation considering wall roughness, and another one is the Reynolds-Averaged Navier-Stokes (RANS) simulation using full-scale model with the nozzle and the Li jet part. The influence of wall roughness on the vortex structure becomes an issue in long-term operation of actual FNSs, and the influence of the secondary flow generated due to the side wall on the surface shape is evaluated in the latter simulation.

Keywords: *Fusion neutron source, Liquid Li target, CFD, Free surface flow, Wall roughness*

Corresponding author: *hoashi@see.eng.osaka-u.ac.jp*

Qualification Tests of an Electrochemically-Based H-Sensor for Application in Liquid Lithium of IFMIF-DONES

P3-052

Nils Holstein [a]*, W. Krauss [a], F.S. Nitti [b]

[a] *Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany*
[b] *ENEA, Brasimone, Camugnano (BO), Italy*

The knowledge of the radiation induced degradation for future fusion systems is planned to be investigated in IFMIF-DONES, where fast neutrons are produced by a reaction of a D-beam with a liquid lithium target. One critical issue is the control of hydrogen impurity concentrations in the Li-melt. Electrochemistry offers an ideal diagnostic tool to measure directly impurity concentrations in liquid metal media by monitoring ElectroMotive Forces (EMF). However, such H-sensors do not exist and their development and qualification is the objective of an EUROFUSION task for operation in the liquid lithium loop of IFMIF-DONES systems.

This presentation will outline the background of measuring non-metallic impurities in molten metals by detecting electrochemical potentials, and their transformation into concentrations. Liquid lithium is a very reactive melt, thus the suitable selection of used chemical materials will be discussed. Based on these issues, a suitable design and a prototype series were developed. The assembling of the sensor components and the synthesis of the electro-active chemicals will be shown together with necessary heat treatments for activation of the measuring cell. The first test campaign was designed to show the functionality of the developed H-sensor, and showed successful response behavior of the sensor with different hydrogen concentrations. The observed EMF potentials were in good accordance compared to with modeled values.

Acknowledgments: This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *Liquid lithium, Impurities, Electrochemical hydrogen sensor, IFMIF-DONES*

*Corresponding author: *nils.holstein@kit.edu*

Experimental Study on Erosion-Corrosion Behavior of CuCrZr Weldment in Rotating Nanofluid

P3-053

Haiyan Jiang [a]*, Weihua Wang [b], Delin Chu [b], Wei Lu [b], Yucheng Wu [a,c]*

[a] *School of Materials Science and Engineering, Hefei University of Technology, Hefei, Anhui, China*
[b] *Institutes of Physical Science and Information Technology, Anhui University, Hefei, Anhui, China*
[c] *National–Local Joint Engineering Research Centre of Nonferrous Metals and Processing Technology, Hefei, Anhui, China*

Due to its higher heat transfer efficiency and improving the critical heat flux(CHF), nanofluid is considered as candidate coolant for plasma facing components(PFC). CuCrZr alloy is proposed for the primary heat sink material in blanket and divertor. Welding is an indispensable technology in the manufacture of fusion reactor components. For the influence of welding heat treatment process, CuCrZr weldment may be very different from the base metal. To ensure long-term stable operation of fusion reactor, several experiments were conducted in rotating corrosion test facilities to evaluate the compatibility of CuCrZr weldment in Al_2O_3-water nanofluid. The experimental parameters were flow velocity of 1.00m/s and 3.25m/s, fluid temperature of 70°C, testing duration of 3000h, nanofluid mass fraction of 1wt.% and 3wt.%, etc. Before and after the test, the weights of specimens were measured and the observation by SEM and analysis in compositions by XPS were performed respectively. The experiment results show that weld seam corrosion is more serious than parent metal and the corrosion rate decreases with increasing corrosion time. More experiments will be conducted to evaluate the corrosion behavior and mechanism of welded CuCrZr alloy.

Keywords: *nanofluid, erosion-corrosion, CuCrZr alloy, weldment*

*Corresponding author: *jianghaiyan@hfut.edu.cn*

Effects of Sintering Conditions on the Microstructure of Li_2TiO_3 Tritium Breeding Materials

P3-054

Jong-Il Kim *, Yi-Hyun Park, Mu-Young Ahn , YoungMin Lee, Seugyon Cho

National Fusion Research Institute (NFRI), Daejeon, Korea

Lithium containing ceramics (Li_2TiO_3, Li_4SiO_4, Li_2O, Li_2ZrO_3 and Li_2AlO_2) have been proposed as candidate breeder materials for fusion reactor blankets. For amicable tritium release, the breeder materials are recommended to have homogeneous microstructure with proper grain sizes which can be strongly affected by the sintering conditions such as temperature, heating rate, holding time, atmosphere and so on. In this study, the effects of sintering conditions on the microstructure of sintered pebbles were investigated. Additionally, in some of experiments, the crucible cover was removed during the sintering to prevent relatively high partial pressure of CO_2 which was generated by the pyrolysis step from the binder. Phase analysis and morphological observation were carried out by Xray diffraction and electron microscopy, respectively. The sintered pebbles tended to show abnormal grain growth with increasing heating rate and exhibited rapidly grain growth at above 1000°C. On the other hand, the sintering temperature of 900°C showed a relatively proper grain size (8um), but the difference of grain growth was observed depending on the presence of crucible cover. Th e grain size is greater when there is no cover. It is presumed to be due to the effect of CO_2. The optimal sintering conditions for homogeneous microstructure of the Li_2TiO_3 pebbles were 900°C and 0.5h in air without cover. It was confirmed through observation of microstructure that these conditions are corresponding to intermediate stage in the sintering step.

Keywords: *Li_2TiO_3 pebbles, grain growth, sintering conditions*

*Corresponding author: *jikim@nfri.re.kr*

Microstructural Stability of Tungsten Coated CFC Plates Aiming to Metal Wall Experiments at JT-60SA

P3-055

Hirotatsu Kishimoto [a,b]*, Watatu Kondo [b], Naofumi Nakazato [a,b], Masakatsu Fukumoto [c]

[a] *OASIS, Muroran Institute of Technology, Hokkaido, Japan*
[b] *College of Design and Manufacturing Technology, Muroran Institute of Technology, Hokkaido, Japan*
[c] *National Institutes for Quantum and Radiological Science and Technology, Ibaraki, Japan*

A satellite tokamak, JT-60SA, is being built up at Naka in Japan. The first plasma of the tokamak is planned to be in 2020. The progressing of the JT-60SA research is about 5 years earlier than of the ITER, being planned to contribute the research of ITER and developments for DEMO. A plan for the contribution is tungsten wall experiments at JT-60SA prior to the experiment at ITER. JT-60SA is planned to be modified the wall from the initial carbon wall to the tungsten one in 2030. Divertors of the JT-60SA will be also exchanged to tungsten, but the installation methods and shapes of the divertors need to be developed because of the significantly heavier weight of tungsten than the carbon. Tungsten coated carbon structure is an idea for the installation of tungsten divertor for JT-60SA. The tungsten plates are able to be joined on the carbon fiber strengthened carbon (CFC) composites using sinter bonding methods. Present research reports the optimization of the joint conditions and microstructural stabilities at elevated temperature.

Keywords: *JT-60SA, Divertor, tungsten, CFC, joining method.*

***Corresponding author:** *hkishi@mmm.muroran-it.ac.jp*

Effects of Specimen Thickness on Creep Properties of F82H Steel

P3-056

Takuya Nagasaka [a]*, Jingjie Shen [a], Takeo Muroga [a], Masami Ando [b], Hideo Sakasegawa [b], Taichiro Kato [b], Hiroyasu Tanigawa [b], Takashi Nozawa [b]

[a] *National Institute for Fusion Science, Toki, Japan*
[b] *National Institutes for Quantum and Radiological Science and Technology, Rokkasho, Japan*

A reduced-activation ferritic/martensitic steel, F82H steel, is the primary candidate structural material for fusion reactor blanket. Neutron irradiation creep properties are evaluated using creep tubes with the wall thickness of around 0.2 mm. Since deformation volume is much smaller than that of standard size specimens, it is essential to reveal the specimen size effect on creep properties to validate the results by the creep tubes. The purpose of the present study is to investigate the effects of thickness of the miniature specimens on creep properties.

Miniature tensile-type specimens with a gauge length of 5 mm and a gauge width of 1.2 mm were machined from a 15 mm-thick plate of F82H-IEA heat with a chemical composition of Fe- 7.71Cr- 1.95W- 0.091C -0.16V -0.02Ta -0.11Si -0.16Mn -0.002P -0.002S -0.006N. The gauge thickness varied from 0.14 mm to 1.2 mm. Creep tests were performed under 75 to 250 MPa at 550 and 650°C. Some of the tests were terminated before fracture for precision measurement of the creep strain in direct with an optical digital microscope.

The creep rupture time under 120 MPa at 650°C was 219, 198, 360, 307, 313 and 320 h for 0.141, 0.268, 0.525, 0.774, 1.037 and 1.226 mm in gauge thickness, respectively. The rapture time of the first and the second thinnest specimens was considerably shorter than the thicker specimens. One of possible mechanisms for the enhanced creep is additional deformation due to dislocation motion activated in the unconstraint surface grains. Unconstraint condition also could enhance other diffusional creep and re-distribution of dislocations around the surface. In order to discuss these mechanisms, microstructures of the surface grains were analyzed.

Keywords: *Small specimen testing technology, blanket structural material, surface effect*

***Corresponding author:** *nagasaka@nifs.ac.jp*

Temperature Dependency of Corrosion Properties of F82H in High Temperature Water

P3-057

Motoki Nakajima *, Takashi Nozawa, Hiroyasu Tanigawa

National Institute for Quantum and Radiological Science and Technology, Aomori, Japan

Reduced activation ferritic/martensitic steel (RAFM), F82H, is the leading candidate structural material for the Japanese water-cooled solid breeder blanket, which is an attractive concept given its compactness and its compatibility with existing technologies of the conventional light water reactor. The coolant water of the blanket system generally shows a temperature gradient and understanding the temperature dependency of corrosion properties, e.g., the corrosion rate, is critical to not only designing the water cooling system but also predicting the amount of corrosion products. This study therefore aims to identify the effects of temperature on corrosion properties of F82H.

The material used in this study was F82H grade BA07. The dissolved oxygen concentration, dissolved hydrogen concentration, and pH were <5 ppb, 3.5 ppm and 7~9 at room temperature, respectively. The corrosion test under flowing water condition was in contrast performed at 423 and 543 K using a rotating disk specimen (O.D. = 275 mm, I.D. =

130 mm, thickness = 3 mm). The circumferential velocity on the specimen outer edge and inner edges were 5.0 m/s and 3.1 m/s, respectively. The surface and cross-section of the specimen were examined to characterize corrosion products by scanning electron microscopy, X-ray diffraction, and electron probe micro-analyzer.

In case of the flowing hydrogen-added deaerated water condition, it was found that the specimen weight continuously decreased at any temperatures of concern, resulting in a clear increase of the corrosion rate with increasing test temperature. It has been reported that the flow corrosion rate of typical carbon steel has a peak around 423 K attributable to a solubility of the magnetite. However, the magnetite did not occur in any temperatures in the flow corrosion condition for F82H, resulting in such increase of the flow corrosion rate at higher temperatures. Detail analysis of oxide morphology, SCC initiation and propagation, and corrosion product will also be reported in this study.

Keywords: *RAFM, Corrosion, Temperature dependency, High temperature water*

Corresponding author: *nakajima.motoki@qst.go.jp*

Simultaneous Effects of Applied Stress and Dissolved Oxygen on Surface Morphology of Steels for Cooling Systems of Blanket Module in Pressurized Water

P3-058

Naofumi Nakazato [a]*, Hiroki Mima [a], Taisei Nemoto [a], Motoki Nakajima [b], Takashi Nozawa [b], Hiroyasu Tanigawa [b], Hirotatsu Kishimoto

[a] *Muroran Institute of Technology, Muroran Hokkaido, Japan*
[b] *Nathonal Institute for Quantum and Radiological Science and Technology, Rokkasho, Aomori, Japan*

The R&Ds of blanket system is one of the important issues for the realization of fusion reactor. A water-cooled solid breeder blanket system is the Japanese promising design, and the candidate structural material of it is planned to be a reduced activation ferritic/martensitic steel, F82H. Cooling channels of F82H steel will be built in the blanket, and connected to the cooling system which will locate at the out of fusion reactor. The cooling pipes of austenitic stainless steel, SUS316L, will be welded to the F82H cooling channels at the back plate of the blanket, and the same cooling water will flow both F82H cooling channels and SUS316L cooling pipes. Therefore, it is necessary to find out an appropriate water chemistry condition that can maintain the soundness of each structural material. Also, it is considered that the stress will be caused in a part of F82H cooling channels and SUS316L cooling pipes by assembling, which needs to evaluate the effect of applied stress on surface morphology of both steels. In the fusion blanket, the cooling water will be irradiated by neutrons and brought about the radiolysis of water. It is assumed that water chemistry conditions will be changed and the oxygen contents will be enhanced. This research focuses on the surface modification on both F82H and SUS316L steels in pressurized water which is adjusted to be water chemistry conditions considering the radiolysis of water. In this presentation, the simultaneous effects of applied stress and dissolved oxygen on surface morphology of F82H and SUS316L will be provided. The compatibility tests are performed using pure water at 290°C under 9 MPa. The dissolved oxygen concentration is up to 100 ppb. The surface modification will be evaluated precisely mainly by XPS, and EPMA and TEM will be also used for the discussion.

Keywords: *water-cooled solid breeder blanket, structural material, compatibility, pressurized water, surface morphology*

Corresponding author: *nakazato@mmm.muroran-it.ac.jp*

Non-Contact Strain Evaluation for Miniature Tensile Specimens of Neutron-Irradiated F82H by Digital Image Correlation

P3-059

Takashi Nozawa [a]*, Hideo Sakasegawa [a], Xiang Chen [b], Taichiro Kato [a], Josina W. Geringer [b], Yutai Katoh [b], Hiroyasu Tanigawa [a]

[a] *National Institutes for Quantum and Radiological Science and Technology, Rokkasho, Aomori, Japan*
[b] *Oak Ridge National Laboratory, Oak Ridge, TN, USA*

The small specimen tensile test is commonly used as the basic test to evaluate the irradiation effect on the strength of fusion structural materials after irradiation, and a lot of attempts were made to qualify the data obtained by small size specimen, but the specific attention on the evaluation of strain were not satisfactory given. This paper therefore aims to propose a novel non-contact strain measurement technique, based on digital image correlation (DIC) method, for reduced-activation ferritic/martensitic (RAFM) steel as the leading candidate structural material of fusion in-vessel components. Material of concern was Japanese RAFM, F82H. A flat rectangular miniature specimen, SS-J3, was adopted. Nonirradiated and neutron-irradiated tensile properties were evaluated. Note that neutron irradiation was conducted in the HFIR at ORNL to doses of ~80dpa at irradiation temperature of ~300°C. In the proposed DIC, very tiny surface machining flaws were recognized by the DIC computation program and utilized to set several pairs of the gauge end points with a fixed gauge length. Multiple line measurements were then conducted over the gauge width and averaged data can provide the representative longitudinal strain of the specimen. In case of the room-temperature test case, it was clearly demonstrated that this procedure could provide more precise and reliable data compared with the previous approach of the single-scan measurement under floated gauge length; Superior linearity was achieved and marked points of concern were traceable during the entire period of the test. This study also addresses on feasibility of the proposed test method at elevated temperatures. Besides, other tensile parameters, e.g., reduction of area, will be discussed to provide a general test guideline.

Keywords: *reduced-activation ferritic/martensitic (RAFM) steel, small specimen test technique (SSTT), tensile, non-contact extensometry, digital image correlation (DIC)*

Corresponding author: *nozawa.takashi@qst.go.jp*

Conceptual Design of Test Modules for DEMO Blanket, Diagnostic Device, and RI Production for A-FNS

P3-060

Masayuki Ohta *, Satoshi Sato, Makoto M. Nakamura, Saerom Kwon, ChangHo Park, Kentaro Ochiai, Atsushi Kasugai

National Institutes for Quantum and Radiological Science and Technology, Rokkasho, Aomori, Japan

Conceptual design activity of advanced fusion neutron source A-FNS is being performed by QST at Rokkasho, Japan. A-FNS is an intense neutron source by d-Li reaction based on achievements of IFMIF Engineering Validation and Engineering Design Activities (IFMIF/EVEDA). We are planning irradiation tests of fusion reactor materials by using eight test modules at A-FNS. In this presentation, we report a progress of design activities on Blanket Nuclear Property Test Module (BNPTM) and Diagnostic Controlling device Test Module (DCTM) associated with DEMO reactor, out of the eight modules. In the BNPTM, neutron flux, tritium production rate, nuclear heating, etc. are to be measured inside DEMO blanket module with the mock-ups in order to evaluate accuracy of nuclear analysis of the DEMO blanket. In the DCTM, irradiation data of functional materials on diagnostic controlling devices which are mirror, window, optical fibers, etc. are to be achieved. In addition to the irradiation tests on the fusion material, various neutron applications are planned at A-FNS. One of the applications is medical isotope production. We also report that of RadioIsotope Production Module (RIPM). In the RIPM, medical RIs such as molybdenum-99 are to be produced. The RIPM equips with a transportation system like a pneumatic tube from outside to inside the test cell of A-FNS, because samples for medical RI production are put in and out frequently during an irradiation term for fusion material tests because of their short half-lives. The RIPM is also to be used for neutron measurement with the activation method. These modules are peculiar to A-FNS which are not planned to be installed at IFMIF-DONES.

Keywords: *fusion neutron source, A-FNS, DEMO blanket, diagnostic controlling device, radioisotope*

Corresponding author: *ohta.masayuki@qst.go.jp*

Verification of Accuracy of Contact-Probe Distance Meter for Lithium Target of Fusion Neutron Source

P3-061

Takafumi Okita [a]*, Shota Entani [a], Kotaro Watanabe [a], Masahiro Higashi [a], Eiji Hoashi [a], Hiroo Kondo [b,c], ChangHo Park [b], Makoto M. Nakamura [b], Kentaro Ochiai [b], Atsushi Kasugai [b]

[a] *Graduate School of Engineering, Osaka University, Osaka, Japan*

[b] *National Institutes for Quantum and Radiological Science and Technology, Aomori, Japan*

[c] *Currently: Mitsubishi Heavy Industries, Ltd., Hyogo, Japan*

An accelerator based intense neutron source, such as the international fusion materials irradiation facility (IFMIF), advanced fusion neutron source (A-FNS) and IFMIF-DEMO-oriented neutron source (DONES), produces intense high-energy neutron flux by a stripping reaction of lithium (Li) with deuteron for testing various materials for fusion reactor. In these fusion neutron sources, a liquid Li free-surface jet is applied as the liquid Li target. For the development of the Li target, the variation in the thickness of the Li jet was measured using the Li circulation facility of Osaka Univ. and the EVEDA Li test loop (ELTL). At the Osaka Univ., a lot of knowledge about the surface fluctuation of the Li jet has been obtained by using the electro-contact type distance meter (referred to as the probe apparatus herein). Besides, for the stable and safety operation of the facility, the probe apparatus is planned to be used as a safety interlock system for beam irradiation in actual intense neutron source because of its simple structure. Therefore, it is desirable to verify its accuracy. Especially, it is expected that false detections and undetected waves, which is particular to the contact-type apparatus, could cause errors. Then in this study, by comparing the result by the probe apparatus and the result by non-contact type distance meter using optical comb, the error of the results of probe apparatus is evaluated. A measurement of the variation of the jet thickness was conducted at Osaka Univ. using the laser probe method. As a result, flow characteristics such as jet thickness and wave amplitude were obtained.

Keywords: *IFMIF, neutron source, liquid lithium target, probe apparatus, laser probe*

Corresponding author: *okita@see.eng.osaka-u.ac.jp*

Atomistic Simulations for the Absorption Process of an SIA Cluster via Self-Climb in BCC-Fe

P3-062

Taira Okita [a]*, Kota Ninomiya [b], Sho Hayakawa [b], Mitsuhiro Itakura [c]

[a] *Research into Artifacts, Center for Engineering, University of Tokyo, Kashiwa, Japan*
[b] *School of Engineering, University of Tokyo, Tokyo, Japan*
[c] *Center for Computational Science & e-Systems, JAEA, Kashiwa, Japan*

Ferritic/martensitic stainless steels are candidate structural materials for fusion reactors owing to their excellent tolerance to irradiation. They are exposed to 14 MeV neutron irradiations, which create several lattice defects, namely, vacancies, self-interstitial atoms (SIAs), and their clusters. These defects diffuse and interact with other defects, leading to various mechanical property degradations in materials. Swelling, a three-dimensional volumetric change under irradiation, is one of the serious degradations of fusion reactor materials. Swelling is caused mainly by preferential absorption of SIA clusters to dislocations, which results in supersaturation and aggregation of vacancies. Hence, it is necessary to evaluate the behavior of SIA clusters to predict the swelling behavior. Further, larger SIA clusters are formed more frequently under fusion reactor conditions than under fission conditions because neutrons with higher energy are generated by the D-T reactions. This indicates that the absorption process of SIA clusters and the resultant swelling behavior under conditions of fusion are completely different from those under conditions of fission. The modeling approach for behaviors of SIA clusters is essential to predict swelling under fusion reactor conditions. In this study, we investigate the absorption process of a large SIA cluster into a dislocation line via self-climb in BCC-Fe. The micro-mechanism of the self-climb is a pipe-diffusion along the periphery of the cluster. We find that a few pipe-diffusions occurred through concurrent diffusion of several atoms, which was not considered in previous models. More importantly, the obtained results imply that the activation energy derived in the climb process is lower than the energy derived in previous studies. We will analyze the observed pipe-diffusion processes and their activation energies in detail in the presentation.

[1] Y.N.Osetsky et al., J. Nucl. Mater., 2000.

[2] D.A. Tenrentyev et al., Phys.Rev. B, 2007.

Keywords: *on-the-fly kinetic Monte Carlo, conservative climb, defect cluster, BCC-Fe*

***Corresponding author:** *okita@race.u-tokyo.ac.jp*

Study of Nuclear Responses (Transmutation, GPA, dpa) in Iron, Chromium and Tungsten for D-T Neutron Irradiation

P3-063

Mayank Rajput [a]*, S. Vala [a], PV Subhash [b], R. Srinivasan [a]

[a] *Institute for Plasma Research, HBNI, Gandhinagar, India*
[b] *ITER-India, HBNI, Gandhinagar, India*

Iron, chromium and tungsten are going to be used in upcoming fusion reactors at many places such as divertor, blanket and shield modules. DT neutron spectrum cause transmutation, gas production and displacement damage in these materials and adversely affect their strength and lifetime. In the present work, transmutation, gas production (GPA) and displacement damage (dpa) have been studied for all the stable isotopes of iron, chromium and tungsten for neutron irradiation of up to 15 MeV energy. Nuclear reaction cross section for transmutation, and gas production and energy differential cross section of PKA species for displacement damage from all the open reaction channels have been calculated with TALYS-1.8 code. Nuclear models in TALYS-1.8 code have been selected based on the comparison of calculated reaction cross section and energy spectrum of outgoing particles with the existing experimental data from EXFOR data library. Damage matrices which are required to calculate the displacement damage cross section have been calculated with NRT and Arc-dpa method. Constant parameters of arc-dpa have been derived with the result of molecular dynamics (MD) simulation of damage cascade carried out with LAMMPS code. MD simulations of displacement damage have been carried out for the native/self PKA in iron, chromium and tungsten at up to 200 KeV damage energies. Gas production and displacement cross section have been calculated at up to 15 MeV neutron energy and used to calculate the dpa and GPA in tungsten for ITER and EU Demo neutron spectrum. Values of GPA comes out to be 14.2 appm/FPY (helium production) and 56.8 appm/FPY (Hydrogen production) for ITER neutron spectrum and 37.3 appm/FPY (helium production) and 56.8 appm/FPY (Hydrogen production) for European demo neutron spectrum respectively at the first wall. Similarly, dpa values comes out to be 3.3 dpa/FPY and 8.6 dpa/FPY for the ITER and EU demo neutron spectrum at first wall. Similar calculation of GPA and dpa along with the study of transmutation have been carried out for iron and chromium.

Keywords: *Displacement damage, gas production, transmutation, D-T neutrons, MD simulations*

***Corresponding author:** *mayank.rajput@ipr.res.in, rajputmayank18@yahoo.in*

Material Strength Standard of F82H for RCC-MRx

P3-064

Hideo Sakasegawa *, Hiroyasu Tanigawa, Takanori Hirose, Taichiro Kato

National Institutes for Quantum and Radiological Science and Technology, Rokkasho, Aomori, Japan

Japanese RAFM (reduced-activation ferritic/martensitic), F82H (Fe-8Cr-2W, V, Ta), has been developed since the early 1990s. Through international collaborative researches such as IEA (International Energy Agency) round-robin test, HFIR (High Flux Isotope Reactor in Oak Ridge National laboratory) Japan-US collaboration test, and Japan-EU BA (Broader Approach) activity, we have extensively accumulated both non-irradiated and irradiated material properties of F82H. It demonstrates appropriate material properties for the application to TBM (test blanket module) as a milestone of DEMO blanket module. At present, material strength standard has to be developed for the design activities of blanket. Material strength standard gives average, minimum, and design values and equations complying to the definition in construction codes such as RCC-MRx and ASME. In particular, since RCC-MRx has been well studied for ITER component, conformance of F82H to RCC-MRx should be studied for develop design limit of F82H. In this work, we summarized and analyzed the data of tensile, creep, and fatigue properties, and then calculated average, minimum, and design values. These values were compared to those of European RAFM, Eurofer (X10CrWVTa9-1), and Mod. 9Cr-1Mo steel (X10CrMoVNb9-1) qualified in RCC-MRx. Though some differences were observed in minimum and average values of yield and tensile strengths due to differences in chemical compositions, significant differences were not observed in design values of S_m and S that can be calculated from minimum yield and tensile strengths. In this presentation, creep and fatigue properties were also summarized and analyzed to obtain other design values such as S_t and develop the material strength standard of F82H for RCC-MRx.

Keywords: *reduced-activation ferritic/martensitic (RAFM) steel, material strength standard, TBM (test blanket module), RCC-MRx*

**Corresponding author: sakasegawa.hideo@qst.go.jp*

Fabrication and Characterization of the 316L(N)-IG ESR Forging Block and Non-ESR Rolling Plate for the ITER Blanket Shield Block

P3-065

Hee-Jin Shim *, **Sawoong Kim**, Byeong-Il Park, Yonghee Lee

National Fusion Research Institute, Blanket Technology Team, Daejeon, Korea

The austenitic stainless steel is an essential structural material and well used because of its corrosion resistance and ability to withstand both high and low temperatures. Especially, 316L(N)-IG stainless steel is mainly employed for the structural material of the core machine considered for the in-vessel components of the ITER, which require a good resistance to corrosion, weld-ability and mechanical properties at elevated temperature.

This paper deals with the manufacturing process and characterization of the 316L(N)-IG stainless steel taking into account the specific working conditions and requirements. This stainless steel fabricated by forging and ESR (Electro-Slag Remelting) block and hot rolling for non-ESR plate, respectively. Moreover, this study is provide the ultrasonic testing results including welded material between forging block and hot rolled plate, and discuss the those results.

Keywords: *316L(N)-IG, Electro-Slag Remelting, Forging, Hot-rolled, Ultrasonic Test*

**Corresponding author: shimj@nfri.re.kr*

The Tests of the Rohacell 71HF - a Candidate Material for the SIC-2 Windows for the ITER HFS Reflectometry

P3-066

Alexander V. Spitsyn [a]*, Dmitry A. Shelukhin [a, b], Dmitry I. Cherkez [a], Nikolay P. Bobyr [a], Vladimir A. Vershkov [a, b], Artem A. Mednikov [a], Dmitry A. Kozlov [a]

[a] *NRC "Kurchatov Institute, Moscow, Russia*
[b] *International Fusion Projects Coordinating Centre (Fusion Centre)*

The safety concept of the ITER requires the additional safety barrier windows that prevent air and tritium exchange between different zoned in tokamak building (safety important class 2). Beside the zone separation the material should bring minor influence on diagnostic performance.

A material for such barrier must demonstrate a very low permeation at elevated temperatures, be able to withstand the accidental pressure

gradient and be transparent to mm-scale microwaves. It is proposed to use rigid closed-cell polymethacrylamide foam (Rohacell 71HF) as a material for the SIC-2 window. The Rohacell 71HF samples were tested: helium, nitrogen and deuterium permeation through the sample into vacuum; nitrogen and deuterium sorption and desorption by the Rohacell 71HF sample; and main microwave parameter (dielectric permittivity and loss tangent) in frequency band 18-170 GHz. The permeation of the Rohacell 71HF was measured in the temperature range from 300 to 430 K during up to 94 h. It was shown that permeation curve of deuterium and helium could be described in terms of a diffusion-limited regime (DLR) and the diffusion coefficients were estimated as $1.4 \cdot 10^{-10}$ and $4.3 \cdot 10^{-10}$ m^2/s respectively for the membrane temperature of 400 K. It was not detected permeation of nitrogen through the sample. The maximum deuterium sorption from D_2 gas at 10^5 Pa was $1.7 \cdot 10^{20}$ molecD_2/g at temperature 300–400 K. The microwave properties was measured with ABmm microwave network analyzer. It was found that dielectric permittivity is constant within error bar along the whole measurements band and equal to 1.0867 ± 0.0067 whereas loss tangent is equal to $(2.007 \pm 0.172) \cdot 10^{-3}$.

Thus, it was proven that the Rohacell 71HF can be considered as SIC-2 safety barrier material for the ITER reflectometry transmission lines.

Keywords: *SIC-2 microwave window, HFS, Rohacell 71HF, gas permeation, gas sorption*

***Corresponding author:** *spitsyn_av@nrcki.ru*

Irradiation of Optical Materials in BRR

P3-067

Ildikó Szenthe [a]*, Ferenc Gillemot [a], Rafael Vila [b]

[a] *Hungarian Academy of Sciences, Centre for Energy Research, KFKI Campus, Budapest, Hungary*

[b] *Fusion National Lab, CIEMAT, Madrid, Spain*

The next step in the European fusion program after the building ITER is the designing and building of a demonstration fusion power unit named DEMO. DEMO needs radiation resistant functional (optical and dielectric) materials. Long practice of neutron and gamma irradiation of structural materials exists, but irradiation study of optical materials is not usual. Eurofusion initiated a subproject to study the degradation of the optical properties of different ceramic materials by neutron irradiation. The optical elements are thin and fragile, their surfaces are sensitive for scratch or abrasion, and they can't be encapsulated and treated like metallic samples. In two campaigns 31+24 samples have been irradiated with 4 different irradiation doses in the Budapest Research Reactor of the Energy Research Centre of the Hungarian Academy of Science. The irradiated materials were different grade silica, alumina, spinel, YAG, sapphires and diamonds. The paper introduces the difficulties during preparation, irradiation, encapsulation and transportation of these samples to the testing laboratories.

Keywords: *irradiation, optical materials*

***Corresponding author:** *szenthe.ildiko@energia.mta.hu*

Leak Tight Joint Method for ODS-Cu/ODS-Cu by Application of the Advanced Brazing Technique

P3-068

Masayuki Tokitani [a], Y. Hamaji [a], Y. Hiraoka [b], S. Masuzaki [a], H. Tamura [a], H. Noto [a], T. Tanaka [a], T. Muroga [a], A. Sagara [a] and the FFHR Design Group [a]

[a] *National Institute for Fusion Science, National Institutes of Natural Sciences, Toki, Gifu, Japan*

[b] *Department of Applied Physics, Okayama University of Science, Okayama, Japan*

The oxide dispersion strengthened copper alloy (ODS-Cu) is one of the candidate materials for divertor heat sink in a fusion reactor. In this study, the joint of the ODS-Cu (ODS-Cu/ODS-Cu) was prepared. The applied ODS-Cu was GlidCop® (Cu-0.3 wt%Al_2O_3). The joint procedures were based on the brazing technique between GlidCop® and pure tungsten (W) with BNi-6 (Ni-11%P) filler material, which technique was developed in our previous work [1]. This brazing technique was performed without compressive load and without any intermediate layer, and was named as the “advanced brazing technique”. In this study, first, the joint GlidCop®/GlidCop® was obtained without any compressive load by application of the advanced brazing technique. The joint was subjected to the three-point bending test and the Vickers hardness test. The joint was also subjected to the microstructure observation and the leak tightness test. The yield strength of the joint was approximately one-half of that of the GlidCop® bulk, and a softened region with the thickness of ~780 μm was recognized. In addition, the joint was not leak tightness against fluids.

Next, therefore, the joint was prepared with a compressive load of ~0.54 MPa which was applied in the direction perpendicular to the joint interface during the heat treatment phase of the brazing. Consequently, the yield strength of the joint was improved almost as high as that of the GlidCop® bulk, and the softened region was much narrower (~180 μm) compared with the case without the compressive load. Furthermore, the joint achieved leak tightness against fluids. The special feature of this joint method is that a limited volume of the GlidCop® surfaces near the bonding interface are melted during the bonding heat treatment phase at 960 °C due to the eutectic reaction of the Cu-P system. Then, after finalizing the solidification, a tight bonding layer can be created without any cavities and cracks. This phenomenon seems to resemble a micro scale welding. In summary, the present brazing technique can promise fabrication of a complex shaped fluids flow path system with leak tightness. This is a large advantage for fabricating the high heat flux component in the fusion reactor, such as a W divertor with ODS-Cu heat sink. We have already produced the prototype component.

[1] M. Tokitani et al., Nucl. Fusion 57 (2017) 076009

Keywords: *brazing, ODS-Cu, bend test, hardness test, leaking test*

***Corresponding author:** *tokitani.masayuki@LHD.nifs.ac.jp*

Tungsten Fibre-Reinforced Copper as an Advanced Heat Sink Material for Highly Heat Loaded Plasma-Facing Components

P3-069

Alexander v. Müller [a,b,*], B. Böswirth [a], V. Cerri [c], D. Ewert [d], H. Greune r[a], R. Neu [a,b], U. Siefken [e], E. Visca [c], J.H. You [a]

[a] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[b] *Technische Universität München, Garching, Germany*
[c] *ENEA, Frascati RM, Italy*
[d] *DITF, Denkendorf, Germany*
[e] *Louis Renner GmbH, Bergkirchen, Germany*

Currently, copper (Cu) alloys are considered as state-of-the-art heat sink materials for highly loaded plasma-facing components (PFCs) in magnetic confinement fusion devices. The most widely used material in this respect is the precipitation hardened alloy CuCrZr which is also applied within the ITER divertor target tungsten monoblock design. However, the use of Cu alloys with respect to a DEMO reactor environment implies design engineering risks. These risks are especially associated with property degradation under reactor relevant operating conditions resulting in a small potential operating temperature window where these materials retain good ductility, fracture toughness and resistance to creep deformation. Against this background, tungsten fibre-reinforced copper (W_f-Cu) is being developed as an advanced PFC heat sink material. The reinforcing phase applied within this material is composed of high-strength drawn tungsten fibres (diameter down to 20 µm, tensile strength at room temperature up to 4 GPa) which can be used to both enhance the mechanical properties and reduce the macroscopic thermal expansion of the composite. The contribution will present results regarding the R&D work performed during recent years with respect to W_f-Cu for divertor PFC applications. The chosen industrially viable manufacturing route for these composite materials which comprises fabrication of fibrous preforms and subsequent liquid Cu infiltration will be described in detail. Moreover, results will be presented from basic material characterisation through to successful application of W_f-Cu to a number of PFC mock-ups. In this respect, results will be presented regarding the high-heat-flux testing of these mock-ups at DEMO relevant hot water cooling conditions within the neutral beam facility GLADIS (IPP Garching) where heat loads up to 32 MW/m^2 were applied without mock-up failure.

Keywords: *tungsten, copper, composite, fibre-reinforced, plasma-facing component*

***Corresponding author:** *alexander.v.mueller@ipp.mpg.de*

Advances in Radiation Hardness Testing of Optical Windows for DEMO

P3-070

Dario Cruz, Rafael Vila *

Lab. Nacional de Fusion- CIEMAT, Madrid Spain

EUROfusion R&D on optical windows and lenses (WP-MAT-FM) is focused on the evaluation of neutron radiation hardness of candidate materials at DEMO relevant doses. These materials have been chosen from those previously studied at lower doses and with good radiation tolerance plus a few compositions that are promising but with insufficient data. Thus, filling the gap in terms of neutron doses and material scanning. From the 3 neutron doses (0.1, 0.4 and 1 dpa), samples from the two lower doses with reduced activation levels low enough to be measured include several grades of sapphire, spinel, and amorphous silica, including some with AR coatings. Some others with higher activation have been yet measured (YAG, BaF_2 and ZnS)

Important consequences for optical windows selection have been obtained. Results are separated in UV, VIS-NIR and IR regions. Whereas UV region is always strongly degraded in the studied materials, there are some quite good solutions for VIS and IR regions, although with different materials for each case. Samples of CaF_2 have revealed a very important degradation even in terms of structural damage.

Keywords: *Materials, Insulators, Diagnostics, Radiation effects, Neutron damage, Al_2O_3*

***Corresponding author:** *rafael.vila@ciemat.es*

Hydrogen Retention Behavior of Primary Precipitates in F82H Steel: Atomistic Calculation Based on the Density Functional Theory

P3-071

Yoshiyuki Watanabe [a]*, Hirotomo Iwakiri [b], Daiji Kato [c], Dai Hamaguchi [a], Hiroyasu Tanigawa[a]

[a] *National Institutes for Quantum and Radiological Science & Technology, Aomori, Japan*
[b] *University of the Ryukyus, Okinawa, Japan*
[c] *National Institute for Fusion Science, Gifu, Japan*

RAFM steel is expected to be a blanket structural material in nuclear fusion reactor. In the blanket material a large amount of helium and hydrogen are produced by nuclear transmutation reaction under fusion neutron irradiation. As well as helium, hydrogen might strongly interact with other lattice defects and enhance the irradiation effects such as embrittlement and swelling. A recent experiment for hydrogen retention in F82H steel by thermal desorption spectroscopy after deuterium-ion irradiation shows the primary precipitates of metal carbide such as $M_{23}C_6$ and MX might play a role of remarkable trap site for hydrogen isotopes; however the trapping and diffusion behaviour of hydrogen isotopes is not understood well. The purpose of this paper is to theoretically understand fundamental behaviour of hydrogen in the metal carbide by mean of atomistic calculations based on the density functional theory (DFT), for a practical design of fusion DEMO blanket.

DFT calculations with the SIESTA code were conducted to investigate the energetics of hydrogen atoms in $Cr_{23}C_6$ and TaC crystal (the main components of $M_{23}C_6$ and MX precipitates) in F82H steel. The calculation results says that the hydrogen solution energy in $C_{r23}C_6$ is -0.48 eV with the most relax configuration of trigonal bipyramidal (TB)-site, while the solution energy in TaC is +0.94 eV with tetrahedral (T)-site. Taking into account the hydrogen solution energy of +0.23 eV in Fe (the main component of matrix in F82H steel), the these solution energies indicate that $M_{23}C_6$ precipitates in F82H steel might be a remarkable trap site for hydrogen, but MX precipitates might NOT. And both the TB-site and T-site can be interpreted by electric interaction between atoms. With the obtained solution energies, an attempt for estimation of hydrogen retention in F82H steel was done assuming the temperature of 0 K, where the maximum amount of hydrogen retention in F82H steel was estimated to be 190 ppm for $M_{23}C_6$ precipices and 9 ppm for MX precipitates. Hydrogen retention at finite temperature will be also discussed.

Keywords: *Hydrogen retention, RAFM, precipitate, DFT calculation*

Corresponding author: *watanabe.yoshiyuki@qst.go.jp*

Fracture Toughness Evaluation of Neutron Irradiated Eurofer97 Variants Using Miniature Bend Bars

P3-072

Xiang Chen [a]*, Arunodaya Bhattacharya [a], Tim Graening [b], Anne A. Campbell [a], Logan N. Clowers [a], Mikhail A. Sokolov [a], Josina W. Geringer [a], Yutai Katoh [a], Michael Rieth [b]

[a] *Oak Ridge National Laboratory, Oak Ridge, TN, USA*
[b] *Karlsruhe Institute of Technology, Karlsruhe, Baden-Württemberg, Germany*

Eurofer97 is the European reference material for the first wall and blanket of a DEMO fusion reactor. During the fusion plant operation, neutron irradiation causes irradiation embrittlement and reduces materials fracture toughness. Therefore, it is critical to understand the irradiation embrittlement response in Eurofer97. Under the framework of EUROfusion, we will present the fracture toughness evaluation of ten neutron irradiated Eurofer97 steel variants with varying compositions and heat treatment conditions using miniature multi-notch bend bar specimens. The neutron irradiation was performed at the Oak Ridge National Laboratory High Flux Isotope Reactor, with the nominal irradiation temperature of 300°C and an irradiation dose of 2.5 displacements per atom. Initial irradiation temperature characterization based on passive SiC thermometry measurements indicated eight materials experienced irradiation temperatures in the range of 259 – 295 °C whereas two materials experienced much higher irradiation temperature of 476 °C. We will characterize materials Master Curve fracture toughness and determine the shift in the transition temperature, T_0, after neutron irradiation. In addition, we will correlate materials fracture toughness with tensile, Vickers microhardness, and microstructures to elucidate structure-irradiation response-property change relationships.

Research sponsored by the U.S. Department of Energy, Office of Fusion Energy Sciences, under contract DE-AC05-00OR22725 with UT-Battelle, LLC and by Karlsruhe Institute of Technology under contract NFE-1606240 with UT Battelle, LLC. A portion of this research at ORNL's High Flux Isotope Reactor was sponsored by the Scientific User Facilities Division, Office of Basic Energy Sciences, US Department of Energy.

Keywords: *fracture toughness, Master Curve, Eurofer97, small specimen testing technique, EUROfusion*

Corresponding author: *chenx2@ornl.gov*

Deuterium Retention in Plasma-Implanted W with Various Damage Distributions

P3-073

Shota Yamazaki [a]*, Akihiro Togari [a], Moeko Nakata [a], Ayaka Koike [a], Takuro Wada [a], Mingzhong Zhao [a], Fei Sun [b], Tatsuya Kuwabara [c], Yuji Hatano [d], Takeshi Toyama [e], Noriyasu Ohno [c], Yasuhisa Oya [a]

[a] *Graduate School of Integrated Science and Technology, Shizuoka University, Shizuoka, Japan*
[b] *Faculty of Science, Shizuoka University, Shizuoka, Japan*
[c] *Graduate School of Engineering, Nagoya University, Nagoya, Japan*
[d] *Hydrogen Isotope Research Center, University of Toyama, Toyama, Japan*
[e] *Institute for Materials Research, Tohoku University, Ibaraki, Japan*

Tungsten (W) is considered as a leading candidate for plasma facing materials (PFMs) in future fusion reactors. During the operation, irradiation defects will be introduced in W uniformly by 14 MeV neutrons. In contrast, damages induced by ions or charge-exchange particles will have a peak at a certain depth. In this study, both of neutron and Fe ion irradiations were performed for W and D trapping was studied by thermal desorption spectroscopy (TDS) to understand the influence of damage distribution on D retention.

Firstly, 14 MeV neutron irradiation was performed up to damage level of 2.4×10^{-7} - 6.3×10^{-4} dpa (displacement per atom). Thereafter, 6 MeV Fe^{2+} ions were irradiated into these samples up to 0.1 or 1 dpa. Both irradiations were conducted at room temperature (R.T.). After implantation of 1.0 keV D ions (flux: 1.0×10^{18} D^+ m^{-2} s^{-1}, fluence: 1.0×10^{22} D^+ m^{-2}, R.T.) or 100 eV D plasma (flux: 1.0×10^{21} D^+ m^{-2} s^{-1}, fluence: 5.0×10^{24} D^+ m^{-2}, 373 K) , D trapping was examined by TDS with heating rate of 0.5 K s^{-1} up to 1173 K. Samples irradiated solely with Fe ions were also examined for comparison.

For both D ion- and plasma-implanted W, larger D desorption was found for the samples damaged solely with Fe^{2+} ions. The combination of neutron and Fe^{2+} irradiations resulted in the reduction in D retention, especially the trapping of D by vacancy clusters and voids. D trapping by voids was hardly found after D plasma exposure. The combination of damage distributions by 14 MeV neutron and Fe^{2+} may enhance the D reemission under implantation from the surface.

Keywords: *Tungsten, 14 MeV neutron, plasma exposure, heavy ion irradiation, Deuterium retention*

Corresponding author: *yamazaki.shota.15@shizuoka.ac.jp*

Mechanical Properties and Microstructure of Three Kinds of ITER-Grade Pure Tungsten with Different Manufacturing Conditions

P3-074

Ju-Hyeon Yu *, Hiroyasu Tanigawa, Dai Hamaguchi, Takashi Nozawa

National Institutes for Quantum and Radiological Science and Technology, Aomori, Japan

The tungsten has been considered as armor material for a fusion reactor because of its high melting point and high thermal conductivity. Technical issues on the use of tungsten in the fusion DEMO reactor divertor is its use condition, where a wide temperature range (350 to 1100°C) and 14 MeV fusion neutron irradiation up to10dpa are expected. A large data dispersion of the pure tungsten tensile strength has been reported especially below 500°C as it behaves as a brittle material at that temperature range, and that makes it difficult to define tungsten property for structural design activity and to define neutron irradiation effects. Therefore, it is essential to supply tungsten with reliable and less disperse property to apply tungsten for fusion DEMO reactor design activity. In this study, we have produced pure tungsten plates with three different manufacturing processes, assuming that those property data dispersion are caused by the flaw or defects induced during the manufacturing process. The first plate (ID: IGW) is made by the same fabrication conditions which are used to procure the ITER divertor tungsten. The second plate (ID: CLW) is made by the same condition as IGW, but cross rolled to the plate. The third plate (ID: CHW) is made by the same condition as CLW but with higher rolling reduction ratio.

Vickers hardness testing and tensile testing were conducted as the evaluation of the basic mechanical property of plates. The Vickers hardness test conditions are 2.5 kgf load, 10 s holding time, measured at 1 mm intervals to check the difference in a thickness direction. Tensile tests were performed at room temperature with a displacement rate of 0.03 mm/min, using the SS-J3 specimen. As a result, hardness showed no significant differences in thickness wise and in between three plates with a small data dispersion. However, the tensile strength showed a larger data dispersion between plates and tensile direction. Preliminary microstructure analyses suggest that imperfectly sintered tungsten grain boundaries could cause those observed data dispersions. The more details including microstructure and fracture toughness evaluation will be reported.

Keywords: *Pure Tungsten, Mechanical Properties, Microstructure3, ITER-Grade, SS-J3*

Corresponding author: *yu.juhyeon@qst.go.jp*

Thermal Interlayers for ITER - Development and Measurements

P3-075

Christoph Zauner [a]*, Hans Meister [b], Florian Penzel [b]

[a] *KRP Mechatec GmbH, Garching, Germany*

[b] *Max Planck Institute for Plasmaphysics, Garching, Germany*

Heat transfer through mechanical interfaces is crucial to keep temperatures of thermally loaded parts low by allowing good heat flow to the heat sink. Thermal interlayers are typically used to improve the thermal performance. In the frame of the ITER bolometer camera development various types of interlayers have been designed by FEM and investigated through experiments. The goal was to identify designs and materials having a high heat thermal conductivity (HTC, being the proportionality constant between the heat flux and the temperature difference) under the required and specific boundary conditions (materials in contact, contact pressure, surface quality, gap size, and gap variation). This HTC value is often the most critical parameter for the systems thermal model and its performance and cannot be derived from literature as values vary between 200 W/(m2K) and 20000 W/(m2K).

For the installation of the bolometer camera to the ITER vacuum vessel the following types of interlayers have been designed and tested:

- Copper foam (test only)
- Corrugated copper sheet (FEM, design and test)
- Solid copper sheet with cutout bending beams (FEM, design and test)

A further test campaign is prepared to identify the performance of MULTILAM ©, which is typically used for providing good electrical contact through many contact points for uneven surfaces.

A dedicated thermal vacuum test setup has been used to apply and measure a heat flow though the respective interface materials and thermal interlayer and to measure the temperature step within the thermal interlayer. Thermal cycles are included in the test campaign to simulate the ITER bake out and to verify the mechanical stability of the interlayers under this thermoelastic loads.

Analysis and test results are presented and a selection guide for the various types of interlayers is proposed.

Keywords: *thermal interlayer, thermal contact conductivity, design, measurement*

Corresponding author: *zauner@krp-m.de*

Application of Ti-doped MoS2 Low Friction /Anti Seize Coating for ITER First Wall

P3-076

Yi Zhou, Jiming Chen *, Pinghuai Wang, Min Dan, Fanya Jin

Southwestern Institute of Physics, Sichuan Chengdu, China

Low friction /anti seize coating required on some components of ITER first wall to reduce the operational loads, minimize fretting caused by impacts, and prevent seizing due to high pressure on contact surfaces under vacuum and high temperature. Properties of coating used should be studied to ensure effectiveness during service, especially the tribological properties in various environment simulating working condition, such as atmosphere, RT vacuum and high temperature vacuum. MoS2 coatings with different Ti content were deposited by magnetron sputtering system on aluminium bronze, the material of pad in ITER first wall. Morphology, friction/wear properties and structure were respectively investigated by SEM/optical microscope, ball on disk tribometer and XRD, besides, humidity resistance ability were investigated because of the unavoidable exposure to air during assembling for a considerable time. Results showed that with the increase of Ti content from0% to 23%, density of coatings increased gradually and coating structure changed from high crystallinity to amorphous forms. Friction and wear test showed that with the increase of Ti content, tribological property first improved and then degraded, 3%Ti coating showed the best properties, which had as low as 0.015 of friction coefficient at RT in vacuum, much lower than the requirement of ITER. Coating doped with 23%Ti showed no lubrication. Elevated temperature(400℃) brought higher friction coefficient increased from 0.015~0.04 at RT to 0.07~0.1 and damage of substrate was observed in pure and 13%Ti coatings. Results showed that doped-Ti could protect coating and substrate from corrosion caused by moisture and oxygen effectively.

Keywords: *ITER first wall, magnetron sputtering, MoS2-Ti, low friction/anti seize coating, tribological property, humidity resistance*

Corresponding author: *chenjm@swip.ac.cn*

Modelling Ion Irradiation and Slip Localisation in Ferritic-Martensitic Steels: The Fusion-Fission Cross-Cutting M4F Project

P3-077

Lorenzo Malerba [a]*, María J. Caturla [b], Ermile Gaganidze [c], Cornelia Heintze [d], Milan Konstantinovic [e], Pär Olsson [f], David Rodney [g], Ana M. Ruiz [h], Marta Serrano [a], François Willaime [i]

[a] *CIEMAT, Division of Energy Materials, Madrid, Spain*
[b] *University of Alicante, Alicante, Spain*
[c] *Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany*
[d] *Helmholtz Zentrum Dresden-Rossendorf, Dresden, Germany*
[e] *SCK●CEN, Institute of Nuclear Materials Science, Mol, Belgium*
[f] *KTH-Royal Institute of Technology, Stockholm, Sweden*
[g] *University of Lyon, Villeurbanne, France*
[h] *Joint Research Centre, Petten, The Netherlands*
[i] *CEA-Centre de Saclay, Université Paris-Saclay, Gif-sur-Yvette, France*

The effects of neutron irradiation on materials' behaviour affect both fission and fusion nuclear reactor technologies. In both, ferritic/martensitic steels have been selected for key reactor components. However, for this class of materials design rules need to be developed. For example, F/M steels suffer from severe loss of uniform elongation at doses in excess of ~1-2 dpa, when irradiated below 350-400 °C, that needs to be accounted for in design codes. Although qualitatively understood, to date no model exists that accounts for this process at the level of continuum mechanics. This calls for a multiscale modelling approach to link scales, correlating microstructural changes with slip localization and describing the consequences of the latter on the macroscopic mechanical behaviour.

Experimental data on loss of uniform elongation come from testing neutron irradiated specimens. Charged particle irradiation could provide a wealth of additional data, provided that the issue of data transferability from ion to neutron irradiation is solved. Also, it is not trivial to deduce information on the mechanical behaviour of materials after irradiation from ion irradiation experiments. These issues can also be addressed using modern multiscale modelling tools, while developing protocols for mechanical measurements on small quantities of material, e.g. nanoindentation.

The Euratom-funded M4F project (Multiscale Modeling for Fusion and Fission Materials) addresses these issues. This presentation reports on the specific objectives and first results of this project, highlighting the aspects of model development and integration, and emphasizing the commonalities between fusion and fission applications.

Keywords: *cross-cutting, plastic localisation, ion irradiation*

Corresponding author: *lorenzo.malerba@ciemat.es*

FEM Analyses of the ITER EC H&CD Torus Diamond Window Unit Towards the Prototyping Activity

P3-078

Gaetano Aiello [a]*, N. Casal [b], P. Estébanez [c], V. Gràcia [c], M. Henderson [b], A. Meier [a], G. Saibene[c], T. Scherer [a], S. Schreck [a], D. Strauss [a]

[a] *KIT IAM, Eggenstein-Leopoldshafen, Germany*
[b] *ITER Organization, St. Paul Lez Durance Cedex, France*
[c] *F4E Joint Undertaking, Barcelona, Spain*

The chemical vapour deposition (CVD) diamond torus window unit is a sub-component of the ITER Electron Cyclotron Heating and Current Drive (EC H&CD) system used for a diverse range of applications including plasma heating and control of plasma magneto-hydrodynamic (MHD) instabilities. It consists of ultra-low loss polycrystalline diamond disk brazed to copper cuffs and then enclosed by a metallic structure. The diamond disk with 1.11 mm thickness already passed successfully the ITER final design review (FDR) in December 2017. In view of the complete window assembly FDR, prototyping activities of the window are essential and, therefore, they shall start in 2019 in order to check the feasibility of the proposed manufacturing and assembling sequence of the component. In this perspective, as the design of the systems surrounding the window is currently in development phase, the paper describes the FEM analyses of the window carried out to prove the soundness of the design used for the prototyping and also to define requirements for the surrounding systems. Specific methodologies were adopted such as the limit analysis approach used for the external loads acting on the window unit. Several combinations of forces and moments were applied to the window unit to find the maximum loads, i.e. the limits loads, that generate stresses in the unit equal to the allowable ones, according to the selected design criteria.

Keywords: *ITER, EC system, Diamond window unit, FEM analyses, Limit analysis*

Corresponding author: *gaetano.aiello@kit.edu*

Towards Large Area CVD Diamond Disks for Brewster-Angle Windows

P3-079

G. Aiello [a]*, Sabine Schreck [a], K.A. Avramidis [b], T. Franke [c,d], G. Gantenbein [b], J. Jelonnek [b], A. Meier [a], T. Scherer [a], D. Strauss [a], M. Thumm [b], M.Q. Tran [e], C. Wild [f], E. Woerner [f]

[a] *IAM, KIT Eggenstein-Leopoldshafen, Germany*
[b] *IHM, KIT, Eggenstein-Leopoldshafen, Germany*
[c] *EUROfusion Consortium, Garching, Germany*
[d] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[e] *Swiss Plasma Center (SPC), EPFL, Lausanne, Switzerland*
[f] *Diamond Materials GmbH, Freiburg, Germany*

In the frame of the EUROfusion Work Package Heating and Current Drive (WP HCD) of the Power Plant Physics and Technology (PPPT) program, CVD diamond disk Brewster-angle windows for gyrotron operation at multi-megawatt RF power levels and long pulses are under development. These windows allow for frequency step-tunable operation. The Brewster-angle of 67.2° for diamond leads to an elliptical connection of the disk to the copper waveguides (WGs), requiring an advanced joining process. For proper transmission of the RF power, the disk consists of low loss CVD diamond of optical grade. The current target for the WG aperture of DEMO is 63.5 mm. It allows for an RF power transmission of 2 MW, but it requires a disk diameter of 180 mm for the 67.2° angle. In addition, a thickness of approximately 2 mm is needed to achieve the proper mechanical stability. State of the art microwave plasma reactors are not capable of growing disks of such size. The maximum available diameter of a CVD diamond disk suited to microwave applications is currently 140 mm. Thus, the industrial partner Diamond Materials GmbH (Freiburg, Germany) is doing extensive diamond growth experiments. A first of its kind, 180 mm thermal grade, crack-free, diamond disk was produced in the microwave plasma reactor with an average unpolished thickness of about 2 mm. First loss tangent measurements have been also performed. This presentation describes the steps and the first results of this non-straightforward path, a challenging new field for diamond manufacturers and a major breakthrough for future frequency step-tunable operation.

Keywords: *DEMO, EC system, Brewster diamond window, Diamond growth*

Corresponding author: *gaetano.aiello@kit.edu*

LIBS Measurements Inside the FTU Vessel Mock-Up by Using a Robotic

P3-080

Salvatore Almaviva [a]*, L. Caneve [a], F. Colao [a], V. Lazic [a], G. Maddaluno [a], P. Mosetti [a], A. Palucci [a], A. Reale [a], P. Gasior [b], W. Gromelski [b], M. Kubkowska[b]

[a] *ENEA FSN C.R Frascati, Frascati, Italy*
[b] *IPPLM Institute of Plasma Physics and Laser Microfusion, Warsaw, Poland*

One of the major concern with the exploitation of ITER is the amount of tritium (T) that can be retained in plasma facing components (PFCs) and in-vessel structures: beyond a limit fixed by safety issues, the machine operation must be stopped and T removed. Laser-Induced-BreakdownSpectroscopy (LIBS) is one of the eligible technique considered for the measurement of T retained in ITER vessel. LIBS induces a small plasma on the PFCs surfaces that can be spectrally analyzed, to detect and quantify the PFCs constituents and contaminants. In the framework of the EUROfusion WPMST2 a new, compact LIBS system has been designed and mounted on the Frascati Tokamak Upgrade (FTU) robotic arm and the feasibility of LIBS measurements has been explored by using the mock-up of a quarter of the FTU stainless steel vacuum vessel. The LIBS system includes a Double Pulse laser and optics for launching the laser beam and detecting the LIBS plasma. It is capable to reach, at a given toroidal position, every point of the vacuum vessel section. In this work, the new LIBS system is described, and measurements carried out on ITER-relevant samples, placed in the FTU mock-up, are shown. The samples, composed of an Al layer implanted with deuterium (D) as a proxy for T, reproduce the expected main mechanism of T retention in ITER, the co-deposition of Be with T (Be replaced by Al in the samples). The D▯ emission line (656.1 nm), as well as all the other optical lines emitted by the LIBS plasma, has been monitored in detail, with air and Ar as background gases. Based on this positive results further LIBS measurements are now foreseen to be carried out on the FTU tokamak at the end of the 2019 experimental campaign.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *LIBS, PFCs, tritium detection, ITER, FTU robotic arm*

Corresponding author: *salvatore.almaviva@enea.it*

Preliminary Analysis on the Thermal Shield of DTT

P3-081

Gianluca Barone *, Selanna Roccella, Emanuela Martelli, Eliseo Visca

Department of Fusion and Technology for Nuclear Safety and Security – ENEA, Italy

The thermal shield (TS) in DTT is devoted to minimize the heat loads from the tokamak warm components, to the superconducting magnets operating at 4.5 K. The TS is subdivided into three main regions covering respectively the vacuum vessel (VTS), the ports (PTS) and the cryostat (CTS). Along the toroidal direction, the TS is arranged in 18 electrically insulated segments (20° each), composed of several cooling modules. Each module consists of a double wall panel, 20 mm thick, enclosing the cooling tubes where the pressurized helium gas circulates (1.8 MPa, ~ 80-100 K). The present work provides a preliminary analysis of the major parameters involved in the design of the TS system. More specifically, heat loads (mainly radiative) to the TS were assessed for normal operation and backing. Consequently, based on the magnets temperature requirements, the performance of a representative module was investigated for different design options, providing, in each case, the resulting power transferred to the superconducting magnets. The investigation involved the thermo-hydraulic characterization of the TS cooling system (e.g. helium flow, pressure losses, temperature difference, heat flux, etc.) integrated in the double wall stainless-steel structure to which it is attached. The analyses were performed employing the CFD code ANSYS CFX (with radiation option) to simulate the behaviour of a single TS modules for different cooling tubes layouts. Furthermore, the thermal hydraulic system code RELAP5-3D© was adopted to simulate the entire helium cooling system. The simulations' outcomes provided, inter alia, valuable suggestions concerning the optimization of the cooling tubes layout within each TS module, as well as indications on the optimal cooling tubes route along the whole TS sector.

Keywords: *Thermal shield, cooling tube, DTT*

****Corresponding author:*** *gianluca.barone@enea.it*

Analyses and Design of the Wendelstein 7-X Port Bellows Protection

P3-082

André Carls*, J. Zhu, V. Bykov, B. Missal and the W7-X team

Max-Planck-Institut für Plasmaphysik, Teilinstitut Greifswald, Member of the Euratom Association, Greifswald, Germany

The Wendelstein 7-X (W7-X) bellow protection have become a prominent part of the radiation load shielding system of the world's largest modular stellarator, which is in preparation for long-pulse operation at the Max-Planck-Institute for Plasma Physics in Greifswald, Germany. Its purpose is the shielding of the port bellows, a flexible piece, connecting the inner and the outer part of the port tube, from ECRH-stray radiation. Since the bellows are only thin walled flexible stainless steel structures, they are significantly loaded through long-time ECRH stray radiation exposition during plasma operation, as the ECRH-stray radiation is present in every port, not completely closed on its plasma facing side.

The paper presents an the multiphysics analyses in support of the bellows protection design and development, as well as the final measures taken, to protect the bellows from the ECRH-stray radiation. The effectiveness of the shielding concept, based on a tubular Cu-shield, tightened with springs is proven and its robustness against effects, like geometric distortion and induced eddy currents is shown.

Keywords: *In-vessel components, ports, bellows, ECRH stray-radiation, finite element analyses*

****Corresponding author:*** *andre.carls@ipp.mpg.de*

Neutronics Related Integration Studies of EU-DEMO Pellet Injection System

P3-083

Andrea Colangeli [a,b,c]*, R. Villari [a], F. Moro [a], D. Flammini [a], A. Frattolillo [a], F. Lucca [b], A. Marin [b], F. Viganò [b], F. Cismondi [d]

[a] *ENEA, FSN Department, Rome, Italy*
[b] *LT Calcoli Srl, Merate (LC), Italy*
[c] *SBAI department, SAPIENZA – University of Rome, Roma, Italy*
[d] *EUROfusion Consortium, PPPT Department, Garching, Germany*

In the frame of the EU DEMO development program within the EUROfusion Consortium, the integration of the in-vessel components is crucial even at an early stage of the design process. The auxiliary heating and fueling systems have to be integrated into the Breeding Blanket structure: consequently, they have to withstand severe nuclear loads during operations and, besides, they have a not negligible impact on the Blanket Tritium Breeding capability and shielding performances. This work presents the neutronics and thermal analyses performed in support of the pellet injection systems integration studies. This study is mainly devoted to optimize the design of a fueling line option consisting of a protrusion of the Vacuum Vessel (VV) in the inboard side supporting a guiding tube to drive the pellets along the ideal trajectory as close as possible to the First Wall. An alternative concept with a free-flight injector through the upper port, has also been studied evaluating its feasibility and the shielding needs.

The three-dimensional neutronics simulations were carried-out with the MCNP5 Monte-Carlo code using a DEMO model with the hereinabove integrated systems to calculate neutron fluxes, damage, gas production and nuclear heating distribution as well as the impact on Tritium Breeding Ratio. The neutronic data have been used as loading condition for the 3D thermal analyses and for a design by analysis-activity, carried-out using ABAQUS code.

Results of neutronics and thermal analyses on the fueling systems design and the impact of the penetrations on the Vacuum Vessel and Breeding Blanket loads and shielding performances are presented and discussed.

Keywords: *DEMO, MCNP, Neutronics, Thermal-analyses, Fueling systems*

Corresponding author: *andrea.colangeli@enea.it*

Design of Endoscopes for Monitoring Water-Cooled Divertor in W7-X

P3-084

Joris Fellinger [a*], Uwe Lippmann [b], Henry Greve [a], Mohamad Alhashimi [a], Mathias Schülke [a], Ralf König [a], Marcin Jakubowski [a], Peter Drewelow [a], Axel Lorenz [a], Reinhard Vilbrandt [a], Andre John [a] and the W7-X team

[a] *Max Planck Institute for Plasma Physics, Greifswald, Germany*
[b] *Fraunhofer Institute for Applied Optics and Precision Engineering IOF, Jena, Germany*

Wendelstein 7-X, the largest modular stellarator in the world, is currently being upgraded with 10 water-cooled divertors, designed to withstand 10 MW/m² in steady state operation. To protect the divertor from damage, each divertor is monitored by an endoscope. The endoscope consists of a vacuum compatible plug-in inside the port and an optical box attached to the outside of the vacuum flange. The light from the divertor passes through a pinhole in a protective water cooled head at the plasma side of the plug-in and is reflected by two front mirrors towards the rear end of the plug-in where it is collected by a telescope system that beams the light through a vacuum window. Outside the vacuum, the light is split into an IR- (3-5 µm) and VIS-channel (350-900 nm). In each channel, the light passes a corrector lens, fold mirrors and an objective lens with a filter before it is captured by a camera.

The paper describes the optical design including tolerance assessment which resulted in technical specifications for all optical components. The optical design is very challenging due to the wide angle view of ~120 °C and the need to distinguish between the bulk temperature of the plasma facing tiles and the leading edges bounding the tile within 0.1° viewing angle. Next the thermo-mechanical design is presented, considering thermal loads from plasma radiation and ECRH stray radiation and mechanical loads due to gravity and electromagnetic forces. The impact of deformations on the optical performance is minimized. The design of the water-cooled head is highlighted as it is the first water cooled structure in W7-X that is 3D printed in stainless steel and is optimized to reach maximum cooling with minimal pressure drop in a very narrow construction space. Finally, the assembly strategy supervised by metrology to meet the tolerance requirements is presented, together with the installation procedure allowing for a simple two step mounting of the plug-in and optical box onto W7X.

Corresponding author: *joris.fellinger@ipp.mpg.de*

Progress on the Manufacturing of ITER Thermal Shield

P3-085

Namil Her [a*], Junyoung Hur [a], Kwanwoo Nam [a], Kyoung-O Kang [a], Won Woo Park [a], Manoj Panchal [b], German Perez Pichel [b], Chang Hyun Noh [b], Dong Kwon Kang [b], Youngkil Kang [c], Il Jim Kim [c] and Kisuk Lim [c]

[a] *ITER Korea, National Fusion Research Institute, Daejeon Korea*
[b] *ITER Organization, St. Paul Lez Durance Cedex, France*
[c] *SFA Engineering Corp., Korea*

The role of the ITER thermal shield (TS) is to minimize the radiation heat load transferred from the warm components to the superconducting magnet operating at 4.5K. Silver-coated TS panels which have welded cooling tubes are cooled by 80K of gaseous helium. The main material of the TS is stainless steel 304LN and the total weight is about 900 tons. Fabrication of the TS components is ongoing. Manufacturing of LCTS (Lower Cryostat TS) cylinder and TS Manifold batch-1 has already been completed. The tolerances of the TS panels, the leak tightness of the cooling tubes and the quality of the silver coated surfaces are main concerns during the manufacturing. To avoid potential clash between the TSs and the adjacent components, tight tolerance requirements are applied on the welding and the final machining. The on-site installation feasibilities is validated by the preassembly test at the factory. Leak tightness of the cooling tubes is guaranteed by the test and

the inspection. The surfaces of the cooling tube inside are inspected visually using the endoscope developed specially by the project and the leak tightness has been qualified by the helium leak test. According to the qualified coating process, the TS panels are electro-plated with silver. This paper describes the manufacturing progress and the technical achievements.

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

Keywords: *ITER, Thermal Shield, Silver Coating, Preassembly*

**Corresponding author:* *niher@nfri.re.kr*

Studies on the Tee Extrusion Process of ITER Thermal Shield Manifold Pipes

P3-086

Junyoung Hur [a]*, Namil Her [a], Kwanwoo Nam [a], Kyoung-O Kang [a], Won Woo Park [a], Manoj Panchal [b], German Perez Pichel [b], Chang Hyun Noh [b], Dong Kwon Kang [b], Youngkil Kang [c], Il Jin Kim [c] and Kisuk Lim [c]

[a] *ITER Korea, National Fusion Research Institute, Daejeon, Korea*
[b] *ITER Organization, St. Paul Lez Durance Cedex, France*
[c] *SFA Engineering Corp., Asan, Korea*

The ITER Thermal shield (TS) minimizes radiation heat load from the vacuum vessel and cryostat to superconducting magnets. ITER TS is actively cooled by pressurized helium gas. The TS manifolds (TSM) provides helium coolant from cold valve box (CVB) to cooling tubes on TS panels. During the fabrication of TSM, the most challenging issue is the tee extrusion process which is applied to the branch of pipes. Procedures of tee extrusion is as following: 1) Material inspection, 2) Hole drilling, 3) Extrusion, 4) Surface milling and 5) Orbital welding. After orbital welding, welded areas are tested by radiographic and visual examinations. When NDE is finished, functional tests are carried out by cold shock, pressure and vacuum leak tests. This paper introduces procedures, requirements and acceptance criteria for the fabrication and examination of tee extrusion process. Furthermore, issues and corresponding technical efforts are presented. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

Keywords: *ITER, Thermal Shield, Manifold, Tee extrusion*

**Corresponding author:* *junyoung@nfri.re.kr*

Analysis of the Magneto-Mechanical Coupled Vibration of the In-Vessel Structures of HL-2M Tokamak Considering Halo Current Effect

P3-087

Xudong Li [a], Shejuan Xie [a], Zhenmao Chen [a]*, Yudong Pan [b]

[a] *Shaanxi Engineering Research Center for NDT and Structural Integrity Evaluation, State Key Laboratory for Strength and Vibration of Mechanical Structures,* Xi'an Jiaotong University, Xi'an, China
[b] *Southwestern Institute of Physics, Chengdu, Sichuan, China*

Tokamak in-vessel structures serve in strong magnetic fields. During plasma disruption of vertical displacement events, vibration of the plasma-facing components caused by the Lorentz force is coupled with the magnetic fields giving rise to magnetic damping. The magnetic damping effect has a considerable influence on the dynamic mechanical response of the in-vessel structures under plasma disruption conditions. This paper presents a numerical approach to analysis the magneto-mechanical coupling vibration based on the hybrid of finite element method and boundary element method. Both the eddy current and the halo current are taken into account in the form of a series of filament current and a pair of current source and sink. The proposed approach is applied to a simplified model of the vacuum vessel of HL-2M tokamak under a typical disruption load. Quantitative evaluations of the magnetic damping effect are carried out in consideration of the effect of Halo current in this paper.

Keywords: *magneto-mechanical coupling, numerical simulation, FEM-BEM hybrid method, halo current, HL-2M*

**Corresponding author:* *chenzm@mail.xjtu.edu.cn*

Design and Implementation of a Mobile Parallel Robot for Assembling and Machining the CFETR Vacuum Vessel

P3-088

Changyang Li [a]*, Huapeng Wu [a], Heikki Handroos [a], Juha Koivisto [a], Harri Eskelinen [a], Wenlong Zhao [b] Yuntao Song [b], Zhihong Liu [b]

[a] *Laboratory of Intelligent Machines, LUT University, Lappeenranta, Finland*
[b] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*

The present paper introduces a mobile robot parallel robot for China Fusion Engineering Test Reactor (CFETR) and presents implementation

progress. The task of the robot is to carry out assembling process inside the CFETR vacuum vessel, the assembling process is consisting of scanning, machining, welding and nondestructive testing. To better meet CFETR configuration, the structure and the kinematics of the mobile parallel robot have been optimized for the CFETR access. The design is based on a similar mobile parallel robot demonstrated in ITER developed by the same research group. The solution is also suitable for the assembly of ITER and DEMO. In this paper, the overall design of the mobile parallel robot is presented firstly. Then some calculations and simulations on static, kinematics and workspace of the mobile parallel robot are presented. Finally, the implementation progress of the mobile parallel robot is introduced.

Keywords: *Mobile parallel robot, Vacuum vessel assembling, mechanism, kinematics, FE analysis*

Corresponding author: *Changyang.li@lut.fi*

Tolerance Analysis of Vacuum Vessel for Chinese Fusion Engineering Test Reactor

P3-089

Sumei Liu [a,b]*, Jinxin Sun [a], Liangliang Guo [a], Mingxuan Lu [b]

[a] *School of Engineering, Anhui Agricultural University, Hefei, China*
[b] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*

The vacuum vessel is the key components of the Chinese Fusion Engineering Test Reactor. The manufacturing accuracy of vacuum vessel is vital, which will affect the physical aim of device caused by the assembled accuracy of inner vessel components. The tolerance analysis of the vacuum vessel was performed using the CETOL tolerance analysis software based on the CATIA platform in this paper. Depending on the three-dimensional solid model, the assembled targets of the vacuum vessel are simplified and subdivided, tolerances are assigned according to the overall assembly requirements of the vacuum vessel, and tolerance analysis models are established. The critical dimensions, which could affect the target tolerances, have been determined through the sensitivity and contribution, and the unreasonable tolerances have been redistribute to meet the tolerance requirements of the component, which is very useful to ensure the assembly quality of the vacuum vessel.

Keywords: *Vacuum vessel, assembly, Tolerance analysis, Sensitivity, Contribution*

Corresponding author: *smliu@ipp.ac.cn*

Evaluation of Shielding Ability of Using Boron Water as Coolant in CFETR Vacuum Vessel

P3-090

Yu Lu, **Zhi Chen***

Department of Engineering and Applied Physics, University of Science and Technology of China, Hefei, Anhui, China

Vacuum vessel is a hermetically sealed steel container made up of double steel walls and inwall shielding, which shields the neutron and protects other components in fusion reactor[1]. Cooling water circulates through the vessel's double steel walls and removes the heat generated during operation[2]. Since boron has a large neutron absorption cross section, we wonder if using boron water instead of just water as coolant will have better shield effect. This paper reports a study using Monte Carlo method to evaluate the effectiveness of two different scheme, using water or boron water as the coolant between the double steel walls. The model of vacuum vessel and blanket[3] used in the simulation is a two-dimensional model in the size of the current design of China Fusion Engineering Test Reactor (CFETR). Neutron source is stratified homogeneous DT fusion gaussian spectrum neutron source. In the simulation, we have tested the shield ability of pure water and different consistence of boron water. Neutronic characteristics including neutron flux density, neutron spectrum and nuclear heat have been calculated. The result shows that boron water can reduce the amount of fast neutron, which make damage to steel structure and the superconducting toroidal field magnets. In conclusion, boron water has better shielding ability and can help in-wall shielding to shield neutron from reactor core.

[1] *Zhihong Liu, Jiefeng Wu, Xiaosong Fan, Jianguo Ma, Rui Wang, A study on assembly technology of the CFETR 1/32 Vacuum Vessel, Fusion Engineering and Design, Volume 128, March 2018, Pages 101-106*

[2] *Zhi Chen, Shiji Qiao, Shuai Jiang, X. George Xu, Activation calculation and radiation analysis for China Fusion Engineering Test Reactor, Fusion Engineering and Design, Volumes 109–111, Part A, 1 November 2016, Pages 290-293*

[3] *Zaixin Li, Kaiming Feng, Zhou Zhao, Fengchao Zhao, Kun Xu, Neutronics study on HCCB blanket for CFETR, Fusion Engineering and Design, Volume 124, November 2017, Pages 1273-1276*

Keywords: *Vacuum Vessel, Shielding, Monte Carlo, CFETR*

Corresponding author: *zchen@ustc.edu.cn*

MEST, a New Magnetic Energy Storage and Transfer System: Application Study to the European DEMO

P3-091

Francesco Lunardon [a], Alberto Maistrello [a], Iacopo Spresian [a], Elena Gaio [a], Roberto Piovan [a], Sergio Ciattaglia [b]

[a] *Consorzio RFX (CNR, ENEA, INFN, Università di Padova, Acciaierie Venete S.p.A.), Padova (Italy)*
[b] *EUROfusion Consortium, Garching, Germany*

The European DEMOnstration fusion reactor, presently under Pre-Conceptual Design, will adopt superconducting coils rated for several tens of kA. The DEMO operation will be pulsed; the present estimation of active power required during the plasma formation is of some hundreds of MW, which could be provided by the generator or could be still required to the electrical grid. However, the acceptability to absorb high peaks of active power from the grid could be further reduced in the future, with the increase of distributed generation networks. Furthermore, if the traditional design approach, based on thyristor converters, was adopted to supply the superconducting coils, a huge amount of reactive power, higher than 2 GVar, would be required during most of the plasma pulse. Thus, large Reactive Power Compensation systems should be provided, which calls for additional plant area occupation and further issues in terms of quality of the compensation to be assured at so high power level.

In the frame of the R&D in progress to face these issues, a new magnetic energy storage and transfer system has been conceived, which can improve the power handling. It is particularly suitable to supply the DEMO Central Solenoid (CS), without the need for resistive switching networks, but can be applied to supply the other coils, too. The operating principle of this system, described for CS coils, is to recover the energy from the CS in an additional Superconducting Magnetic Energy Storage coil (TC), pre-charged along with the CS one to the same current value, to perform the flux decrease. After the CS current zero crossing, the energy stored in the TC is transferred back to the CS for the plasma sustainment. The magnetic energy transfer between the CS and the TC is obtained via switched-capacitor. With this approach, the energy is exchanged between the load and the storage system, thus flattening the active power profile to be required to the grid and substantially nullify the reactive power absorbed.

In this paper, the application of this concept to the European DEMO is studied, starting from the present CS magnet and circuits configuration and from the current and voltage scenario under consideration for the plasma breakdown and ramp-up, with the main aim to evaluate if the required dynamics for the switching system is compatible with the so high power level of this specific application. A tentative rating of the system components will be reported, discussing also the future R&D stages to explore the industrial feasibility of such a scheme.

***Corresponding author:** *francesco.lunardon@igi.cnr.it*

Fluid-Dynamic Investigation of Water-Boron Flow in the Vacuum Vessel of DTT Facility

P3-092

Emanuella Martelli*, S. Roccella, G. Barone, Eliseo Visca

ENEA, Department of Fusion and Nuclear Safety Technology, Rome, Italy

The Divertor Tokamak Test (DTT) facility is the intermediate step between ITER and DEMO for the realization of a nuclear fusion power plant. The aim of the facility is to investigate high thermal loads. The DTT Vacuum Vessel (VV) provides an enclosed, vacuum environment for the plasma and acts as a first confinement barrier. The VV is designed as a D-shaped, double wall structure in which borated water flows to moderate the neutron streaming and, consequently, to reduce the nuclear heating density in the toroidal field coils winding pack to 1 mW/cm^3. The temperature of the vacuum vessel in normal experimental operation should be maintained between 50°C and 80°C to ensure high boron concentration (B<0.8 weight% - 95% ^{10}B enriched) and to avoid the problem of the corrosion at high temperature. The borated water flow path should ensure that there are not stagnant or recirculation regions.

A research activity has been performed to investigate the hydraulic behavior of the borated water. The analyses have been conducted through a Computational Fluid Dynamic (CFD) approach, thus a finite volume model of the fluid domain of one Vacuum Vessel sector (20°) has been developed. Steady state simulations have been carried out using ANSYS CFX code to evaluate pressure drops of the borated water and to identify any recirculation or stagnant region.

Keywords: *DTT, Vacuum Vessel, CFD, thermal-hydraulic, borated water*

***Corresponding author:** *emanuela.martelli@enea.it*

Hydrostatic Pressure Test of the ITER Lower Port Stub Extension for Factory Acceptance

P3-093

Hokyu Moon [a*], Hak-Kun Kim [b], Hyun-Soo Kim [a], Yu-Gyeong Kim [a], Woo-Ho Chung [a]

[a] *ITER Korea, National Fusion Research Institute, Daejeon, Korea*
[b] *ITER Organization, Saint Paul Lez Durance, France*

To check the structural integrity of the ITER Lower Port Stub Extension (LPSE), the hydrostatic pressure test has been conducted as part of the Factory Acceptance Test (FAT). The test is conducted as complying with the RCC-MR 2007 code and the French regulations of nuclear pressure equipment (ESPN) because the LPSE is classified as a nuclear pressure equipment (NPE). Due to the fact that the LPSE is double wall structure with inner shell, ribs and outer shell, the inside coolant passage is filled with water for test. Pressurization is kept until 37.8 bar using the certified deionized water. During the test, before and after pressurization, the permanent deformation is verified. To check the permanent deformation, the actual displacement is measured as applying dimensional inspection method using a laser tracker at the 18 points of the LPSE, which are specified by the structural analysis results. As a result, there is no visible leakage on the welded and any interface area. Especially, after pressurization, the measuring displacement is appeared similar to the numerically estimated value, the maximum is 0.38 mm. Finally, after relief the pressure, no permanent deformation is occurred, so that the structural integrity of the LPSE is confirmed through the hydrostatic pressure test.

Keywords: *ITER, Vacuum Vessel Port, Factory Acceptance Test, Pressure Test, Dimensional Inspection*

Corresponding author: *hkmoon@nfri.re.kr*

Progress on the Qualification of Key Process for ITER Correction Coils

P3-094

Erwu Niu [a*], Jing Wei [b], Delong Luo [a], Kaihui He [a], Wen Zhao [a], Yalan Zhu [a]

[a] *China International Nuclear Fusion Program Execution Center, China*
[b] *Institute of Plasma Physics Chinese Academy of Science, China*

18 superconducting coils composed the ITER correction coils, which are used to correct magnet asymmetries and reduce the magnet error caused by the manufacturing and assembly of the TOKAMAK. Depending on the function and the operating environment of the magnets system, the correction coils are mainly composed of the winding pack with insulation, Helium inlet and outlet, superconducting joint, the stainless steel case outside the winding pack. These components would directly affect the final property of the correction coil, therefore all the key process during the coil manufacturing need to be qualified before the series production.

The key qualification process includes the process of winding, welding for the helium inlet and outlet, vacuum pressure impregnation, case closure welding and the superconducting joint manufacture. This article describes each qualified process in detail, including the requirement of the qualification both for the product and the process, the mock up designed both for the sample and the prototype, the activities for the manufacturing and test as well as the results.

Corresponding author: *niuew@iterchina.cn*

Design and Test of the Vacuum-tight Electrically-Insulated Crossed Joints of the New Vacuum Vessel for the RFX-mod2 Experiment

P3-095

Simone Peruzzo *, Mauro Dalla Palma, Andrea Rizzolo, Federico Rossetto, Marco Siragusa

Consorzio RFX, Padova, Italy

An upgrade of the RFX-mod experiment is in progress, involving a major change and reconfiguration of the inner components of the machine assembly. In particular, the present vacuum vessel will be replaced by the external stainless steel toroidal support structure, which will be significantly modified with the integration of approximately 150 vacuum-sealed ports, interfaced with existing machine sub-systems (diagnostics, pumping and fuelling), and the implementation of 2 vertical and 2 horizontal vacuum-tight electrically-insulated crossed joints, necessary to allow suitable penetration of electromagnetic fields within the plasma chamber.

The design of such peculiar joints of the vessel required the development of several mock-ups in which different solutions and arrangement of materials have been tested. The final solution implies the use of high-performance polymers combined with acrylic based syntactic foam materials, which provide proper vacuum tightness (required leak rates < 1E–9 mbar·l/s) and electrical insulation (required dielectric strength > 1 kV/mm), with suitable viscoelastic characteristic that guarantees the compensation of thermal deformations expected during operation.

The paper will describe the numerical FEM analyses performed to verify the thermo-mechanical design of the system at the operating conditions and the experimental tests, in particular leak tests and

outgassing tests, carried out on mock-ups to assess the reliability of the design solution.

Keywords: *RFX-mod2, Vacuum sealing, High-performance polymers, Syntactic foam.*

**Corresponding author:* *simone.peruzzo@igi.cnr.it*

Design of the Ex-Vessel Optical System of the ITER Core CXRS Diagnostic System

P3-096

Gergő I. Pokol [a,b]*, Gábor Szarvas [c], Tétény Baross [b], Gábor Tari [b], Sándor Zoletnik [b], Andreas Krimmer[d], Philippe Mertens [d]

[a] *Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary*
[b] *Wigner Research Center for Physics, Budapest, Hungary*
[c] *Optimal Optik Kft., Budapest, Hungary*
[d] *Forschungszentrum Jülich GmbH, Institute of Energy and Climate Research, Jülich, Germany*

Design of the core Charge Exchange Recombination Spectroscopy (CXRS) diagnostic system of the ITER tokamak poses a great number of challenges due to the combination of demanding measurement requirements, nuclear operating environment and great number of interfaces. Present contribution summarizes the challenges and subsequent solutions of the design of the ex-vessel part of the ITER core CXRS diagnostic system. Ex-vessel system components are distributed in regions of port interspace, bioshield and port cell. Issues considering the design specification, optical design choices and mechanical design solutions are discussed. Interfacing to the surrounding tokamak components and to the rest of the diagnostic system is deliberated in detail. Some general conclusions can be applied to any optical diagnostic system of a nuclear fusion device.

Keywords: *tokamak, ITER, CXRS, optical diagnostic*

**Corresponding author:* *pokol@reak.bme.hu*

Design of the European DEMO Vacuum Vessel Inboard Wall

P3-097

Rocco Mozzillo [a]*, Domenico Marzullo [a], Christian Bachmann [b], Giacomo Aiello [c]

[a] *CREATE, University of Naples Federico II,DII, Naples Italy*
[b] *EUROfusion PMU, Garching, Germany*
[c] *CEA Saclay, Gif-Sur-Yvette, France*

The pre-concept design of the DEMO Vacuum Vessel is going on in view of the 2020 gate review; moreover the nuclear heat loads on the vessel inner shell were determined and found to be about one order of magnitude higher compared to ITER. A subsequent thermal-structural analysis of the vessel inner shell revealed high thermal stresses and a large temperature gradient through the inner shell thickness. Given the simultaneous occurrence of primary membrane stresses in the entire vessel inboard wall and, in proximity of the vessel ribs, high bending stresses due to the coolant pressure, a survey of all options within the design rules was required to identify the inter-dependencies of the individual stress limits (primary membrane, primary bending, thermal membrane plus bending). In order to face this kind of issues a detailed assessment on the design of the inboard wall of DEMO Vacuum Vessel has been conducted and here presented. The current work evaluates both the P and S type damages for the inboard wall of DEMO Vacuum Vessel in case of high nuclear heat load, vacuum vessel coolant pressure and toroidal field coils fast discharge radial pressure. The elastic analysis method has been used to check the rules for prevention of both types of damage. The rules applied to prevent the aforementioned damages are compliant to Level A criteria, in case of negligible creep and negligible irradiation. In order to check the structural integrity of the inboard wall of DEMO VV against high thermal and mechanical loads, optimization analyses have been run, starting form parametric modeling of single slice of DEMO VV inboard wall. The aim of the present work consisted on the check of the structural integrity of DEMO Vacuum Vessel inboard wall applying all provided rules in the relevant vessel design code (RCC MRx).

Keywords: *DEMO, Vacuum Vessel, Ratcheting, FEM*

**Corresponding author:* *rocco.mozzillo@unina.it*

Numerical Simulation of the Temperature State of T-15MD Vacuum Vessel and In-Vessel Components

P3-098

Konstantin Senik *, Konstantin Borisenko, Sergey Grigoriev, Evgeny Kuzmin, Oleg Smirnov, Victor Tanchuk

JSC "NIIEFA", Saint-Petersburg, Russia

The T-15MD tokamak is an important contribution to the progress of the Russian controlled fusion program. The vacuum vessel of the T-15MD tokamak is one of the main systems of the reactor designed to obtain the vacuum volume with the characteristics ensuring the formation and confinement of the plasma in the impulse regime of plasma burning at a thermal power of 10-14 MW, pulse length of 5 s and repetition rate of 900 s.

The finite-element model of a 22.5° regular sector of the vacuum vessel (VV) with all in-vessel components and external heaters developed by the authors was applied for numerical simulation of the normal operation and VV baking process. The performed analysis has allowed the authors to investigate the possibility for the tokamak to operate at long-duration plasma burning (up to 18 s).

The regime of vessel baking to a temperature not lower than 200°C is used to remove residual gases and water vapors adsorbed in pores of the vacuum vessel shell and in-vessel components. The vacuum vessel of the T-15MD tokamak is proposed to bake by ohmic heaters fastened on the VV shell. The numerical simulation of the baking regime carried out on the developed model of the sector has made it possible to define a) the baking scenario and duration; b) temperature non-uniformities during the baking process; c) requirements for power supply sources of the heaters.

On the basis of the numerical analysis the requirements have been stated for the baking systems and water cooling systems for the upper and lower divertors; their parameters ensuring the required temperature operation regimes have been defined; their temperature states and levels of heat losses from the surface of the thermally insulated vacuum vessel have been determined.

Keywords: *tokamak, vacuum vessel, in-vessel components, normal operation, baking*

Corresponding author: *senixon@rambler.ru*

Manufacturing and Welding Assembly of the Vacuum Vessel on JT-60SA

P3-099

Yusuke Shibama [a]*, Kaname Kizu [a], Fuminori Okano [a], Junnichi Yagyu [a], Keiichi Sagawa [b], Toshihisa Okuyama [b], Shoichi Mizumaki [b], Tamotsu Morimoto [b], Atsuro Hayakawa [b]

[a] *National Institutes for Quantum and Radiological Science and Technology, Ibaraki, Japan*
[b] *Toshiba Energy Systems & Solutions Corporation, Yokohama, Japan*

The superconducting tokamak JT-60SA is being constructed for national project and Japan-EU Satellite Tokamak Program. The accuracy of this construction is required to be less than several millimeters to achieve the 0.01% field error in the toroidal magnetic field. This extremely low field error requires an assembly to maintain the torus symmetric axis of the large components with a scale of more than ten meters. The vacuum vessel (VV, 150 tons) was designed as a torus vessel, the 10 m outer diameter and 6.6 m high, for core plasma to keep the ultra-high vacuum space, and made of low cobalt stainless steel 316L. The double wall structure is adopted, as the wall thickness of 18 mm and rib jointed of 22 mm, to keep the structural stiffness against the large electromagnetic force at plasma disruption and the toroidal one-turn resistance of 15 μΩ. The vessel temperature during the plasma operation is at 50°C and the radiation shielding water is circulated in the double wall at plasma operation to reduce the nuclear heating of the superconducting magnets. The vessel during baking operation is kept at 200°C with nitrogen gas circulation instead.

The VV was manufactured as the 10 sectors split in the factory and weld-assembled in the onsite. The dimensions as the torus were well controlled regardless of the thin wall thickness and a large amount of welding line due to the 72 port penetrations, and the complete penetration was inspected for structural integrity by the ultrasonic test and radio graphical test combinations. The final VV 20-degree sector was installed into the Tokamak with the TF Coil, and the VV was weld-jointed as a torus. This paper reports the VV manufacturing and weld-assembling results with its design concept.

Keywords: *Vacuum Vessel, Welding Assembly, Torus, Tokamak, JT-60SA*

Corresponding author: *shibama.yusuke@qst.go.jp*

RF Discharge Mirror Cleaning System Development for ITER Diagnostics

P3-100

Pavel Shigin [a*], Nikita Babinov [b], Artem Dmitriev [b], Jens Larsen [c], Frank Leipold [d], Rene Madsen [c], Laurent Marot [e], Lucas Moser [e], Eugene Mukhin [b], Mikhail Kochergin [a], Alexey Razdobarin [b], Roger Reichle [a], Dmitry Samsonov [b], Victor Udintsev [a], George Vayakis [a], Michael Walsh [a], Rong Yan [f]

[a] *ITER Organization, St Paul Lez Durance Cedex, France*
[b] *Ioffe Institute, St.Petersburg, Russia*
[c] *Polyteknik AS, Oestervraa, Denmark*
[d] *Department of Physics, Technical University of Denmark, Lyngby, Denmark*
[e] *University of Basel, Basel, Switzerland*
[f] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*

Optical diagnostic systems are critical for the successful operation of ITER. Plasma facing mirrors (first mirror) of diagnostic systems are critical components responsible for the system operation. A number of plasma wall interaction phenomena will lead to degradation of the reflectivity of the first mirror surface thus decreasing the performance of whole diagnostic system. Plasma facing mirrors are located behind openings inside the diagnostic first wall. Access to diagnostic systems in ITER is generally challenging after initial assembly. Taking into account the very limited access, first mirrors will be equipped with *in-situ* cleaning systems to periodically recover mirror reflectivity. The system must operate in the ITER harsh environment with high reliability. There are two main branches of active mirror performance recovery systems in ITER: pulsed DC discharge and RF discharge based systems. This report focuses on the development of RF discharge mirror cleaning system.

Several R&D tasks have been executed in order to reveal the basic features of the RF discharge based mirror cleaning systems in the given ITER constraints and conditions. Engineering development, orientated towards the design of the mirror cleaning system and its implementation in ITER has been launched. This report summarizes the status of these R&D and engineering activities. On the basis of these findings key requirements and specifications for such mirror cleaning systems for ITER conditions have being developed and will be presented as well.

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

Keywords: *ITER, first mirror, mirror cleaning, diagnostics, RF discharge*

Corresponding author: *pavel.shigin@iter.org*

Structural Pre-Conceptual Design Studies for a EU DEMO Equatorial EC Port Plug and its Port Integration

P3-101

Peter Spaeh [*a], Christian Bachmann [b], René Chavan [c], Aljaz Cufar [d], Thomas Franke [b,e], Dirk Strauss [a], Minh Quang Tran [c]

[a] *Karlsruhe Institute of Technology, Karlsruhe, Germany*
[b] *EUROfusion Power Plant Physics and Technology (PPPT) Department, Garching, Germany*
[c] *Swiss Plasma Centre EPFL, Lausanne, Switzerland*
[d] *Reactor physics department, Jožef Stefan Institute, Ljubljana, Slovenia*
[e] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*

For the EU DEMO Tokamak, EC launching systems for plasma heating and stabilization are under development. Various concepts for the optical system are currently studied of which the Mid Steering Antenna (MSA), the Open Ended Waveguide (OEWG) concept and the Remote Steering Antenna (RSA) are the basic ones. Also hybrid solutions, which means a port plug with different launchers concepts are taken into consideration. In parallel, design drafts for a generic equatorial port plug are sketched with the aim to provide a versatile structural system, which allows customized installation of the potential optical systems. This paper presents a pre-conceptual design of an equatorial EU DEMO EC port plug based on MSA in detail, taking into account the exact port position with respect to the toroidal field coil, mechanical integrity, heat dissipation, neutronic shielding requirements, design integration and maintenance concepts. Based on this, general proposals for the further RSA and OEWG integration will also be made.

Keywords: *ECRH, Port Plug, Heating and Current Drive, Plasma stabilization, Structural design*

Corresponding author: *peter.spaeh@kit.edu*

Condition Monitoring for a Robot Machine in the Assembly of Fusion Reactor Vacuum Vessel

P3-102

Shayan Moradkhani*, Huapeng Wu, Heikki Handroos

Lappeenranta University of Technology, Lappeenranta, Finnland

The assembly of vacuum vessel (VV) is a vital task for the fusion reactor. this paper introduces a parallel robot machine built up for the assembly of fusion reactor CFETR to carry out welding , splice plate transportation and machining inside the VV . To ensure the reliability

and safety of the automated process, a condition monitoring system of robot machine based on virtual-reality simulation environment is developed which parallelly works with the control of robot. The faults estimation module based on the residue between Realtime simulation and sensor information is constructed, the faults such as collision, milling tool wear out and over cutting are tested . The results are validated with laboratory experiments on the actual robot machine.

**Corresponding author: shayan.moradkhani@lut.fi*

Safety Analysis of the DONES Primary Heat Removal System

P3-103

Pietro Arena [a]*, Pietro Alessandro Di Maio [b], Francesco Saverio Nitti [a]

[a] *ENEA C.R. Brasimone, (BO), Italy*
[b] *Department of Engineering, University of Palermo, Palermo, Italy*

The development of a neutron source able to reproduce the irradiation conditions typical of a nuclear fusion reactor, in order to test candidate structural materials, is the main goal of the Work Package Early Neutron Source (WPENS) of the EUROfusion action. This source, named Demo Oriented NEutron Source (DONES), is a facility where neutrons are produced by means of D-Li interactions. More in detail, a beam of 125 mA deuterium ions at the energy of 40 MeV strikes a lithium jet flowing in a purposely shaped channel in order to obtain an intense and stable neutron flux for the irradiation of material samples.

In the framework of these activities, safety analysis are a key aspect in the DONES design and development. Among the postulated initiating events (PIEs) identified during the preliminary Failure Mode Analysis, the Loss Of Flow Accident (LOFA) in the Primary Heat Removal System (PHRS) of the lithium loop, due to a trip of the electro-magnetic pump, is one of the most severe. In fact, the loss of lithium flow, combined with the failed stop of the accelerator, could lead to the destruction of the lithium flow channel in correspondence of the component named back-plate. For this reason it has been chosen to investigate the LOFA adopting the deterministic system code Relap5-3D.

Results obtained are critically discussed and compared with those obtained by a similar calculation carried out with MELCOR 1.8.6 code, in order to assess the Relap5-3D capability of describing systems adopting lithium as working fluid.

Keywords: *DONES, Relap5-3D, lithium, safety analysis, MELCOR*

**Corresponding author: pietro.arena@enea.it*

Exploring the Adoption of Mobile Augmented Reality for Assistance in Fusion Plant Repair and Maintenance

P3-104

Thomas Bale [a]*, Andrew Calway [a], Kirsten Cater [a], Robert Skilton [b]

[a] *University of Bristol, Senate House, Tyndall Ave, Bristol, UK*
[b] *Culham Centre for Fusion Energy, Abingdon, Oxford, UK*

Augmented Reality (AR) applications combine real and virtual content to create a 'mixed reality' experience for users. To aid repair and maintenance, mobile AR applications can be used to layer 'virtual' instructions and visualisations over meaningful referents, facilitating user interaction, informing perception or facilitating collaboration. Previous studies have reported industrial maintenance workers supported by AR can achieve faster completion times and reduced error rates whilst completing common tasks. This review explores the opportunities, challenges and risks associated with adoption of AR tools specifically in Fusion Plant repair and maintenance. Drawing on industrial stakeholder interviews, review of existing practice, and Fusion Plant operational observations; we explore the potential of mobile AR to improve performance. The results make several contributions: Firstly we explore the perceived strategic opportunities and risks associated with AR assistance in a Fusion Plant environment. From an operational perspective we investigate relevant tasks where AR assistance may deliver performance gains. Lastly we investigate user interaction issues relevant to the integration of AR assistance into maintenance and repair workflows. These results provide a holistic view of AR within a Fusion Plant context and highlight areas of further work needed.

Keywords: *Augmented Reality, Fusion, Maintenance, Repair, AR, Computer Vision*

**Corresponding author: thomas.bale@bristol.ac.uk*

Primary Heat Transfer System Design of the WCCB Blanket for Multiple Operation Modes of CFETR

P3-105

Xiaoman Cheng [a]*, Xuebin Ma [b], Lei Chen [a], Songlin Liu [a]

[a] *Institute of Plasma Physics Chinese Academy of Sciences, Hefei, China*
[b] *Shenzhen University, Shenzhen, China*

The water cooled ceramic breeder (WCCB) blanket is one of the blanket candidates for Chinese Fusion Engineering Test Reactor (CFETR). The WCCB blanket design has been updated according to the latest core parameters of CFETR (major radius R=7.2 m, minor radius r=2.2 m). The Primary Heat Transfer System (PHTS) of the WCCB blanket is required to remove heat and convert fusion energy under multiple operation modes of CFETR without changing any part of the system so as to save construction costs and reduce waste inventory. It is challenging as the heat source of the PHTS differs greatly with the fusion power ranging from 200 MW to 1.5 GW for multiple operation modes of CFETR. Therefore, the PHTS is designed taking 1.5 GW of fusion power as full power. Then lower fusion power modes can be satisfied with margin. Based on the WCCB blanket module structure, the PHTS is characterized by 3 independent systems, namely System 1 for the cooling of the First Wall (FW), System 2 & 3 for Breeding Zones (BZs). At 200 MW of low fusion power, System 3 is idle. At 500 MW to 1.5 GW of high fusion power, all 3 systems are put into use to ensure adequate cooling. Detailed design of the PHTS is performed, including system layout, pipe sizing, system components configuration (e.g., pump, steam generator, pressurizer). The reliability of the PHTS design is verified under steady states of 200 MW, 500 MW, 1.0 GW and 1.5 GW of fusion power.

Keywords: *CFETR, WCCB Blanket, PHTS design*

Corresponding author: *cxm@ipp.ac.cn*

EU DEMO Plant and Building Layout Criteria

P3-106

Sergio Ciattaglia [a]*, G. Federici [a], L. Barucca [b], M. De Magistris [c]; E. Gaio [d], C. Gliss [a], M. Koerber [e], I. Moscato [f], MT Porfiri [g], F. Riedl [e], A. Tarallo [c]

[a] *EUROfusion Consortium, Programme Management Unit, Garching, Germany*
[b] *ANSALDO Ansaldo Nucleare, Genova, Italy*
[c] *CREATE, University of Naples Federico II, Naples, Italy*
[d] *Consorzio RFX, Padova, Italy*
[e] *Framatome GmbH, Erlangen, Germany*
[f] *Ingegneria, Università di Palermo, Palermo, Italy*
[g] *ENEA, Fusion and Technology for Nuclear Safety and Security Department, Italy*

An early attention to the layout of both plant site and its buildings is essential in a complex plant under preliminary design as DEMO in order to meet the assigned targets, namely i) the licensing requirements ii) a good availability in delivery electricity to the grid.

The layout definition has to follow several criteria that become more complex and stringent for nuclear buildings, e.g. functional, maintenance, fire protection, safety, human factors, shielding, and remote handling. The criterion As Low As Reasonable Achievable with respect to the dose to the staff has to be applied in design, operation, maintenance and decommissioning phases.

The tokamak building, where several complex systems have to converge to the torus to create and control the plasma, to take out its energy and to produce and extract tritium, provides the second and ultimate confinement barrier between the environment and the hazardous and radioactive materials present inside that might be mobilised, in case of accident, by the high energetic fluids stored in DEMO systems.

The layout criteria are focused on avoiding any challenge to the safety functions: e.g. no common mode failures of the safety classified systems for all reference design basis events. Furthermore the safety classified equipment have to maintain their safety function all over the plant life in such challenging environmental conditions; an accurate layout might allow the qualification possible, making milder the environment, e.g. defining adequate shielding and areas where radiation dose, magnetic field and accidental environmental conditions are reasonable for sensible equipment. The experience of NPPs and ITER is also recognized as the basis of such design criteria for DEMO. The paper will outline the

main design basis events and the layout criteria presenting some applications for the tokamak building that reflect the recent progress of the DEMO design.

Keywords: *Layout, safety, building, tokamak, criteria*

***Corresponding author:** *sergio.ciattaglia@euro-fusion.org*

Design and Verification of a Non-Self-Supported Cryostat for the DEMO Tokamak

P3-107

Łukasz Ciupiński [a*], Tomasz Zagrajek [b], Piotr Marek [b], Grzegorz Krzesiński [b], Christian Bachmann [c]

[a] *Faculty of Materials Science and Engineering, Warsaw University of Technology, Woloska,* Warsaw, Poland
[b] *Faculty of Power and Aeronautical Engineering, Warsaw University of Technology, Warsaw, Poland*
[c] *EUROfusion - Programme Management Unit, Garching, Germany*

Europe initiated, in 2014, a comprehensive design study of a DEMOnstration Fusion Reactor (DEMO) as an action implementing the Roadmap to Fusion Electricity Horizon 2020. Within the frame of this development a concept of the cryostat is proposed being supported by the reactor bioshield rather than being self-supported. This paper describes the design and its structural verification by means of Finite Element modelling.

The cryostat is a large pressure vessel providing the vacuum required to operate the superconducting coils at cryogenic temperatures. Cryostats of existing machines and ITER typically are cylindrical and self-support the external pressure. In a nuclear machine like DEMO, a massive bioshield encloses the tokamak providing radiological protection to maintenance areas in the primary building. The cryostat concept presented in this article is supported by the bioshield making use of its significant strength. This allows substantially reducing the amount of steel needed for the cryostat construction.

The cryostat is a conventional pressure vessel and designed according to ASME VIII, Div. 2. In addition to withstand 1 bar of external pressure the 2nd critical load case is the ingress of air or Helium. Any gas inside the cryostat will be cooled down by the large mass of the cryogenic magnet coils and consequently cool down the cryostat.

Linear and nonlinear structural and thermal-structural assessments show that the tube segments introduced in the cryostat design provide both, the required membrane strength to withstand the external pressure as well as the required flexibility to allow the thermal contraction of the cryostat in case of a loss of vacuum event. It was also found that the relatively thin shells of the cryostat are not capable of bearing any significant internal overpressure. It is hence suggested to install a rupture disk in the cryostat to release Helium into the building in case of large internal leaks of liquid Helium.

Keywords: *DEMO cryostat, design, structural verification, Finite Element Method*

***Corresponding author:** *Lukasz.Ciupinski@pw.edu.pl*

Thermal-Hydraulic Modeling and Analysis of the Water Cooling System for the ITER Test Blanket Module

P3-108

Cristiano Ciurluini [a*], Fabio Giannetti [a], Amelia Tincani [b], Alessandro Del Nevo [b], Gianfranco Caruso [a], Italo Ricapito [c], Fabio Cismondi [d]

[a] *Sapienza University of Rome, Department of Astronautical, Electrical and Energy Engineering (DIAEE), Italy*
[b] *ENEA FSN-ING, ENEA CR Brasimone, Camugnano (BO), Italy*
[c] *F4E, ITER Department, Barcelona, Spain*
[d] *EUROfusion Consortium, PPPT Department, Garching, Germany*

ITER Test Blanket Module program is aimed at providing the first experimental data on the breeding blankets (BB) performances in an integrated fusion nuclear environment. This is the first step to test scaled DEMO BB components and circuits. The program foresees the test of six breeding blankets mock-ups, called Test Blanket Module (TBM) with all the related ancillary systems. The Water Cooled Lithium Lead (WCLL) is one of the selected breeding blanket concepts to be investigated in the EUROfusion Breeding Blanket Project (WPBB), which last from 2014 to 2020, and it was also recently chosen as one of the six mock-up for ITER TBM program. A pre-conceptual design of the Water Cooling System (WCS) of the ITER WCLL-TBM was developed considering the same functions of the EU-DEMO WCLL-BB primary heat transfer system (PHTS). The DEMO BB PHTS is composed by two independent cooling circuits, one for the breeding zone and one for the first wall. Since, for ITER TBM, the power produced is scaled, only one loop (WCS) was used to cool both the breeding blanket and the first wall. Furthermore, there are different constraints in terms of heat sink temperatures, much smaller than the ones of DEMO Power Conversion System, since the cooling water is provided from the Component Cooling Water System of ITER reactor. The TBM was analyzed to evaluate the influence of the circuit design in operating conditions. A pressurizer was sized, maintaining the TBM inlet water pressure at the required value.

A thermal-hydraulic model of the water cooling system, including all the main components, has been developed using RELAP5 system code to verify components sizing and to investigate the system behavior during steady-state and transient conditions. Pressure drops along the system and heat exchanger performances during a Normal Operational State (NOS) were checked as part of the nodalization qualification process. Moreover, an operational transient was simulated, starting from Hot Standby Operational State (HSOS) to NOS.

Keywords: *ITER, WCLL-TBM, WCS, RELAP5, NOS*

*Corresponding author: cristiano.ciurluini@uniroma1.it

High-Speed Generation of Neutronics-Ready CAD Models for DEMO Design

P3-109

Matti Coleman [a,c*], S. McIntosh [a,b], J. Shimwell [a], A. Davis [a]

[a] *United Kingdom Atomic Energy Authority, Culham Science Centre, Abingdon, Oxfordshire, UK*
[b] *ITER Organization, St Paul Lez Durance Cedex, France*
[c] *Department of Engineering, University of Cambridge, Cambridge, UK*

Since the 1960's computer-aided design (CAD) has been at the forefront of engineering. Originally and literally quite a broad term, "CAD" appears to have narrowed in definition, and is nowadays usually used to refer to three-dimensional (3-D) geometric models. Such 3-D CAD models are the currency of modern engineering, enabling a wide variety of different design and analysis activities to take place.

The creation of 3-D CAD models is one of the key steps in the design cycle, and is very much a recurring process. A common division of labour sees engineers, draughtspeople, and analysts iterate ideas, requirements and constraints, 3-D CAD model(s), and analyses, respectively, to iteratively converge upon a working design. We contend that, during the conceptual design, this traditional design cycle is no longer optimal, due to the large overheads associated with dealing with relatively mercurial geometries, load cases, and boundary conditions.

In this work, we demonstrate that the automatic generation of 3-D CAD models for conceptual DEMO-class tokamak reactors is possible. The models, which comprise all major components ranging from the in-vessel components to the concrete radiation shield, are automatically produced for a given design point in approximately 30 seconds. This represents a significant acceleration of the traditional manual 3-D CAD model generation process for a typical reactor (∼1 month). The CAD data are generated within a reactor design code and can be used for a range of purposes. We highlight the use case for neutronics, by demonstrating that a 3-D global neutronics model can be rapidly initialised and run automatically for a design point using surface meshes, smeared material properties, and a fusion neutron source term parameterisation.

Keywords: *DEMO, CAD, neutronics, fusion*

*Corresponding author: matti.coleman@ukaea.uk

Shielding Concept and Neutronic Assessment of the DEMO Lower Remote Handling and Pumping Ports

P3-110

Aljaž Čufar [a*], Christian Bachmann [b], Davide Flammini [c], Curt Gliss [b], Ivan A. Kodeli [a], Domenico Marzullo [d], Giuseppe Mazzone [c], Christian Vorpahl [b], Andrew Wilde [e]

[a] *Reactor Physics Department, Jožef Stefan Institute, Ljubljana, Slovenia*
[b] *EUROfusion Programme Management Unit, PPPT, Garching, Germany*
[c] *ENEA, Department of Fusion and Nuclear Safety Technology, Frascati (Rome), Italy*
[d] *CREATE, University of Naples Federico II, DII, Naples, Italy*
[e] *CCFE, Culham Science Centre, Abingdon, Oxfordshire, UK*

Within the EUROfusion Power Plant Physics and Technology Department the DEMOnstrational fusion power plant (DEMO) is being developed. One of the fundamental challenges is the integration of ports in the vacuum vessel to allow for access of various tokamak systems while at the same time provide the necessary neutron and gamma radiation shielding in areas affected by the ports.

The lower port of the DEMO machine is particularly challenging due to tight space constraints imposed by the toroidal field (TF) coils and the requirement to provide a large open duct through both the divertor and inside the port to enable for vacuum pumping. In addition, feeding pipes of divertor and tritium breeding blanket need to be integrated and access space must be provided for various remote handling operations.

Several neutronics requirements need to be fulfilled. First, the nuclear heating of the superconducting TF coils must be limited. This is achieved through design iterations modifying the shape of the port, its wall thickness, and the implementation of additional shields. Secondly, the gamma radiation levels inside the cryostat and behind the lower port need to be limited to reduce occupational exposure to personnel during maintenance. Finally, the irradiation damage and neutron heating in different divertor components and in the vacuum vessel need to be considered in the design and limited when excessive. The corresponding results of neutronic analyses are presented in this article and the direction of future design developments discussed.

Keywords: *DEMO, lower port, nuclear heating, material damage, port integration*

*Corresponding author: aljaz.cufar@ijs.si

Nuclear Assessment of the IFMIF-DONES Lithium Target System

P3-111

Manuela Frisoni [a]*, Davide Bernardi [b], Francesco Saverio Nitti [b]

[a] *ENEA, C.R. Bologna, Italy*

[b] *ENEA, C.R. Brasimone, Italy*

In the framework of the Work Package Early Neutron Source (WPENS) of the EUROfusion Consortium, the engineering design of the IFMIF-DONES (International Fusion Material Irradiation Facility- Demo Oriented Neutron Source) Lithium target system has been accomplished. The model actually includes a target system based on the so-called integral concept consisting in a fully welded component. Moreover, the Quench Tank is sited inside the Test Cell. The development of the design required that new neutronic calculations had to be performed in ENEA, in order to update those already accomplished in the past for the previous geometrical layout. The aim of these evaluations is to provide quantities, such as power deposition, dpa and gas production, useful for the thermo-mechanical analysis that is required for assessing the structural behavior of the system subjected to irradiation. Coupled neutron-gamma transport calculations have been carried out by using the MCNP5 1.6 code integrated with the McDeLicious-11 neutron source. The geometrical input for MCNP used in the calculations was provided by KIT. Neutron activation calculations have been also performed by means of the FISPACT-II activation code package in order to provide radioactive quantities, such as decay heat and contact dose rates, in various parts of the system, useful for thermo-mechanical analysis and safety purposes, like waste management considerations. This paper presents the main results obtained from the above analyses.

Keywords: *IFMIF-DONES, lithium target, neutronics, activation*

Corresponding author: *manuela.frisoni@enea.it*

Integration of DEMO RadioActive Fluids Piping into the Tokamak Building

P3-112

Curt Gliss [a]*, C. Bachmann [a] , S. Ciattaglia [a], G. Federici [a], M. Koerber [b], I. Moscato [c], F. Riedl [b], M. Utili [d]

[a] *EUROfusion Consortium, Garching, Germany*

[b] *Framatome Erlangen, Germany*

[c] *Ingegneria, Università di Palermo, Palermo, Italy*

[d] *ENEA FSN-ING-PAN, ENEA CR Brasimone, Camugnano (BO), Italy*

The complexity of a Nuclear Fusion Power Plant, like EU DEMO, asks for a harmonic development of the design of the major systems in order to deliver a plant able to reach the assigned targets, namely i) the licensing safety requirements, ii) personnel security in context with operation and maintenance, iii) a good availability and iv) the delivery of few hundreds MW electric net to the electrical grid.

The homogenous progress of the entire design allows a better and continuous control of the numerous physical and functional interfaces among the systems and structures assuring an optimization of the overall design focused on the above targets.

An early focus to the plant site layout and to the layout of the buildings is a major key in that respect.

Few DEMO plant systems inside the tokamak building contain radioactive material that emits energetic radiation gamma as activated corrosion products in the water primary coolant circuits and Li-Pb system as well as N16 and N17 produced by the activation of water oxygen. Therefore the layout of the piping of these fluids, running outside the cryostat, requires special attention in order to limit the challenges for the qualified life of systems and components localized in the tokamak building, e.g. electric and electronic equipment, organic seals of valves, and relevant actuators. Furthermore the personnel risk and the dose to the staff during operation and maintenance has to be limited to as low as reasonable achievable.

The experiences of NPPs and ITER are recognized as important basis of the piping layout for DEMO.

The paper presents the piping layout criteria inside the tokamak building and its first integration in order to meet the criteria above quoted.

Keywords: *Layout, piping, tokamak, radioactive, fluids*

Corresponding author: *curt.gliss@euro-fusion.org*

Comparative Neutronics Analysis of the European HCPB DEMO Using SuperMC

P3-113

Bin Li [a], Bin Wu [a], Guangyao Sun [a], Lijuan Hao [a]*, Ulrich Fischer[b], **Jiangtao Jia** [a], FDS Team [a]

[a] *Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Hefei, Anhui, China*
[b] *Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany*

In a D-T fueled tokamak, neutrons not only carry 80% of the energy released per fusion reaction, but also are the source of radioactivity in the fusion system. Therefore, high-fidelity neutronics simulations are required to support the design and safety analysis for such reactors.

In the present work, a comparative neutronics analysis for the European HCPB DEMO, developed by KIT in 2015, has been carried out by using SuperMC. The results are compared with those of MCNP to demonstrate both SuperMC's correctness and its advanced capabilities for fusion neutronics applications. The results confirm that the current blanket scheme satisfy the design requirements from the viewpoint of neutronics and also demonstrate that SuperMC provides consistent results with MCNP for the neutron flux/current, nuclear responses (such as TBR, DPA, etc.) and nuclear heating using both unbiased and biased simulation techniques. Compared to MCNP, SuperMC has some advantages with regard to automated modeling capabilities for complex systems, efficient calculations, and 3D interactive visualizations.

Keywords: *SuperMC, HCPB DEMO, Neutronics, TBR*

**Corresponding author:* *lijuan.hao@fds.org.cn*

Analysis of Various Fusion Power Plant Turbine Cycles

P3-114

Anikó Lilla Hegedűs*

Budapest University of Technology and Economics, Budapest, Hungary

Fusion was discovered earlier than fission, we know how powerful it can get, yet there's no commercial fusion power plant. One of the many yet unsolved problems is the power generation itself: after the fusion reactor steady-state operation is obtained, electricity should be harnessed. The two types of solutions are direct and indirect conversion, I'll discuss the latter. Although it has lower efficiency, it's faster and easier to construct, therefore it brings us closer to commercial fusion energy. In this paper I'll examine the existing steam (Rankine), helium (Brayton) and supercritical CO_2 cycles, in order to find out which one is the best for some better-developed fusion reactor types. Comparations will be done with operating nuclear power plants, when it's possible. I validate my results with detailed cycle analyses in Cycle Tempo.

Keywords: *commercial fusion power, steam turbine, helium turbine, supercritical CO_2 turbine, Cycle Tempo*

**Corresponding author:* *h.anikolilla@gmail.com*

Impact of Neutronic Constraints on Design and Performance of a Tokamak Fusion Reactor

P3-115

Bong Guen Hong*

Chonbuk National University, Jeollabuk-do, Korea

An optimal configuration and system parameters of a tokamak reactor were found by utilizing a new simulation method which couples a conventional tokamak systems analysis and a radiation transport analysis. Neutron impacts on the reactor design were self-consistently incorporated, together with plasma physics and tokamak engineering constraints, which were moderately extrapolated from the International Thermonuclear Experimental Reactor (ITER) model. In a low-aspect ratio tokamak reactor, the minimum major radius to produce a desired fusion power was mainly determined by the shielding requirements, while in a normal aspect ratio tokamak reactor, it was determined not only by the requirements on the shielding, but also by the requirements on the tritium breeding and the magnetic flux density at the toroidal field (TF) coil. As the aspect ratio increased, the minimum major radius and the system size decreased as long as the tritium self-sufficiency was satisfied with only an outboard blanket, but they began to increase as the inboard blanket thickness increased to meet the requirements for tritium self-sufficiency and the TF coil bore radius increased to meet the requirements for the magnetic flux density at the TF coil. The tritium breeding capabilities of the five blanket concepts to be tested as test blanket modules (TBMs) in the ITER were evaluated by varying the blanket thickness and the lithium-6 (Li-6) enrichment. Among the solid breeders, a helium-cooled solid breeder (HCSB) concept showed a better performance compared to a water–cooled ceramic breeder (WCCB) and a helium-cooled ceramic reflector (HCCR) concepts, while the helium-cooled lithium lead (HCLL) concept showed a better performance with a smaller inboard blanket thickness and a smaller major radius compared with a lead lithium ceramic breeder (LLCB) concept.

Keywords: *tokamak fusion reactor, systems analysis, neutronic constraint, breeding blanket*

**Corresponding author:* *bghong@jbnu.ac.kr*

Maximum Allowable Fluid Velocity and Concern on Piping Stability of ITER Tokamak Cooling Water System

P3-116

Seokho Kim*

US ITER, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA

The ITER facility is an international research project that is constructed in Cadarache, France. The main objective of ITER is to demonstrate the scientific and technical feasibility of a controlled fusion reaction allowing the production of 500 MW of fusion power for durations of several hundred seconds. US ITER is responsible for the design, engineering, and procurement of the Tokamak Cooling Water System (TCWS). Piping size in TCWS varies from DN40 to DN500. A design guideline was suggested that any piping smaller than DN150 is to have velocity below 6 m/s fluid while up to 9 m/s fluid velocity is allowed for the piping larger than DN150.

Fluid velocity affects various design parameters including pipe size, pressure drop, pump size, piping support, etc. When determining maximum allowable fluid velocity, one would consider erosion and corrosion, pump capacity, cavitation, hydrodynamic instability, and actual nuclear plant experience. The previous study performed to support the ITER piping design indicated that the hydrodynamic instability was a reason to limit maximum allowable fluid velocity below 6 m/s regardless of the piping size. However, the study found that buckling and fluttering because of hydrodynamic force would not result in the piping instability if it could be properly supported. Instead, a cavitation erosion may be a real issue that limits maximum allowable fluid velocity. The cavitation is a pipeline phenomenon that forms vapor cavities, which then go unstable and collapse violently in low-pressure, turbulent, flow-separation regions inside valves, elbows, pipe expansions, and other fluid handling components. Considering the cavitation issue, pressure drop, a typical practice in a nuclear plant system, it was advised to limit the fluid velocity below 6 m/s for the piping size smaller than DN150, and 9 m/s for the larger size. The paper describes a hydraulic stability analysis, generally accepted fluid velocity in nuclear plants, and cavitation-erosion limit in fluid velocity.

Keywords: *Cooling System Design, Piping Stability, Cavitation Erosion, Fluid Velocity*

***Corresponding author:** *kims@ornl.gov*

Design Updates of the Korean Fusion Demonstration Reactor Superconducting Toroidal Field Magnet System

P3-117

Hyun Wook Kim [a], Sangjun Oh [a], Hyun Jung Lee [a], Yong Chu [a], Sungjin Kwon [a], Hong-Tack Kim [a], Peter Titus [b], Keeman Kim [a,*]

[a] *National Fusion Research Institute, Daejeon, Korea*

[b] *Princeton Plasma Physics Laboratory, Princeton, NJ, USA*

A conceptual design study for a steady-state Korean fusion demonstration reactor (K-DEMO) was initiated in 2012. The superconducting magnet system is one of the key components of the K-DEMO and the preliminary study on superconducting magnet system was done in 2015. The superconducting toroidal field magnet system of K-DEMO consists of 16 coils which use two different types of internally-cooled Cable-In-Conduit Conductors (CICC). A high performance Nb3Sn strand will be used and able to provide 7.4 Tesla at the plasma center. For a validation of the magnet design, joints and current leads, thermo-hydraulic stability, Helium pressure drop and mechanical stability were considered. This work presents the conceptual design of K-DEMO TF magnet system.

***Corresponding author:** *kkeeman@nfri.re.kr*

Thermo-Hydraulic Analysis of the K-DMEO CS Conductor Depending on the Design Change

P3-118

H. J. Lee [a,*], H. W. Kim [a,], **Keeman Kim** [a], S. J. Kwon [a], Y, Chu [a], J. S. Kim [a], S. J. Oh [a], S. H. Hong [a], C. E. Kessel [b]

[a] *National Fusion Research Institute, Korea*

[b] *Princeton Plasma Physics Laboratory, Princeton, NJ, USA*

The K-DEMO superconducting magnet system is being developed for electricity realization in the fusion power demonstration reactor since 2012. The preliminary design of the conductor was introduced in 2015 and the design are being updated according to fusion program. The K-DEMO CS coil has been designed to adopt a pancake winding scheme and be able to generate sufficiently flux swing. In 2018, the coil design was modified to reduce the number of cooling channel. A central spiral hole is applied to the CS conductor to minimize the pressure drop in a long cooling channel and the shape of conductor has been changed to square accordingly. This paper presents the updated CS coil design and the thermo-hydraulic analyses performed on the conductor both a nominal operation and quench condition. A maximum temperature

margin and current ramp rate are studied to design of plasma scenario during blip and duration.

Keywords: *K-DEMO, CS conductor, thermos-hydraulic, quench*

**Corresponding author: yaeban@nfri.re.kr*

Design and Analysis of the Secondary Circuit of the DEMO Fusion Power Plant for the HCPB BB Option without the Energy Storage System and with the Auxiliary Boiler

P3-119

Leszek Malinowski [a], **Monika Lewandowska** [a]*, Evaldas Bubelis [b], Wolfgang Hering [b]

[a] *West Pomeranian University of Technology, Szczecin, Poland*
[b] *Karlsruhe Institute of Technology (KIT), Institute for Neutron Physics and Reactor Technology (INR), Eggenstein-Leopoldshafen, Germany*

The EU-DEMO (European DEMOnstration Fusion Power Plant) is being designed to produce fusion electricity at a level of several hundred MW, by about 2060. The Primary Heat Transfer System (PHTS) transfers heat from the fusion reactor heat sources, to the secondary Power Conversion System (PCS), which is responsible for the generation of the electrical energy. Several cooling concepts for the DEMO blanket and the related PHTS are considered. In some variants the Intermediate Heat Transfer System (IHTS) with the Energy Storage System (ESS) have been added between the PHTS and PCS, in order to mitigate transient effects resulting from the pulsed DEMO operation. In the present work a detailed GateCycle model of the steam/water PCS, for the option Helium Cooled Pebble Bed Breeding Blanket (HCPB BB) without the ESS is created and its operation at the nominal conditions (plasma burn) and at the reduced heating power (dwell period) is studied. The proposed circuit utilizes thermal power extracted from the four reactor heat sources, namely: Breeding Blanket with the First Wall cooled in series with the Breeding Zone, Divertor Plasma Facing Components, Divertor Casette and Vacuum Vessel, as well as the auxiliary boiler fueled with natural gas, which plays the role of the main heat source during the dwell phase. The proposed model of the PCS cycle is used to demonstrate possibility of stable operation of a DEMO plant without the IHTS/ESS during both, pulse and dwell phases. The study of the effect of the operating conditions on the cycle power and efficiency is also performed.

Keywords: *DEMO, Power Conversion System, Helium Cooled Pebble Bed Breeding Blanket, GateCycle*

**Corresponding author: monika.lewandowska@zut.edu.pl*

Optimization of CFETR Physics Design to Meet the Engineering Feasibility

P3-120

Guoqiang Li [a]*, Vincent Chan [b], Qingwei Yang [c], Ge Zhuang [b], Defeng Kong [a], Jinping Qian [a], Jiale Chen [a], Rui Ding [a], Jiangang Li [a], Yuanxi Wan [a,b] and CFETR Physics Design Team

[a] *Institute of Plasma Physics, CAS, Hefei, Anhui, China*
[b] *University of Science and Technology of China, Hefei, Anhui, China*
[c] *Southwestern Institute of Physics, Chengdu, Sichuan, China*

Chinese Fusion Engineering Test Reactor (CFETR) has finished the conceptual design, and in 2018 it started the engineering design stage. Physics design is the basis for engineering design and it must meet the engineering feasibility. In the past year, through iterations between physics design and engineering design, the major aspects of CFETR physics design have been optimized, including the plasma shape, divertor configuration, heating and current drive (CD) scheme, parameters of centersolenoid (CS) and poloidal flux (PF) coils. Recently we focused on the 1 GW fusion power operation of phase II. Two operation scenarios are developed, steady-state scenario and hybrid scenario. The scenarios aim to provide pulse length of at least 4 hours (~ the Tritium recycling time), so that CFETR can demonstrate Tritium self-sufficiency. The triangularity is set as 0.42, which is limited by the divertor components space constraint. The divertor configuration could be ITER-like divertor or snowflake+ divertor, and these two types of divertor configuration are compatible in one divertor structure. The position and size of the CS and PF coils are optimized to avoid the ports, form the magnetic configuration, and provide enough voltage-second for ramp-up and to sustain the hybrid scenario for at least 4 hours. Based on 1.5D transport simulations for scenarios and the limitation of mid-plane ports, the heating and CD schemes are selected as ~30 MW NBIs from mid-plane ports, and ~50 MW ECCD from the top ports. In addition, other heating methods such as lower hybrid wave and helicon wave are still open options for the heating scheme.

Keywords: *CFETR, physics design, engineering design*

**Corresponding author: ligq@ipp.ac.cn*

Investigation of Heat Transfer in a Steam Generator Bayonet Tube for the Development of PbLi Technology for EU DEMO Fusion Reactor

P3-121

Emanuela Martelli [a*], A. Del Nevo [b], P. Lorusso [c], F. Giannetti [c], V. Narcisi [c], M. Tarantino [b]

[a] *ENEA, Department of Fusion and Nuclear Safety Technology, Frascati, Rome, Italy*
[b] *ENEA, Department of Fusion and Nuclear Safety Technology, Camugnano (BO), Italy*
[c] *DIAEE Department, Sapienza University of Rome, Roma, Italy*

In the frame of the EUROfusion roadmap for the development of the DEMOnstration (DEMO) power plant, a research activity was carried out to develop a Lithium-Lead (PbLi)/water heat exchanger. The component should be capable to remove nuclear heat deposited in the liquid metal of the Dual Coolant Lithium Lead breeding blanket and feeding a steam turbine, ensuring an efficient thermal power conversion to electricity. One of the selected configurations is the steam generator bayonet tube. The HERO (Heavy Liquid mEtal pRessurized water cOoled tubes) test section is an experimental mock-up in a relevant scale of this steam generator, consisting of a bundle of seven double-wall bayonet tubes with a leakage monitor system. This test section, developed by ENEA at Brasimone R.C. and installed in the main vessel of the CIRCE (CIRColazione Eutettico) pool facility, aims to investigate the thermal-hydraulic features of the system, providing a database for thermal-hydraulic system codes validation.

An experimental campaign was carried out to demonstrate technological feasibility and performances of the prototypical heat exchanger, suitable as steam generator for the PbLi loop of the Dual Coolant Lithium Lead breeding blanket. A preliminary test analysis is realized with RELAP5-3D code in order to characterize heat transfer in liquid metal side, comparing results with experimental data and available correlations (i.e. Ushakov and Mikityuk). Moreover, the experimental data will be useful to validate RELAP5/mod3.3 code modified version in simulating heat transfer in PbLi heat exchangers. Considering the safety features of the component: i.e. the reduced possibility of water-lead/lead-alloy interaction, thanks to a double physical separation between them and easier control of eventual leakages from the coolant by pressurizing the separation region with inert gas, it is also exploitable as heat exchanger in the PbLi loop of the Water Cooled Lithium Lead breeding blanket.

Keywords: *DEMO, RELAP5, heat transfer, PbLi loop*

Corresponding author: *emanuela.martelli@enea.it*

Statistical Analysis of Tritium Breeding Ratio Deviations in the DEMO Due to Nuclear Data Uncertainties

P3-122

Jin Hun Park *, Pavel Pereslavtsev

Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Karlsruhe, Germany

For the stable and self-sufficient functioning of the DEMO fusion reactor one of the most important parameter that must be demonstrated is the Tritium Breeding Ratio (TBR). The reliable assessment of the TBR with safety margins is a matter of the fusion reactor viability. The global uncertainty of the TBR in the neutronic simulations includes many different aspects such as the uncertainty due to the simplification of the geometry models used uncertainty of the reactor layout, uncertainty introduced due to neutronic calculations etc. The last one can be reduced by applying high fidelity Monte Carlo simulations for TBR estimations. Nevertheless, these calculations have inherent statistical errors controlled by number of neutron histories, straightforward for global quantity like TBR underlying errors due to nuclear data uncertainties. In fact, every evaluated data file can be replaced with the set of the random data files representing the particular deviation of the nuclear model parameters, each of them being correct and valid for applications. To account for the uncertainty of the nuclear model parameters introduced in the evaluated data file a global Monte Carlo method can be used to analyze the uncertainty of TBR owing to the nuclear data used for calculations. To this end two 3D fully heterogeneous geometry models of the Helium Cooled Pebble Bed (HCPB) and Water Cooled Lithium Lead (WCLL) European DEMOs were utilized for the calculations of the TBR. The global Monte Carlo calculations were performed making use of the TENDL-2017 nuclear data library random files with high enough statistics providing well resolved Gaussian distribution of the TBR value. The assessment was done for estimation of the TBR uncertainty due to nuclear data for entire material compositions and for separate materials: structural, breeder and neutron multipliers.

Keywords: *DEMO, TBR, MCNP, Monte Carlo,*

Corresponding author: *jinhun.park@kit.edu*

Design and Comparison Study of Steam Generator Concepts and Power Conversion Cycles for Fusion Reactors

P3-123

Piyush Prajapati [a,c]*, Shrishail Padasalagi [b,c], Paritosh Chaudhuri [a,c], Dilshad Sulaiman [b], Shishir Deshpande [a,b,c]

[a] *Institute for Plasma Research, Bhat Gandhinagar, India*
[b] *ITER India and Institute for Plasma Research, Bhat Gandhinagar, India*
[c] *Homi Bhabha National Institute, Training School Complex, Anushakti Nagar, Mumbai, India*

The commercial viability of fusion technology would be established by ITER operation. The power extraction technology from the fusion reactors also needs attention in parallel so as to extract the power from the fusion technology. The power generation using fission technology is fully established, to explore the possibility to use some of the fission technology in fusion reactors. One of the technologies like steam generator have been considered in this paper, the detailed comparison study has been carried out on different steam generators being used in fission reactors. In this study the primary and secondary fluids are considered as helium and water respectively. Based on the comparison study, two types of steam generators have been identified as potential candidates one is Shell and Tube Heat Exchanger (STHE) another is Printed Circuit Heat Exchanger (PCHE). The design calculations have been performed for these Steam generators assuming 500 MWTh power from fusion reactor. The detailed analysis of the steam generator and also the comparative investigations of Brayton and Rankine power cycle along with their efficiencies shall be discussed in this paper.

Keywords: *Steam Generator Design, Power Cycles, Fusion Energy Extraction*

**Corresponding author: Piyush.prajapati@ipr.res.in*

The DTT Secondary Cooling Water Systems

P3-124

Alexander Rydzy *

ENEA, Department of Fusion and Nuclear Safety Technology, Frascati (Rome) Italy

The new Tokamak machine DTT, planned under construction by Enea Frascati Research Center, is a machine actively cooled by water. Although DTT is an intermittently operating machine, the thermal power that must be cooled is more or less 127 MW emitted within 100 seconds. Geographically the DTT site, at the Enea Frascati center, doesn't allow the construction of water basins and the cooling wet towers. Futhermore, it doesn't have enough water supply coming from the municipal aqueduct. Therefore, the best solution is to project a close loop cooling water system, divided into 2 circuits: primary circuits (Divertor, First Wall, ECRH, ICRH, NBI, Electrical Power Supply and Cryoplant) filled with demineralized water and a secondary circuit filled with cooling water designed for working with pressure under 16 Bars. The thermal Power transferred by the primary circuits using dedicated heat exchangers (plates or shell-and-tube) is delivered to a centralized "warm tank" developed in order to store all the energy emitted during the plasma discharge and to prevent the total water temperature in the tank from reaching boiling point. Afterwards, the warm fluid is transferred to another "cold tank" where the chillers are continously working between two successive machine pulses every 3600 seconds. The two tanks are designed to optimize the minimum power required by the chillers. Moreover a recovery energy system will be incorporated in order to heat all components (First Wall, Divertor, Vacuum Vessel) which should be maintaned warm between the two pulses. The same logic of centralization is applied to the demineralization with reverse osmosis.

**Corresponding author: alexander.rydzy@enea.it*

Complete Neutronic Analysis for the Edge Charge Exchange Recombination Spectroscopy in Equatorial Port of ITER

P3-125

Arkady Serikov [a]*, Luciano Bertalot [b], Maarten De Bock [b], Ulrich Fischer [a]

[a] *Karlsruhe Institute of Technology (KIT), Institute for Neutron Physics and Reactor Technology, Eggenstein-Leopoldshafen, Germany*
[b] *ITER Organization, St. Paul Lez Durance Cedex, France*

The ITER tokamak is a Nuclear Facility INB-174. Therefore, neutronics analyses must be provided for any ITER systems which possibly be irradiated. This paper presents the results of the neutronic analysis performed for an active diagnostic system installed inside the ITER Equatorial Port #3 (EP#3). The system is called the Charge Exchange Recombination Spectroscopy (CXRS) with a purpose to view the edge ITER plasma, so its full name is Edge-CXRS. The overarching goal of this analysis is to provide neutronic support for the design development of Edge-CXRS, accordingly to the rigorous Instructions for Nuclear Analyses established by the ITER Organization (IO). The technical content of this work (CAD design of the CXRS system, its material compositions, and models) is consistent with the approved ITER baseline. The newly developed MCNP model of CXRS has been integrated into the C-Model

2018 issue with the EP#3 modular design. Local approach has been applied for the CXRS modeling, assuming closing of all other neighboring diagnostics systems hosting in EP#3. The models geometry was converted with SuperMC. Radiation transport was performed with MCNP6, Shut-Down Dose Rate (SDDR) calculations – with R2Smesh and D1S-UNED. This work produced complete set of neutronics results requested for the IO Preliminary Design Review (PDR). The results included 3D maps of neutron and gamma fluxes, radiation loads on the Toroidal and Poloidal Field Coils around EP #3, nuclear heating inside the CXRS mirrors. The results of local SDDR calculated in EP#3 interspace indicated of suitability of the doglegs and labyrinths of the CXRS optical pathways and sufficiency of the EP#3 shielding performance. The CXRS design is still under development. Local approach leads to the relative, not absolute results, which will be updated in the final version.

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization (IO). This work has been funded by the IO service contract. This paper does not commit the IO as a nuclear operator.

Keywords: *ITER, fusion neutronics, diagnostics, radiation shielding, nuclear heating*

**Corresponding author:* *arkady.serikov@kit.edu*

Overview about Cryogenic Distillation Control and Safety Approach for Isotopes Separation Facility (ISF)

P3-126

Iuliana Stefan *, Ovidiu Balteanu, Ciprian Bucur, Mihai Vijulie, Nicolae Sofilca, Carmen Moraru, Liviu Stefan, Anisia Bornea, Marius Zamfirache

National Research and Development Institute for Cryogenics and Isotopic Technologies – ICSI Rm. Valcea, Rm. Valcea, Romania

Isotopes separation technologies are used in both TRF associated with CANDU reactors (to remove tritium from heavy water) and at fusion applications (eq. ITER ISS) to recover and enrich tritium for further operation. In Pilot Plant for Tritium and Deuterium Separation (PESTD), ICSI Rm. Valcea developed a cryogenic distillation system (CDS) which can be used as back-end for tritium recovery applications, for both front-end technologies, CECE for low tritium separation from water, LPCE for detritiation purpose. Cryogenic Distillation feed is a mixture of hydrogen's isotopes which presents hazards in operation (explosion and tritium contamination), therefore CDS of PESTD was designed to meet industrial and nuclear applicable standards. In this paper we present an overview of the CDS focusing on the process control and safety approach which meet specific conditions of Romanian codes and standards required for operation. However, specific information may be used later as reference for design consideration for other CDS. Cryogenic distillation process from ICSI Rm. Valcea is designed as four columns cascade inside a cold-box which acts as second barrier against hydrogen release and heat transfer protection. CDS process control is based on Programmable Automation Controller (PACs), Supervisory Control and Data Acquisition (SCADA) and specific low temperature field equipment and devices. Control room operator supervises the process using specific designed Human Machine Interface (HMI) which permit to monitor process parameters, alarms acknowledgment and acts to adjust the process or override normal PACs control.

Keywords: *tritium, cryogenic distillation, process control, fusion*

**Corresponding author:* *iulia.stefan@icsi.ro*

Advancements in CAD Implementation of EU-DEMO Water Cooled Lithium Lead Breeding Blanket Primary Heat Transfer Systems

P3-127

Andrea Tarallo [a]*, A. Del Nevo [b], E. Martelli [c], L. Barucca [d]

[a] *CREATE Consortium - University of Naples Federico II, Napoli, Italy*
[b] *ENEA Brasimone – Camugnano, Italy*
[c] *ENEA Frascati – Frascati, Italy*
[d] *ANSALDO Nucleare, Genova, Italy*

This work focuses on the 3D CAD implementation of the pipework and the main equipment of the Primary Heat Transfer System (PHTS) of Water-Cooled Lithium Lead Breeding Blanket (WCLL BB) EU-DEMO fusion power plant. During nominal plant operation the largest thermal reactor power is generated in the BB modules, while a minor contribution is provided by the Divertor and the Vacuum Vessel. Thermal power generated in BB First Wall (FW) is rejected by the FW PHTS loop to the Power Conversion System (PCS) through a molten salt Intermediate Heat Transport System. The latter is equipped with an Energy Storage System to allow for continuous operation also during the dwell time at a power level close to the burn phase. Thermal power generated in Breeding Zone modules is instead transferred by BZ PHTS directly to the PCS via a couple of Once-Through Steam Generators. Pipework layout of FW PHTS and BZ PHTS inside the Tokamak Building considered the major design constraints, such as the reinforced concrete structures and the dimensions of the main pieces of equipment. The size of the stainless-steel pipes was selected based on design operating pressure, temperature and primary coolant velocity. Results provided a first estimation of length and weight of the piping systems as well as their space occupancy. These data are useful for the development activities

related to: the plant integration, fluid process, safety and plant cost.

Keywords: *EU-DEMO; Water-Cooled Lithium Lead Breeding Blanket; Primary Heat Transfer System; Piping design*

***Corresponding author:** *andrea.tarallo@unina.it*

Preliminary Efficiency Analysis of Thermal-Electric Conversion System for China Fusion Engineering Test Reactor with Helium Cooled Solid Breeder Blanket

P3-128

Shuai Wang *, Shuang Wang, Qingqing Xu, Hongli Chen

Department of Engineering and Applied Physics, School of Physical Sciences, University of Science and Technology of China (USTC), Hefei, Anhui, China

China Fusion Engineering Test Reactor (CFETR) based on the tokamak approach with superconducting magnet technology is envisioned to provide 2000 MW fusion power in its latest design. In addition, CFETR has the goal to show electricity power generation to prove the possible industrial application of nuclear fusion reactor. A reasonable thermal-electric conversion system design is necessary for fusion reactor due to the more electricity consumption for plasma operation, and to provide the steady-state electricity production under the pulse plasma operation.

In this paper, a thermal-electric conversion system has been presented based on the Helium Cooled Solid Breeder (HCSB) blanket for CFETR, considering almost all the heat from blanket, divertor and vacuum vessel and to convert it into electricity as high efficiency as possible. Moreover, an Intermediate Heat Transfer System (IHTS) between Primary Heat Transfer System (PHTS) and Power Conversion System (PCS) is proposed to solve the power cycles to achieve long steady-state operation. Based on the suitable PHTS, IHTS and PCS design, the thermal-electric conversion system efficiency results, including the gross plant power and efficiency together with the net plant power and efficiency have been calculated.

Keywords: *CFETR, HCSB blanket, thermal-electric conversion system, net plant power, net plant efficiency*

***Corresponding author:** *wangs830@ustc.edu.cn*

Progress on Design and R&D of ITER Radial X-Ray Camera

P3-129

Kaiyun Chen, Liqun Hu *, Hongrui Cao, Shi Li, Jizong Zhang, Hong Yu, Xiuli Sheng, Jinlong Zhao, Luying Niu, Chaozhi Li, Nengtao Zhou, Shijun Qin, Zijun Zhang

Institute of Plasma Physics, Chinese Academy of Science, Hefei, China

A Great progress has been made to the final design and R&D of ITER Radial X-ray Camera (RXC). The design of external camera structure was simplified to facilitate the manufacture and maintenance. The cooling loop protecting detector from baking environment was optimized and new cooling scheme considering safety issues was studied. To provide nuclear shielding for camera electronics, R&Ds on shielding material B4C were made on processing technology and fixing structure. The design of camera electronics was optimized including pre-amplifier, mid-amplifier, chassis, integration and radiation hardness. The mockup of electronics was manufactured and test results supported the design. The design of Instrumentation and Control (I&C) system based on Control, Data Access and Communication (CODAC) Core System frame was updated and tests were completed on high-speed data acquisition, real-time long-cable signal transmission and control. In addition, efforts were made on development and study on advanced detectors which should be radiation-tolerant to survive harsh nuclear irradiation. The advanced detector studied is Low Voltage Ionization Chamber (LVIC) which is made of mainly planar metal electrodes and has the advantage of structure simplicity. The LVIC detector was designed considering ITER discharge scenarios, and the mockup has been manufactured and tested. Filled with argon gas of ~1 bar pressure, the detector has sensitivity close to theoretical value, indicating great application prospect. Based on the test results, it is expected the performance of LVIC can be defined in the second half of the year.

Keywords: *ITER radial x-ray camera, low voltage ionization chamber, boron carbide*

***Corresponding author:** *lqhu@ipp.ac.cn*

An Intelligent Controller Design Based on the Neuroendocrine Algorithm for Plasma Density Control System on Tokamak Devices

P3-130

Shuangbao Shu [a*], Jiarong Luo [b], Yuzhong Zhang [a], Xin Ke [c], Yang Zhang [a], Xiaojie Tao [a], Zhixue Cui [b], Chenyao Xing [a]

[a] *School of Instrument Science and Opto-electronics Engineering, Hefei University of Technology, Hefei, China*
[b] *College of Science, Donghua University, Shanghai 201620, China*
[c] *School of Electrical and Electronic Engineering, Huazhong University of Science and Technology, Wuhan, China*

Plasma electron density is one of the most fundamental parameters in the study of tokamak plasma physics. Realization of real-time, accurate and efficient control of plasma electron density is one of the key points to the long-time steady-state operation of Tokamak. Gas-pumping is a usual method for density feedback control. This paper takes the J-TEXT tokamak device as an example, uses the HCN laser interferometer to obtain the density signal, and controls the gaspumping valve through the intelligent control algorithm to achieve accurate the density control. Based on the mechanism model of the plasma density along with the neuroendocrine regulator principle, we design an intelligent controller with an ultrashort feedback link inside and study its application on the J-TEXT tokamak plasma density control system in this paper. The controller mainly includes a hypothalamic regulation module, a single-neuron proportion integration differential (PID) module, and an ultrashort feedback module. It is designed and referenced to the long feedback, short feedback, and ultra-short feedback loop mechanisms of neuroendocrine hormone regulation and abides by the principle of human neuroendocrine hormone regulation. The antagonistic hormone regulation module achieves rapid and stable elimination of errors through the fusion of enhanced regulation with regulation inhibition, and the single-neuron PID module enhances the adaptive and self-learning capabilities of the control system. The proposed controller makes the PID parameters of the control system be adjusted on-line in real time during the experiment. The simulation and experimental results show that the proposed method, compared to the conventional PID control method, can synthetically improve the performance of the control system. It makes a contribution to the development of the intelligent control system of Tokamak plasma density.

Keywords: *Tokamak device; Plasma density control system; PID; Neuroendocrine algorithm*

****Corresponding author:*** *shu@hfut.edu.cn*

Systems Engineering Challenges of IFMIF-DONES

P3-131

András Zsákai [a], A. Muñoz *[b], A. Diez [c], R. Román [d], E. Marco [b], M. Verdugo [b], A. García [d], A. Ibarra [d]

[a] *Wigner Research Center, Budapest, Hungary*
[b] *Empresarios Agrupados, Madrid, Spain*
[c] *VISURE Solutions*
[d] *CIEMAT, Madrid, Spain*

In the framework of the EU fusion roadmap implementing activities, an accelerator-based Li(d,n) neutron source called DONES (Demo-Oriented early NEutron Source) is being designed as an essential irradiation facility for testing candidate materials for DEMO reactor and future fusion power plants. DONES facility design is being developed within the EUROfusion workpackage WPENS, which main objective is to be ready for IFMIF-DONES construction as soon as 2020. Fourteen Research Units (RU) around Europe, as well as industry Third Parties, are involved working in different aspects of the DONES scope of works.

Taking into account the complexity of the facility and the geographical dispersion of the partners involved, it is of a paramount importance to properly develop the DONES Systems Engineering, creating and executing interdisciplinary processes to ensure that the DONES defined objectives are reached and that the facility fulfils the expected criteria.

This paper presents the Systems Engineering processes that are being developed for the IFMIF-DONES Project. A dedicated group has been set up to coordinate the identification, documentation and validation of Requirements and Interfaces (R&I) processes to ensure a successful design and subsequent construction of all Structure, Systems and Components. Requirements traceability is ensured by requirement matrices populated by System Responsible Engineers and controlled by Configuration Control and R&I group. The implementation of requirements tracking software is investigated. Interface control and configuration is ensured by a specially developed software adapted for the needs of the interdisciplinary workspace of the project.

Acknowledgments: This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Keywords: *systems engineering, IFMIF-DONES, interface, requirement,*

****Corresponding author:*** *amcervantes@empre.es*

Status of R&D on the Tritium Technology of the Fusion Tritium Plant at CIAE

P3-132

Zhizhen Liu [a], Xiangqun Lu [b], Qin Zhan [a], Qilin Yao [a], Yuqi Li [a], Hongguang Yang [a*]

[a] *China Institute of Atomic Energy, Beijing, China*

[b] *Beijing U. of Posts and Telecommmunications, West Tucheng Road, Beijing, China*

This paper presents the R&D on the fusion tritium technology at CIAE, such as the tritium processing, tritium analyzing and monitoring as well as the models and simulations of the key tritium systems of the tritium plant. The tritium instruments and equipments with ion chambers and proportional counter tubes had been calibrated with better measuring repeatability and accuracy in the tritium activities range corresponding to the tritium processing flow. High analysis sensitivity of tritium depth profile on the solid surface had been achieved with the self-developed BIXS (tritium beta-decay induced X-ray spectrometry).

In order to support the design of the CFTER tritium plant, the storage beds with metal hydride had been studied on the thermodynamic and dynamic behaviors, and the methods of the modeling and virtual simulation had been built for the fuel storage and delivery systems as well as the fuel delivering processes between the fuel storage beds and the fuelling systems.

Keywords: *tritium monitor, tritium calibration, tritium system, tritium plant, virtual simulation*

Corresponding author: *yanghg@139.com*

Local Safety System of the Neutral Beam Injection for W7-X

P3-133

Ralf Schroeder [a], Y. Drider [a], D. Hartmann [a], B. Heinemann [b], R. Kairys [a], P. McNeely [a*], W7-X Team [a]

[a] *Max-Planck-Institut für Plasmaphysik-Teilinstitut Greifswald, Wendelsteinstrasse, Greifswald, Germany*

[b] *Max-Planck-Institut für Plasmaphysik, Garching, Germany*

The Neutral Beam Injection (NBI) control system on the W7-X Stellerator consists of a standard PLC based on PCS7 from Siemens AG and a separate safety controller based on PNOZmulti from Pilz GmbH. The requirements of the NBI safety control are derived from an analysis of the specific personal and plant hazards in accordance with the international standards IEC 61508 / IEC61511. Particular challenges are the usage of high voltage, open high frequency and high power sub systems, explosive gases that are distributed over four locations and interfaces to the central safety system of W7-X.

Firstly, the paper describes the hardware structure of the NBI safety system. The configurable safety controller PNOZmulti was chosen since it can implement safety levels up to SIL3. Next the paper, as an example, describes in detail the technical solution for access to the high voltage area of the NBI injector box. This example shows that higher SIL requirements (equal or greater than SIL2) cannot always be technically implemented and therefore need to be secured by additional organizational security measures. The example further demonstrates the complexity of the documentation and validation necessary to meet the required safety level.

Keywords: *Safety, SIL, Control system*

Corresponding author: *ralf.schroeder@ipp.mpg.de*

Nanoscale Structure Damage in Irradiated W-Ta Alloys for Nuclear Fusion Reactors

P3-134

Iuliia Ipatova [a*], R.W. Harrison [b], E. Jimenez-Metero [c]

[a] *Nuclear Futures Institute, Bangor University, Dean Street, Bangor Gwynedd, UK*

[b] *School of Mechanical, Aerospace and Civil Engineering, Materials Performance Centre, The University of Manchester, Manchester, UK*

[c] *Materials Performance Centre, School of Materials, The University of Manchester, Manchester, UK*

The attractiveness of tungsten as structural material lies in its high resistance to plasma-induced sputtering, erosion and radiation-induced void swelling, together with its thermal conductivity and high-temperature strength. The brittle nature of tungsten hampers the manufacture of reactor components and can also lead to catastrophic failure during reactor operations. Alloying tungsten with tantalum and considering alternative compositions to substitute tungsten are two potential routes to improve the structural performance of tungsten and, hence, to extend the temperature window of W-based alloys for safe reactor operation.

In this research, W, Ta and their mutual alloys were analyzed in a transmission electron microscope (TEM) after proton irradiation (ex-situ) and directly during irradiation exposure (in-situ). Materials under investigations were exposed to proton irradiation either of 3 MeV (ex-situ) or 40 KeV (in-situ) energies using Pelletron ion accelerator placed at the Dalton Cumbrian Facility and MIAMI-1 system in the University of Huddersfield respectively. Interstitial-type a/2 <111> dislocation loops form under irradiation, and their size increases in W-5Ta, where loop saturation takes place. In contrast, the loop length in W increases progressively. The dislocation loops and networks observed in both materials at later stages act as effective hydrogen trapping sites, so as to generate hydrogen bubbles and surface blisters.

In-depth in-situ/ex-situ transmission electron microscopy analysis of

the structural damage caused by proton exposure, revealed the similar tendency of hindrance of the lattice defect development in the binary alloys with respect to the pure metals. This effect relates to both the evolution of radiation-induced dislocation structures in W, and consequently the appearance of hydrogen blisters.

Keywords: *tungsten alloys, materials for fusion, radiation damage, faceted voids, transmission electron microscopy*

Corresponding author: *i.ipatova@bangor.ac.uk*

Verification and Validation of the GEANT4 Monte Carlo Code Toolkit for DEMO Neutronics Applications

P3-135

Elena Nunnenmann *, Ulrich Fischer

Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany

The high-energy particle physics Monte Carlo code toolkit GEANT4 has been expanded for fusion energy-range neutron transport simulations based on evaluated nuclear cross-section libraries. In a previous work, we started to investigate GEANT4's suitability for fusion neutronics applications focusing on basic neutron transport. Following up, this paper presents benchmarking against an engi-neering design oriented breeder blanket experiment and nuclear analyses in comparison with MCNP of the DEMO reactor equipped with a current HCPB (Helium Cooled Pebble Bed) breeding blanket. The nuclear data were taken from the JEFF-3.3 library. The HCPB Tritium Breeder Module Mock-up experiment, previously conducted at the Frascati Neutron Generator (FNG), was selected as suitable benchmark experiment. Experimental results and input data required for the calculations were taken from the Shielding Integral Benchmark Archive and Database (SINBAD). For the GEANT4 geometry description, the available MCNP input file was converted into CAD format, processed with the SpaceClaim software, and converted into GDML format with the McCAD conversion tool. The data describing the FNG neutron source in SDEF MCNP input format were processed and sampling implemented in a GEANT4 subroutine. In the paper, the results of the GEANT4 simulation are compared with measured data and results of MCNP calculations. For the DEMO analyses, the CAD geometry description was again converted into GDML using SpaceClaim and McCAD. The plasma neutron source description was converted from the available MCNP Fortran90 subroutine into a GEANT4 C++ subroutine. The paper presents the GEANT4 results in comparison with MCNP.

Keywords: *Neutronics, GEANT4, HCPB, SINBAD, DEMO*

Corresponding author: *elena.nunnenmann@kit.edu*