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BOOK OF ABSTRACTS

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KN 1 ABSTRACT-9cc7

F. Nuclear System Design

The ITER Project: progress amid challenges

Juan Knaster

Fusion for Energy

The ITER project is demonstrating important advancements in fusion engineering and associated technology through the manufacturing and delivery of unique components (toroidal field, poloidal field, central solenoid and correction superconducting magnets, vacuum vessel sectors, unprecedented powerful heating systems and many other sophisticated equipment), as well as through tokamak assembly, support system installation and early stage system commissioning. The cooling water system and cryogenics plant are in sub-system commissioning, all components of the cryostat are complete, and magnet conversion equipment installation is well along. This progress report will describe progress on the ITER site as well as substantial contributions from all ITER Members.

The project has identified challenges with manufacturing of key components: specifically, vacuum vessel sectors and thermal shields. A project-wide coordinated effort is in place to repair or replace these components, as appropriate. Lessons learned related to fusion engineering design, manufacturing, and assembly will be carefully evaluated, as these lessons will be relevant to global fusion R&D efforts.

In parallel, the project is reviewing its baseline schedule and approach for assembly and operation, considering the impacts of the Covid-19 pandemic and previous manufacturing challenges, working closely with the French safety regulator, and evaluating potential adjustments that could offset these schedule impacts. Adjustments under evaluation include additional testing of components before installation, optimization of some machine aspects, and a shift from “first plasma” to a more complex first experimental campaign. The ITER project goal remains the achievement of DT operations, to demonstrate sustained self-heated fusion power. An updated baseline proposal will be submitted to the ITER Council in 2024.

As the project progresses, ITER’s experience with design, fabrication, and assembly of components is receiving constant attention from public and private fusion projects who hope to use ITER’s experience to benefit and inform their ongoing work.

Keywords

It is a keynote speak on ITER.

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KN 2
H. Models and Experiments for FNT

Recent progress and plans for fusion program in China

Yuntao Song

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Recently, a 403s H-mode operation and a 10 MW high power discharge were achieved on EAST tokamak, whose auxiliary heating systems and divertor were significantly upgraded in 2021. Long-pulse H-mode operation with H_{98y2}>1.0 and 100s 10 MW operation will be the next target. Important progress have also been made in more Chinese tokamak facilities, as HL-2M and J-TEXT.

After the engineering design of the Chinese Fusion Engineering Testing Reactor (CFETR), the construction of the Comprehensive Research Facility for Fusion Technology (CRAFT) facilities was started, which is used to explore the key R&D technologies for CFETR. The superconducting magnet research system and the divertor research system are the two main systems of the CRAFT. Developing Nb_3Sn/NbTi hybrid magnet and CORC type high-temperature superconducting magnet that can reach 15 T magnetic field are the main tasks of the superconducting magnet research system, on which the large-scale CICC conductors for CFETR can be tested. In order to mimic the realistic working conditions of the CFETR plasma-facing components, especially the divertor, a large linear plasma testing facility that can be operated steadily for 1000 s with particle fluxes higher than 10^{24} $m^{-2} \cdot s^{-1}$ is being built.

In addition, the manufacturing of the CFETR 1/8 vacuum vessel prototype has been completed. In order to bridge the gaps for ITER and CFETR with compact high field advanced performance for steady-state operation, a new machine called Burning plasma Experimental Superconducting Tokamak (BEST) is aimed to be built on which the burning plasma physics, tritium inventory technology of the CFETR and future fusion reactor can be investigated. The key parameters that BEST needs to achieve include plasma current Ip=4~7 MA; major radius R=3.6m; toroidal field Bt=6.15 T; fusion power P_F= 20~200 MW.

Meanwhile, more alternative fusion research facilities in China like HL-2M, SUNIST, J-TEXT, KTX et al. and the status of ITER package activities in China will also be introduced in this talk.

Keywords

EAST, tokamak, H-mode.

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KN 3 ABSTRACT-9aba
F. Nuclear System Design

Recent progress and plans for Korea’s fusion program

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The development of fusion energy in Korea began officially in 1995 according to the National Fusion R&D Plan. There are now several important fusion programs running in Korea under the Fusion Energy Promotion Act enacted in 2006 such as KSTAR, ITER, DEMO, basic R&D, and international collaboration programs.

The fusion experimental device, named Korea Superconducting Tokamak Advanced Research (KSTAR), built in 2007, has achieved many excellent experimental results including a long pulse operation with a plasma ion temperature of 100 million degrees for more than 30 seconds. The main goal of this device is to develop a steady-state operation scenario for the fusion reactor. Korea participated in the ITER project as a full member in 2003, which has played an important role with the KSTAR in accelerating the Korea’s fusion program for the Korean DEMO.

The technical concept of the Korean DEMO was officially defined by National Fusion Energy Committee in February this year. Based on the established concept, the roadmap for Korean DEMO will be officially prepared by the end of this year and its conceptual design will also begin within this year.

In this presentation, including the above, the latest progress and plans for Korea’s fusion program will be introduced in more detail and also its prospects will be addressed.

Keywords

Korea, fusion, program, KSTAR, ITER, DEMO.

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KN 4 ABSTRACT-2ce1
J. Burning Plasma Control and Operation

Recent Progress and Plans in the EUROfusion Program

Ambrogio Fasoli

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Established during the Horizon 2020 period, EUROfusion has made substantial progress in integrating various fields of science and technology, academic and industrial approaches, and systems required for a fusion power plant.

A key contribution has been the coordination of national facilities. In addition to JET, EUROfusion researchers jointly use TCV in Switzerland, ASDEX-U in Germany, MAST-U in the UK, and WEST in France. A second example is the creation of advanced computing hubs in Finland, Poland, Germany, Switzerland, and Spain, exploiting new synergies with applied mathematics, computer science, imaging and artificial intelligence.

EUROfusion has recently entered the Horizon Europe framework (2021-2027), during which new devices as JT-60SA, COMPASS-U, DTT and ITER will start or approach first operation, and the W7-X stellarator will start operating with an actively-cooled divertor. Construction of IFMIF-DONES will begin, and the conceptual design of DEMO will be in fully swing.

The latter constitutes a cornerstone in the acceleration of fusion developments, motivated by the recent results on JET, NIF and elsewhere, a renewed perception of the urgent need for a sustainable source of baseload electricity, and the ensuing interest of politics, industry and private investors.

The major challenge in fusion resides in the integration of all individual physics and technology issues into an economically competitive power plant. DEMO will prove that such integration is possible, providing net electricity production, self-sufficient fuel cycle, and minimal waste production. To develop DEMO by mid-century requires us to proceed in parallel with ITER, drawing crucial lessons from it through all of its phases, from design to assembly, first operations, and full power DT campaigns.

The presentation will summarise EUROfusion achievements and discuss the challenges ahead, including how to approach public-private partnerships, in view of a development of DEMO based on fully industrial practices.

Keywords

EUROfusion, progress, plans, roadmap, DEMO.

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PLENARY SESSIONS

PL 1 ABSTRACT-df24
H. Models and Experiments for FNT

Key technological aspects of recent DT operations at JET

Rosaria Villari[1], P. Batistoni[1], X. Litaudon[2], L. Packer[3], Z. Ghani[3], N. Fonnesu[1], A. Peacock[3], L. Jones[3], S. Reynolds[3], M. Dentan[4], R. Vila[5], N. Terranova[1], J. Figueiredo[6], V. Naulin[7], J. Alguacil[8], S. Bradnam[3], J. P. Catalan[8], M. Cecchetto[9],A. Colangeli[1], D. Croft[3], M. Fabbri[10]

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A unique high-performance Deuterium-Tritium (DTE2) campaign was conducted at JET tokamak in 2021, producing a total of $8.5x10^{20}$ 14.1 MeV neutrons with more than 10^{20} neutrons in one day. An outstanding amount of nuclear fusion relevant data are collected in the frame of EUROfusion technological exploitation of JET DT operations started within JET3 program and currently ongoing within the Preparation of ITER Operations (PrIO) project. Several technology-oriented experiments and analyses were carefully prepared and performed aimed at improving the knowledge of nuclear technology and safety, developing, and validating nuclear codes, data, and experimental techniques to reduce the risks to ITER operations and maintenance by taking advantage of DT operations at JET. The technological exploitation of JET DTE2 comprise activation measurements and analyses of real ITER material; functional materials damage studies; neutron streaming and shutdown dose rate benchmark experiments; test of neutron and tritium detectors for breeding blankets; verification of 14 MeV neutron calibration; collection and processing of data of Occupational Radiation Exposure (ORE) and waste. Further experiments have been prepared for the forthcoming DT campaign (DTE3) at JET aimed to perform careful study and validation of the predictions on water activation in real tokamak cooling loops, neutron induced Single Events Effect (SEE) on electronics and testing of shielding performance of flexible neutron shields. This paper presents the main results and achievements, the lessons learnt and the identified issues as well as the potential implication to the ITER program and operation and key outcomings for DEMO design.The development of fusion energy in Korea began officially in 1995 according to the National Fusion R&D Plan. There are now several important fusion programs running in Korea under the Fusion Energy Promotion Act enacted in 2006 such as KSTAR, ITER, DEMO, basic R&D, and international collaboration programs.

Keywords

JET, nuclear, neutronics, safety, technology, DTE2, DTE3.

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PL 2 ABSTRACT-a14d
F. Nuclear System Design

DEMO-Related Design Activities in Europe

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The design of DEMO in Europe benefits largely from the experience gained from the design, licencing, and construction of ITER, which remains the crucial machine for the validation of the DEMO physics and part of the technology basis. However, work done in the Pre-Concept Design Phase and the Gate Review revealed that the DEMO design and operating space is heavily constrained by physics and technology and technological challenges beyond ITER remain paramount.

A series of technical assessments were recently launched by the DEMO Central Team to revisit the process to define the optimum DEMO design space, aiming to minimise either the machine size or the technical risks, and to facilitate an earlier DEMO deployment. This included studies to assess (1) the impact of varying some of the stakeholder requirements; (2) the impact and associated risks of using high-field Toroidal Field magnets, benefiting from promising High Temperature Superconductors; and (3) machine configurations at low aspect ratio.

The outcome of this work has clearly highlighted the DEMO operation risks arising from the lack of relevant complementary facilities to validate the performance and reliability of some of the most relevant DEMO system technologies, i.e., the breeding blanket. To mitigate these risks, the DCT is considering options for a plasma volumetric neutron source to serve as facility for testing and qualifying crucial nuclear fusion technology components in parallel to DEMO design and construction. This facility would be complementary to both ITER, which is focused on burning plasma physics, and to DONES, which is focused on large dpa in small material samples. Aside from reducing the DEMO technical risks, the facility would also eliminate the requirement for high-fluence, transforming DEMO from a 'qualification' device into a fusion power demonstrator, resulting in an accelerated effective deployment.

The results of this work are presented in this contribution.

Keywords

DEMO, blanket, nuclear testing.

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PL 3 ABSTRACT-83c9

J. Burning Plasma Control and Operation

An overview of the challenges and opportunities of private-sector fusion

Dennis Whyte

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The last few years have witnessed the emergence of multiple companies, supported by private investment. These companies are pursuing a wide variety of approaches with respect to achieving the plasma conditions that achieve sufficient energy gain and/or ignition. This presentation will provide a high-level overview of several of these approaches including their stated strategy, and their associated scientific and technical challenges. One of the most advanced projects, the SPARC tokamak under construction by Commonwealth Fusion Systems with MIT collaboration, will be highlighted. SPARC is designed to achieve 150 MW of D-T fusion with a plasma energy gain Qp~10, and will be operational by mid-decade. Progress on the site construction and technical advancements will be provided. Finally the opportunities and challenges in fusion science and technology provided by this emerging effort will be discussed, particularly the need for private-public partnerships that will be required to support and accelerate fusion..

Keywords

Energy gain, burning plasma, commercial fusion.

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PL 4 ABSTRACT-f209

K. Inertial Confinement Fusion Studies and Technologies

ICF current status after ignition on NIF: overview of worlwide initiatives towards an IFE power plant

Alexis Casner

CEA CESTA

The historic achievment of ignition for the indirect drive Inertial Confinement Fusion approach on the NIF in December 2022 had a tremendous impact. Together with recent progresses on various Magnetized Confinement Fusion devices, it contributes to put fusion power in the headlines and on political agendas worldwide. However how far are we from a First-of-a-Kind Inertial Fusion Energy power –plant ? What are the remaining physical and technical challenges and show stoppers ? After a quick recall of the scientific path followed by LLNL colleagues to achieve ignition, I will underline the advantages / drawbacks of the laser direct drive approach compared to the successful indirect-drive one. While more favourable in terms of potential achievable gain, the academic direct-drive approach suffers from a lack of representative, state-of-the-art laser facility worlwide with a deliverable laser energy in excess of 30 kJ. The OMEGA laser facility is the only-one devoted to such experimental studies with an upgrade envisioned at the end of the decade. Indeed, NIF and LMJ laser configuration, and their cryogenic capabilities are optimized for indirect drive only. Another noticeable difficulty would be to increase the shot (experiments) rate from one a day with cryogenic capabilities and almost perfectly engineered targets to a few shots per second with mass-producted targets. This would also require moderate to high repetition rate kJ laser modules and the relevant final end optics sustaining the laser fluence. This may look unrealistic, while being enginneering issues rather than physical ones. Indeed, numerous recently founded private laser-fusion companies claim bringing electricity to the grid sooner rather than later. I will try to draw an unbiased, scientific perspective of their various approachs, while being enthusiastic about the bright perspectives ahead for any ICF scientist.

Keywords

Inertial Confinement Fusion, Inertial Fusion Energy.

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PL 5 ABSTRACT-d968
H. Models and Experiments for FNT

Overview of Broader Approach Activities

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The Broader Approach (BA) activities aim to complement the ITER project and early realization of fusion energy through research, development, and tests of technologies supporting the future demonstration fusion reactor (DEMO). These activities are implemented under the BA agreement, which was signed and ratified in 2007 between Euratom and the Government of Japan. In essence, the BA activities consist of three projects:

a) The Satellite Tokamak Programme Project JT-60SA, the world's largest superconducting tokamak until ITER starts, aims to support the assembly, commissioning and preliminary operation of ITER, and carry out demonstration and optimisation of steady-state operation of advanced plasma configurations for DEMO;

b) The International Fusion Energy Research Centre (IFERC), which comprises three different sub-projects: the DEMO Design Research and Development Coordination Centre to coordinate design and R&D on materials and components for DEMO, the Fusion Computer Simulation Centre for the simulations of fusion plasmas, the analysis of experimental data, modelling of ITER operation, and contribution to the design of DEMO, and the Remote Experimentation Centre to allow scientists to participate remotely in fusion experiments from its control room in Japan;

c) The Engineering Validation and Engineering Design Activities for the International Fusion Materials Irradiation Facility (IFMIF/EVEDA) focused on the design and validation of key components needed for the future Fusion Neutron Source facilities to characterize materials envisioned for DEMO.

After 13 years of fruitful collaboration, Euratom and Japan launched the second phase of the BA activities in 2020. It is focused on exploiting and enhancing the facilities that have already been built and on working more closely than ever with ITER as the latter advances towards the first plasma. This paper will present a synthesis of the achievements already obtained thanks to the three BA projects and will address the future developments planned in 2023 and beyond.

Keywords

Broader Approach, JT60-SA, IFMIF-EVEDA, IFERC.

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PL 6 ABSTRACT-cc10
A. Plasma-Facing High Heat Flux Components

Key features of KSTAR's new tungsten divertor

Woong Chae Kim[1], Soo-Hyeon Park[1], Sang Woo Kwag[1], Kwang Pyo Kim[1], Hwang-Rae Cho[2], Jong-Man Lee[2], Siwoo Yoon[1]

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As the plasma performance and pulse length in KSTAR have increased, the usefulness of the current graphite divertor of open geometry and contact cooling structure has reached its limit.

Therefore, it was decided to build a new divertor of closed geometry with a tungsten monoblock of active cooling structure. This new divertor can accommodate various plasma shapes and facilitates a detached divertor with a high-power, long-pulse operation.

The new divertor can operate at a heat load of 10.0 MW/m^{2,} while the current divertor can operate at a heat load of 3.5 MW/m^2. The new one has a cassette structure and consists of 64 cassettes. The tungsten monoblock consists of a tungsten block, an oxygen-free copper interlayer, and a copper alloy tube with a swirl tube attached. Prototype modules have passed the thermal stress test with 10 MW/m^2 10-sec pulses 1000 times and 20 MW/m^2 10-sec 500 times.

The cassette is hot-helium leak tested at 250°C, and pressure drop tested to verify the flow rate. The assembly tolerance of the cassette is ±0.25 mm, and the overall accuracy after installation is ±1.0 mm.

As of February 2023, the cassette modules are in production, and preparations for installation work in the vacuum vessel are underway. We expect the installation to be completed by the end of July 2023.

This presentation will report on the main features of the design, fabrication, installation, and final commissioning results after installation.

Keywords

KSTAR, Divertor, Tungsten Monoblock, Cassette.

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PL 7 ABSTRACT-1413
F. Nuclear System Design

Progress of Research and Development activities toward Fusion DEMO Reactor at QST in Japan

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The progress and the current status of research and development (R&D) activities toward a fusion DEMO reactor at the National Institutes for Quantum Science and Technology (QST) in Japan is reported. In order to advance the Japanese DEMO activity, not only Japanese domestic activity but also international collaborations of ITER, ITER-related activities and Broader Approach (BA) activities are conducted in QST.

In ITER and ITER related activities, while proceeding with ITER procurement activities, how to effectively utilize ITER is discussed continuously toward DEMO reactor, including ITER-TBM (Test Blanket Module) program. Also, in the BA activities under the collaboration between EU and Japan, following three projects are carried out for a fusion DEMO reactor in QST. There are, 1) satellite Tokamak (JT-60SA) project to complement ITER and aim for efficient plasma operation for a DEMO reactor, 2) IFMIF/EVEDA project for preparation of fusion neutron source for blanket materials test, and 3) IFERC project for DEMO reactor design & related R&D using a supercomputer system & a remote experimentation system.

Furthermore, this year, the government passed a Cabinet decision on the "Basic Policy for Realizing GX," and in the reference document, the fusion DEMO reactor was positioned as one of the next-generation innovative reactors, and the importance of accelerating development. Based on the government discussions, QST will strengthen cooperation with more companies, universities, and research institutes toward the DEMO reactor, aiming for early social implementation of individual nuclear fusion technologies. In particular, Japan is 100% dependent on imports for lithium and beryllium, which are blanket functional materials, and QST is working on further cooperation with companies to secure these resources.

Keywords

KFusion DEMO, ITER, JT-60SA, International Fusion Materials Irradiation Facility, Advanced Fusion Neutron Source, ITER Test Blanket System, ITER Remote Experimentation Center, Green Transformation (GX).

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PL 8 ABSTRACT-ffe1
G. Safety Issues and Waste Management

Safety approach for future fusion power plant

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The current international, European, and national approach to nuclear safety regulation and safety rules have been shaped by the risks associated with nuclear fission based power plants (NPPs) for which the technologies have been developed over decades. Although NPPs have a high hazard potential, robust safety led engineering with defense-in-depth ensures that the risk to the public is as low as possible.

Fusion power plant (FPP) design is in its infancy with several design options under consideration, hence there is a need for flexibility in the requirements and regulatory approach to be applied whilst maintaining the fundamental safety goals of protecting workers, the environment, and members of the public. To date no fusion power plant design has been subject to regulatory scrutiny but fusion based research facilities such as JET and ITER have been subject to safety regulation (UK industrial-radiological and environmental regulation for JET and French nuclear regulation for ITER).

When considering what regulatory approach should be applied to FPPs, a number of factors need to be taken into account because a regulatory framework is dependent on a country's legal system. In some countries the legal framework allows for regulation to be based upon a goal setting approach, where the law, via the regulators, set the safety goals that must be achieved and the FPP operator is free to show how the design of the FPP and its operation will meet the goals. Other countries have legal frameworks that require a prescriptive approach to regulation where the law prescribes what the FPP designer and operator must do to ensure safety. For emerging technologies where there are several competing technology approaches, flexibility and specific adaptation of codes and standards are needed to accommodate the development of the various approaches met in fusion machine.

The experience of ITER in France has shown that the application of a non-prescriptive goal-setting regulatory approach is appropriate for FPPs, and is preferable to prescriptive approach.

From safety engineering point of view, the implications of the licensing process are discussed and possible adaptations of the licensing process to the safety challenges and specificities of the future fusion power plants are proposed.

Keywords

Safety, ITER, FPPs.

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PL 9 ABSTRACT-33c4
A. Plasma-Facing High Heat Flux Components

The role of the DTT facility for the development of high heat flux tokamak components

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The heat exhaust in fusion devices is one of the main challenges towards the realization of fusion as an energy source. The heat generated by fusion reaction after crossing the last closed magnetic surface flows in a narrow layer to the divertor where it can reach values of 60MW/m2, similar to the heat load at the surface of the Sun. For this reason the European Fusion Roadmap explicitly lists the heat and particle exhaust as one of the seven mission of fusion research.

The Divertor Tokamak Test (DTT) facility is specifically designed to address this challenge. DTT has been designed to produce plasma conditions in the divertor similar to those in ITER and DEMO, through a substantial amount of additional heating (up to 45MW) to achieve DEMO relevant heat loads. DTT (R=2.19m, a=0.70m, B=6T, I_p=5.5MA) will investigate alternative solutions to the heat exhaust by producing: a) magnetic configurations that allow large divertor wetted areas; b) partially detached plasma conditions; c) advanced plasma facing components technology; d) impurity seeding to increase core radiation; and, e) liquid metals for plasma facing component. A specific attention will be devoted to find integrated core-edge solutions such e.g. regimes with simultaneously high radiated fraction and good confinement. To guarantee such a broad program, the design has ensured flexibility, accessibility, and a sophisticated diagnostic system. At the same time DTT will use DEMO relevant solutions for the plasma facing component and will be the only device in the world of the 5MA class equipped with full tungsten actively cooled components.

The facility is under construction. About 200Meuro have been committed in industrial contracts and the permit has been obtained from the licensing authority. The status of the construction, schedule and preparation of the research program will be presented.

Keywords

Fusion, heat exhaust.

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PL 10 ABSTRACT-6494
H. Models and Experiments for FNT

Overview of IFMIF-DONES: an irradiation facility relevant for fusion materials

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The International Fusion Materials Irradiation Facility – Demo Oriented Neutron Source (IFMIF-DONES) is a research infrastructure for irradiation the materials to be used 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 of future fusion reactors.

Materials irradiation data under such conditions are of fundamental interest for the fusion community to consolidate the fusion reactors engineering design and licensing and to validate modelling tools for materials radiation damage. The facility will be also able to address some tritium technologies related experiments.

Its Construction Phase just started in the proposed site in the Escúzar Metropolitan Park (located in the Granada 18 km southwest from Granada city).

This paper will present an overview of the implementation, engineering design and main irradiation characteristics of the facility.

Keywords

Fusion neutron source, fusion materials, radiation effects.

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PL 11 ABSTRACT-a004
B. Blanket Technology

Status of the ITER TBM Program and overview of its technical objectives

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The ITER TBM Program foresees that two ITER equatorial ports are dedicated to the operation and testing of mock-ups of four different concepts of tritium breeding blankets (TBB) ensuring tritium breeding self-sufficiency for a demonstration fusion reactor (DEMO). The in-vessel part is called Test Blanket Module (TBM). Lithium-ceramics pebble-beds and liquid lithium-lead are the two breeder materials used in the TBMs. Each TBM is part of a Test Blanket System (TBS) that includes also the various associated sub-systems needed for its independent operation. They are partly located in two equatorial port cells and partly in other rooms of the ITER tokamak complex.

This paper starts with the description of the main features of the four TBSs present in the so-called "first TBS configuration" that will be installed and operated in the first ten years of the ITER operations. It addresses that the design various TBS components and sub-systems, such as TBMs, cooling systems (helium or water), tritium extraction systems, coolant purification systems, tritium accountancy systems, and various measurement systems. It addresses also the main associated R&D aspects that are being performed or planned by the procuring ITER Members.

The second part of the paper will discuss the objectives and expected technical achievements for the TBM Program in support of the development of the corresponding DEMO breeding blankets. They include not only the results of the various tests that are planned during the TBS operations (related to the demonstration of the feasibility of the tritium breeding self-sufficiency) but also the lesson-learned and associated information obtained from the present design and R&D activities, such as the development of the TBM structural materials, the approach to the tritium management, the most relevant TBS safety aspects, the development of measurements and control systems and the assessment of TBS components reliability/availability requirements.

Keywords

TBM, breeding blanket, tritium, TBS.

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PL 12 ABSTRACT-acc3

A. Plasma-Facing High Heat Flux Components

Key technology developments for high performance Steady-State tokamak Fusion Reactor in China

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Advanced high-performance Steady-State operation is need for future fusion reactor. Among all the technical challenges for tokamak fusion reactor, high performance superconducting magnets, H&CD system, actively cooled W divertor and advanced integrated control are the key issues. NbTi EAST superconducting magnets, 34 MW CW H&CD systems, and 3 generation actively cooled divetors have been developed on EAST. With the integration of ST magnets, H&CD, PWI and advanced control, up to 1000 super-I mode and 300s H-mode have been obtained on EAST tokamak. Detail of EAST superconducting magnets, H&CD system, actively cooled W divertor, and integrated advanced control will be given in this talk.

Comprehensive Research fAcility for Fusion Technology (CRAFT) is a R&D project for Chinese Fusion Engineering Testing Reactor (CFETR). Progresses of design and R&D for CFETR superconducting magnets which consist TF and CS, PF, 4 H&CD (ICRF, ECRH, NBI and LHCD) systems, and new fully actively cooled W divertor with high heat load handling technology and experiences for long pulse testing for CFETR will be given. And finally, magnets, H&CD and Divertor of Burning plasma Experimental Superconducting Tokamak (BEST) project will be introduced. BEST is aiming Q>1 for steady-state operation (>1000s) and Q=5-10 (10s) for short pulse with DT fuels.

Keywords

Divertor, Magnet, Advanced control.

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PL 13 ABSTRACT-fd81
D. Material Engineering for FNT

Status of design considerations of the US Fusion Prototypic Neutron Source

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Fusion power plants (FPP) will require structural and plasma-facing materials with sufficient dimensional stability and resistance to neutron degradation of thermomechanical and physical properties. FPP design activities will thus require models that can predict time-dependent materials performance within acceptable uncertainty bounds. Materials deployed in an FPP must also meet environmental and safety requirements, which will limit the allowed amount of long-lived radioactivity, concentrations of short-term volatile radioactive species, and decay heat in used components. We currently lack the materials performance experimental data required to support sufficient model development and inform design criteria. While data from existing neutron irradiation sources are helpful for predicting materials performance at lower neutron energy fluences and temperatures, which may be sufficient for informed materials down selection, the community must still develop advanced materials for service in commercial FPPs. The community therefore requires experimental data at significantly higher temperature and higher neutron energy fluences in order to accurately predict performance in structural and plasma-facing materials. In this context, the US fusion program has identified the building of a Fusion Prototypic Neutron Source (FPNS) as having the highest priority to advance the construction of a FPP within the Bold Decadal Vision articulated by the US-DOE. The desired operational parameters for FPNS have been discussed by members of the the US fusion materials program in a series of workshops that reflect the best practices in fusion materials testing, modeling, and incorporate upgraded design criteria based on updated ITER design limits as well as private company compact reactor concepts. Selection of a suitable FPNS concept requires a multifactorial approach where advanced modeling, state-of-the-art fusion technology, and quality of experimental nuclear databases, as well as cost and logistical considerations, all intersect. In this talk we will discuss the latest design considerations towards a FPNS that fits within the above constraints.

Keywords

Fusion neutron spectrum, materials testing, irradiation damage, high temperature irradiation, high dose irradiation.

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PL 14 ABSTRACT-9 a9b

L. Fission-Fusion Synergy and Cross Cutting Technologies

Private Public Partnership for Early Commercialization and Innovative Fusion Nuclear Technology

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A number of fusion challenges in the recent years are lead by private company or private-public partnerships. They are funded by various investments encompassing low carbon energy technology, and are not subject to the budget constraints of public scientific programs. There are diversity of concepts not limited to conventional magnetic configuration such as tokamak or stellerator, they cover mirror, pinch, FRC, spheromac, magnetized target, and inertial confinements. Most of all are unique and innovative, and apply latest technology to make fusion smaller, earlier, and more attractive. When compared with the conventional approach by governments, such as ITER and following DEMO based on the conservative and less technical risk approach, these technical challenges intend to choose more advanced technology, such as higher magnetic fields supplied by High Temperature Superconducting magnets. In the blanket and nuclear technology design, almost all concepts choose liquid breeder and in some cases wet wall concepts. They will usually require higher fusion energy flux, but smaller tritium inventory. Although concrete designs are not available yet in most of the projects, small scale DT operations are planned in the next several years with small amount of tritium processing to be involved, leading to the demonstration of energy conversion for power generation. Safety design and nuclear regulations will require significantly different approach from the public DEMO programs. These challenges also have a distinctive feature of project management in terms of funding, personnel and supply chain featured by private startup companies' management. Private sector exchanges technology through business channel, that takes advantage of world wide selection of cutting edge technical products, while export control and supply chain are issues. This talk reviews the recent development of earlier fusion industrialization and provides consideration from various aspects; its nuclear technology issues and their background from commercial and business management.

Keywords

Private, commercialization, industry,tritium, energy conversion.

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PARALLEL ORAL SESSIONS

P1A1 ABSTRACT-d4c7

B. Blanket Technology

Overview of the design activities of the EU DEMO Helium Cooled Pebble Bed breeding blanket

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Within the EUROfusion framework, the Pre-Concept Design (PCD) Phase of the European DEMO program ran from 2014 to 2020 aiming to establish a Pre-Concept Design of the DEMO plant. In 2020, the European DEMO program went through the first formal DEMO-wide Gate Review 1 (G1) by independent panels. Two driver-blanket candidates for the European DEMO, the Helium Cooled Pebble Bed (HCPB) and Water Cooled Lithium Lead (WCLL) breeding blankets, passed the G1. Although substantial achievements have been accomplished, many challenges and risks are still outstanding.

Starting in 2021, the European DEMO program enters the Concept Design (CD) Phase. In the CD phase (2021-2027), key challenges and risks will be tackled through design and research activities to reach the milestone of Blanket Selection Gate Review, in which the driver-blanket will be selected between the HCPB and WCLL based on figure of merit. One of the key challenges facing all blanket concepts is the low reliability of blanket systems under DEMO conditions, which drives the availability of the DEMO plant. To resolve this low reliability challenge, a fault-tolerant concept of HCPB blanket is proposed. The newly-proposed design also tackles the low shielding performance by designing an efficient shield. The new flow scheme will solve the long-standing issue of having large thermal stress at the blanket backplate. The implications of the newly-proposed design on the Safety and Tritium Extraction and Removal (TER) system are investigated. Various nuclear, thermal hydraulics, structural, tritium transport, safety and electromagnetic analyses are conducted to justify the newly-proposed design. This paper gives an overview on the design status of the HCPB breeding blanket and its associated TER system for the European DEMO. It concludes with the outlook of future design and research activities in the CD phase to achieve a safe, high tritium breeding ratio, fault-tolerant HCPB breeding blanket.

Keywords

EU DEMO, Helium Cooled Pebble Bed, Breeding Blanket, Design.

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P1A2 ABSTRACT-681c
B. Blanket Technology

Accelerating stellarator reactor engineering: design and integration of the Dual Coolant Lithium Lead Breeding Blanket for HELIAS

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To bring the stellarator concept to technological maturity as an alternative to tokamaks under EUROfusion Mission 8, engineering activities for a Helical-Axis Advanced Stellarator (HELIAS) power plant have started covering the design and integration of breeding blankets (BB). The Dual Coolant Lithium-Lead (DCLL) BB has high potential to answer the challenges posed by complex stellarator configurations, with a liquid breeder and decoupled cooling circuits for the first wall (FW) and BB. Here, both characteristics are exploited to simplify BB maintenance and integration by: i) adopting a fully detached FW and ii) adapting the PbLi path and BB shape to the complex HELIAS morphology and non-uniform magnetic field. A quasi-toroidal BB segmentation has been proposed since this minimizes the pressure drop due to coupling of the magnetic field with PbLi. This drastically simplifies the BB design as the need for electrical insulation systems (FCI, coatings or ceramic walls) is suppressed. Such a segmentation would however complicate BB maintenance through a conventional port scheme as taken from tokamaks; however, freedom is provided by the absence of a settled existing stellarator maintenance scheme, and this motivates novel strategies and integration solutions such as the use of a detached FW to transfer the maintenance problem mainly to small FW panels manageable through ports. On this basis, a FW based on Capillary Porous System (CPS) has been defined and assessed by thermal-hydraulic and neutronic analyses. To accelerate the DCLL HELIAS design and overcome bottlenecks identified during FP8 regarding the lack of tools to represent 3D complex geometries, *ad-hoc* tools have been developed for the parametrization of models and convergence toward a viable design, speeding-up coupling of the CAD modelling with neutronic, thermal-hydraulic and multi-scale thermomechanical analyses. The tools have been validated through benchmarks to confirm their suitability to be used in the stellarator framework.

Keywords

HELIAS , DCLL BB , neutronics , thermomechanics , MHD , Multiphysics, Capillary Porous System.

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P1A3 ABSTRACT-dc0e
B. Blanket Technology

Preparing for the First Integrated Test of a Fusion Breeding Blanket Prototype in the CHIMERA Facility

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The development of technologies for a fusion power reactor to the point of readiness for series production will require progressively more comprehensive integrated testing for design and qualification. Perhaps the most complex and unproven such system is the breeding blanket, which must meet highly challenging requirements and undergoes complex simultaneous loading conditions. A primary blanket concept in Europe, for both the EU DEMOnstration reactor and for the ITER Test Blanket programme, is the water-cooled lithium lead (WCLL), which foresees liquid lithium-lead eutectic as tritium breeder and high temperature water at 325°C, 15.5 MPa as primary coolant. The construction of the CHIMERA facility offers, for the first time, a unique opportunity to test this blanket technology under integrated conditions of relevant magnetic field, in-vacuum heat flux, and heat exchange to water at the correct conditions. This blanket test programme is a considerable undertaking and is being realized jointly by UKAEA and EUROfusion. This paper recaps the CHIMERA core capabilities and reports the status of preparations for the WCLL blanket test.

In parallel with CHIMERA core facility construction, a project is underway with EUROfusion for the design and installation of a Pb-Li circulation loop, for which the engineering design is progressing through maturation. A key part of this project is the design of the WCLL prototype to be tested including progressive manufacturing viability and specification of on-board instrumentation.

Also underway is the development of a highly accurate engineering simulation *digital replica* of the blanket prototype under test. Among the many motivations for this digital replica are 1) validation of magnetohydrodynamics simulations against CHIMERA test data, and 2) the ability to introduce additional models for effects such as internal volumetric (nuclear) heating and radiation damage, which combined with the CHIMERA test data enable fully integral *virtual qualification* of the component.

Keywords

CHIMERA, blanket test facility, WCLL, magnetic field, heat flux, MHD.

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P1A4 ABSTRACT-34a4
B. Blanket Technology

Structural Integrity Assessment of the Central Outboard Segment of the EU DEMO HCPB Breeding Blanket

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The Helium Cooled Pebble Bed (HCPB) breeding blanket (BB) is one of the two "driver" blanket concepts being considered for the European demonstration fusion power plant (EU DEMO). The current HCPB blanket design is based on the Single Module Segment (SMS) concept, in which a vertically continuous U-shaped first wall structure, two caps, and a back-plate structure form the outer shell. Within these shell, the fuel-breeder pins are arranged in a hexagonal arrangement along with additional supporting plates. The entire blanket system is divided into 16 blanket sectors, which are located within the vacuum vessel. In a blanket sector, there are 3 outboard blanket segments (left outboard segment, central outboard segment, and right outboard segment) and 2 inboard blanket segments (left inboard segment and right inboard segment). Due to its position in the tokamak and its wide range of functions, the blanket segment is exposed to gravity, pressure, thermal, electromagnetic (EM), and potentially seismic loads during normal and off-normal operations. Therefore, ensuring the structural integrity of this system is a paramount and non-trivial endeavor during its design.

This paper presents the status of the structural assessment studies performed on the conceptual design phase of the HCPB BB. It includes the development of an appropriate simplified finite element representation of the HCPB central outboard segment (global-model), its verification studies and its thermo-structural analysis incorporating the thermal simulation results from a detailed section of HCPB BB (sub-model). Major observations from the assessment are summarized and the future design work to address these issues are discussed.

Keywords

DEMO, breeding blanket, HCPB, single module segment, structural assessment.

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P1A5 ABSTRACT-6cb6
B. Blanket Technology

Development of the high temperature PbLi experimental loop for CFETR

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Chinese Fusion Engineering and Test Reactor (CFETR) aims to demonstrate fusion energy production up to 200 MW, and finally reach commercial electricity power level 1GW. Moreover, it will rely on the blanket for tritium production. The blanket is in charge of tritium breeding, neutron shielding and energy conversion. It can be classified into the solid and liquid blanket in view of the breeder materials state. Among them, the liquid blanket has the advantages of simpler structure, tritium extraction and fuel replenishment online, as well as high thermoelectric conversion efficiency. Moreover, the supercritical carbon dioxide (sCO_2) cOoled Lithium-Lead (COOL) blanket has been proposed. However, the key issue is the Magnetohydrodynamics (MHD) that needs to perform experiments. Therefore, the high temperature PbLi experimental loop is being development, which has finished the system design, key component selection and three-dimensional layout. For the system design, it includes the three loops of PbLi, oil and water. Among them, the oil loop is between the PbLi and water to avoid chemical reaction when the LOCA accident occurs. The main components includes the mechanical pump, main heater, cold trap as well as the superconductor magnetic, et al. For the experimental section, the mass flow rate, temperature and magnetic intensity will be up to 32 kg/s, 700℃ and 3 T, respectively. The ultrasonic Doppler and potential probe instruments will be installed on the test section to obtain the flow field. The main experiments will include the turbulent heat transfer, MHD as well as the mixed convection, and the main goal is to establish flow heat transfer models, which are the critical input for the liquid blanket design and optimization.

Keywords

Fusion blanket, COOL blanket, PbLi experimental loop.

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P1B1 ABSTRACT-e3f5

A. Plasma-Facing High Heat Flux Components

Overview of the divertor development programme in EUROfusionDesign

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In view of the severe operating conditions of a divertor in fusion devices, the development of optimized divertor structures and components is of very high priority. The components not only have to withstand high steady state power loads (10 - 20 MW/m^2), but also a high number of thermal cycles and shocks. Moreover, in the case of a future reactor, damage and transmutation through fusion neutrons has to be considered when designing components and selecting the adequate armour and structural materials. The technologies foreseen for the plasma facing components (PFCs) are moderately extrapolated from the one developed and tested for ITER, in order to provide a reliable heat transfer for steady state plasma operation. The reliable joining of armour and heat sink in PFCs is a complex process and their integration into the complete divertor components ultimately needs to be optimized as a whole.

There are several devices planned, under construction or intending to switch to all metal PFCs where actively cooled (divertor) PFCs are needed. The European activities for R&D on PFCs for JT-60SA, W7-X, DTT and DEMO are performed under the umbrella of the EUROfusion work package 'Divertor' (WPDIV). For JT-60SA and W7-X, a flat-tile design is presently favoured for the divertor PFCs and WPDIV solely concentrates on the development of the PFCs. In the case of DTT and

DEMO W mono-blocks similar to those of ITER are under consideration. For these two devices not only the PFCs but also their integration into the complete divertor components are subject of the research within WPDIV. Whereas in the case of DEMO the effects of neutron irradiation are a strong design driver for the divertor engineering, flexibility in view of advanced divertor configurations is central aspect in the research related to DTT.

Keywords

Divertor, Heat exhaust, Plasma Wall Interaction, Plasma Facing Components.

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P1B2 ABSTRACT-02bd
A. Plasma-Facing High Heat Flux Components

Analysis of full WEST divertor tiles after C4 campaign by TOF ERDA

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In the present work, retention of plasma fuel and composition of co-deposited layers on W-coated divertor plasma-facing units (PFUs) was determined after the WEST C4 campaign, where a dedicated Helium (He) phase was run at the end of the campaign [1], following Deuterium (D) operation. Here, special focus was put on determining the poloidal distribution of He on the PFUs to complement other surface analysis performed [2]. Measurements were performed using Time-of-flight Elastic Recoil Detection analysis (TOF ERDA). In addition to He and D, measurements of the boron and carbon content are reported. In order to fully map the complex elemental composition of the tiles after exposure to plasma, TOF ERDA was performed along the poloidal direction, from high-field side (HFS) to low-field side (LFS).

As expected, helium was very well localized around the inner strike point (ISP) and outer strike point (OSP) areas. Boron was mainly found on the HFS of the inner tile which is in agreement with the results obtained on the ITER-like PFUs with other techniques such as Nuclear Reaction Analysis (NRA) and Secondary Ion Mass Spectrometry (SIMS). However, two regions with increased boron concentration (around 40 at.%) were also detected in an area corresponding to the high-field side far scrape-off layer (SOL) while the second peak is on the private flux region. On the outer tile, the boron peak is on the low-field side SOL. Concentration of carbon was at the level of 10-15 at.% at both, inner and outer tile, except at the high-field side SOL from ISP where the thick deposit layer is placed, where high carbon concentration (about 65 at. %) was measured. Near surface profile of tungsten was also determined.

[1] E. Tsitrone et al 2022 Nucl. Fusion 62 076028

[2] M Balden et al 2021 Phys. Scr. 96 124020

Keywords

WEST C4 campaign, TOF ERDA, He, light elements.

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P1B3 ABSTRACT-9b16

A. Plasma-Facing High Heat Flux Components

A Versatile Divertor for the Italian Divertor Tokamak Test Facility

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[8]DTT S.C.a.r.l.
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The Italian Divertor Tokamak Test facility (DTT) is a high field (B_t = 6 T) high current (I_{max} = 5.5 MA) medium size (R/a = 2.19/0.70 m) long pulse (t_{pulse} ≈ 100 s) superconducting device presently under construction in Frascati, Italy. Its main purpose is to study power exhaust solutions in regimes as close as possible to those foreseen in a fusion reactor in terms of power exhaust. The European consortium EUROfusion supports the design and construction of DTT which is seen as a device to test advanced divertor geometries as a risk mitigation strategy for the conventional ITER-like divertor. Since at the moment there is no advanced configuration clearly favoured, a very flexible divertor layout has been adopted and is currently in the final engineering design phase. The chosen divertor shape is the result of numerous plasma simulation considering plasma equilibria with a standard single null and an X-point divertor with a large variation of the divertor leg lengths. Additionally, the compatibility with single null negative triangularity plasmas was explored. As a result of the modelling, the engineering design led to a divertor consisting of rather conventional inner and outer vertical targets combined with a flat dome and a large outer baffle. All these areas will be equipped with actively cooled W monoblocks allowing everywhere power loads higher than 10 MW/m^2. Further design criteria were a sufficiently high pumping capability, which is achieved by 9 -10 cryopumps in the vertical ports below the divertor, as well as the remote handling capability which is achieved through the extraction of divertor segments through the lower horizontal ports. The contribution will present the latest status of the engineering design of DTT divertor also highlighting the role of the foreseen in-vessel coils as well as the results of first PFC mock-up tests.

Keywords

DTT, Divertor, Heat exhaust, Plasma Wall Interaction.

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P1B4 ABSTRACT-7f9c

A. Plasma-Facing High Heat Flux Components

Design and Analysis of Actively-cooled, Edge-transport Diagnostic for Long-pulsed Operation in WEST

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Next step fusion devices that will operate in steady-state will require complex plasma-facing components (PFCs) that can survive the harsh environment over long timescales not common in current devices. This will require robust plasma facing surfaces that are integrated with active cooling systems. In a collaboration between CEA and ORNL, a plasma-interacting diagnostic is being designed for the W Environment in Steady-state Tokamak (WEST) in Cadarache, France which requires plasma-facing protection like those needed for steady-state PFCs. This integrated diagnostic studies edge transport and impurity migration within WEST, and includes imbedded temperature Langmuir probe sensors, as well as removeable sample slots for ex-situ surface analysis of plasma-material interactions. The entire assembly is expected to move into the plasma edge for periods up to 5+ mins having an energy removeable capability of ~10 kW and seeing a peak heat flux approaching ~9.5 MW/m^2. The assembly compliments WEST high fluence campaigns that plan for multiple pulses as long as 1000 seconds each. Within these specifications, the assembly will require a refractory metal plasma facing surface and integral cooling in order to function within the limited space allotted for such diagnostics. Because of the limited space and linear actuator needs, additive manufacturing of the high heat flux working end of the assembly is being considered which could allow for precision cooling-channels and lighter weight designs. Conceptual design along with simulation and analysis results will be presented for this complex diagnostic with novel PFCs.

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Keywords

Plasma facing components, high heat flux, edge transport.

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P1B5 ABSTRACT-7d2a
A. Plasma-Facing High Heat Flux Components

Study on the reliability and fatigue life prediction of hypervapotron under electromagnetic-thermal coupling condition

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The divertor is one of the components closest to the plasma, and the enormous energy contained in the plasma acts on the divertor mainly in the form of electromagnetic and thermal loads. In this paper, the mechanism of Halo current and eddy current generation is discussed, and the electromagnetic analysis under Halo current and eddy current is carried out for the hypervapotron target of the divertor. The magnitude of electromagnetic forces was obtained through simulation and theoretical calculations, and the reliability of the structure was initially evaluated based on electromagnetic mechanical analysis. Furthermore, the real stress state under the impact of electromagnetic load is simulated, the transient dynamics analysis of the target structure is carried out, and the safety performance evaluation of the structure is carried out according to the criteria of the internal components of the fusion device, which shows that the structure has high reliability. Then, combined with previous work on fatigue life of hypervapotron target under steady-state thermal load, a fatigue life prediction model under electromagnetic and thermal coupling condition was established by using multiple linear regression method. The relationship between key parameters, such as heat flow density, cooling water flow rate, Halo current and fatigue life of the target is explored, and the fatigue life of the divertor can be effectively predicted according to different combinations of design parameters. The reliability level of the divertor system under the electromagnetic and thermal coupling of the future fusion reactor can be more accurately assessed. The established fatigue life model has important reference significance for the safe and stable operation of the divertor.

Keywords

Divertor, Hypervapotron, Electromagnetic-thermal coupling, Multiple linear regression, Fatigue life.

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P1C1 ABSTRACT-2515
J. Burning Plasma Control and Operation

Development of the DEMO plasma diagnostic and control system

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The DEMO diagnostic and control (D&C) system will have to cope with demanding requirements on accuracy and reliability of operation, while its components mounted inside the tokamak will be exposed to strong loads (neutron and gamma irradiation, heat and particle fluxes, stresses). Facing these boundary conditions, and while the details of the DEMO plasma scenario and machine design are still evolving, the development of the basis for the DEMO D&C concept has progressed in the recent years up to a point where details of the methods, components and number of channels for the D&C subsystems as well as their expected performance are becoming known and being further elaborated. Specifically, a list of the main DEMO plasma control issues and control requirements has been compiled, together with an allocation of the diagnostic systems and actuators needed for plasma control. Here, the diagnostic methods have been selected for their robustness under DEMO conditions and for the required coverage of control issues. Initial diagnostic integration studies into the tokamak have been started to investigate the space requirements and identify possible issues such as interface challenges. For some of these diagnostic methods the main technologies and components as well as the number of required channels can already be described and the anticipated performance can be roughly predicted, or interpretative and predictive modelling tools are under development. Concerning the control strategies, numerical tools for kinetic, equilibrium and MHD mode control are employed and being further developed to conduct simulations for optimisation and validation of the concept. However, for some of the diagnostic and control methods, important design choices are still open and dedicated research and development (R&D) is needed for clarification aiming to demonstrate a sufficient maturity of the D&C system as part of overall DEMO concept by 2027.

Keywords

DEMO plasma diagnostics and control.

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P1C2 ABSTRACT-5d29
J. Burning Plasma Control and Operation

Fusion Neutron Diagnostics with CVD Diamond Detectors

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The measurement of DD and DT fusion neutrons is crucial for plasma diagnostics at nuclear fusion facilities, such as ITER. Elevated temperatures and high radiation levels challenge radiation detectors. Chemical Vapour Deposition (CVD) diamond withstands harsh environments and is a distinguished candidate for reliable plasma instrumentation. Neutron induced nuclear reactions on Carbon nuclei of diamond substrates are the basis for fusion neutron detection with CVD diamond detectors. DD fusion neutrons have an energy of 2.45 MeV. The corresponding response function of diamond sensors is purely elastic scattering and has a characteristic shape. DT fusion neutrons have 14.1 MeV kinetic energy, and their response function reveals different nuclear reactions. The 12C(n,a)9Be peak is the most prominent structure in this response function.

The principles of neutron diagnostics with CVD diamond detectors are discussed in this paper and the characteristic response functions of diamond sensors for 2.45 MeV and 14 MeV neutrons are presented. Experimental results are compared to Geant4 simulations and an estimation of the neutron detection efficiency is given for DD and DT fusion neutrons.

Keywords

Neutron detection, CVD diamond detectors.

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P1C3 ABSTRACT-fdda
J. Burning Plasma Control and Operation

Development of Hybrid Operational Scenario on CFETR and EAST Tokamak

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CFETR is the next step machine in China's roadmap for magnetic confinement fusion development. It aims to bridge the gap between the ITER fusion experimental reactor and the demonstration fusion power reactor (DEMO). Hybrid operational scenario has been considered as one of advanced scenarios for CFETR. Experiments of hybrid operational scenario have been carried out on EAST tokamak with ITER-like tungsten (W) divertors recently. In the hybrid H-mode plasma, the internal transport barrier (ITB) has been obtained with central flat q prfile and it is found that the fishbone mode (m/n=1/1) can be beneficial to sustain the central flat q(0）=1 profile with a stable ITB. The behavior of W in the core of hybrid plasma scenario on EAST with ITER-like divertor is studied. W accumulation is often observed and seriously degrades the plasma performance. It is found that the toroidal rotation and density peaking of the bulk plasma are usually large in the central region, which is particularly prone to the W accumulation. The simulation reproduces the experimental observations of W accumulation and identifies the neoclassical inward convection/pinch velocity of W due to the large density peaking of the bulk plasma and toroidal rotation in the central region as one of the main reasons for the W accumulation.

Keywords

Hybrid Scenario, ITB plasma, Tungsten transport, CFETR, EAST tokamak.

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P1C4 ABSTRACT-38cd
J. Burning Plasma Control and Operation

Feasibility of a Collective Thomson Scattering diagnostic for burning plasma control on DEMO

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Diagnostic systems are essential for burning plasma control and operation of DEMO but at the same time challenging in design and integration. The harsh environment around the DEMO plasma, and the space restrictions and need to maximize the first-wall area used for tritium breeding, set limitations on the number and type of diagnostics to be installed. This will focus the efforts on diagnostics needed for control of the DEMO plasma. The robustness and versatility [1] of a microwave-based Collective Thomson Scattering (CTS) diagnostic make it worthwhile to investigate the potential of a DEMO CTS diagnostic. The Technical University of Denmark has undertaken this exploratory effort under a contract with EUROfusion.

The feasibility study for a CTS diagnostic for DEMO builds on the experience of past experiments from both TEXTOR and ASDEX Upgrade, and on the recent development of the ITER CTS system that recently passed the Final Design Review [2]. The ITER CTS diagnostic will focus on measurements of fast ion dynamics in the burning ITER plasma [3]. The original target of the DEMO CTS diagnostic was to use an ECRH gyrotron beam as the probing source beam, with the receiving quasi-optical system being a dedicated CTS setup. Based on raytracing calculations including signal-to-noise estimates, it was found that such a setup is not viable. Here we present studies of alternative solutions, including assessments of which DEMO plasma parameters that the CTS diagnostic may contribute to determine. We will discuss the first steps towards integration of the diagnostic in DEMO.

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Keywords

Diagnostic, DEMO, Collective Thomson Scattering.

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P1C5 ABSTRACT-e1e1
J. Burning Plasma Control and Operation

An overview of the evolution of the modelling of reflectometry diagnostics in fusion plasmas using finite-difference time-domain codes

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Microwave reflectometry, having its origins in ionosphere probing techniques to evaluate electronic density, has become one of the most important diagnostics in fusion plasmas and will play a major role in next-generation machines, in particular in DEMO, where it is expected to provide plasma positioning, shaping and tracking. The ability to have an ever-increasing comprehensive description of reflectometry is particularly important since it allows to assess the measuring capabilities of existing experimental systems and to predict the performance of future ones. Nevertheless, propagation in a thermonuclear plasma is far from trivial and the need for a numerical full-wave treatment of the problem is fundamental. Several Computational ElectroMagnetic (CEM) techniques have been used to tackle this problem, being finite-difference time-domain (FDTD) using Yee's scheme one of the most important. We will expose the audience to the fundamentals of this technique and introduce them to the use and evolution of FDTD in reflectometry, using as an example the synthetic diagnostics setup with the family of REFMUL* codes and employed in the assessment of the performance of several reflectometry systems in different fusion machines.

Keywords

Reflectometry, finite-difference time-domain, synthetic diagnostics.

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P1D1 ABSTRACT-220b
K. Inertial Confinement Fusion Studies and Technologies

Hydrodynamics experiments on the shock ignition approach to direct-drive inertial fusion

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The recent NIF breakthrough showed the credibility of the inertial fusion approach. However, it does not seem possible to extrapolate the indirect drive scheme followed at NIF to future energy production due to complicated targets, intrinsic low efficiency, and finally the implication for defense programs.

Alternatively, direct laser drive offers higher efficiency and simpler scheme, but it is more prone to laser non-uniformities and the impact of Rayleigh-Taylor instability. Decoupling compression and ignition, the basis of the shock ignition (SI) approach, could mitigate such instability. Here compression is followed by high intensity irradiation (10^{16} W/cm^2) creating a strong shock converging at the center of the compressed target and increasing the temperature thus triggering nuclear reactions.

Presently there are many unknowns in SI, in particular on Hot electrons (HE) from laser–plasma interaction. Here, the results of two experiments conducted at the Omega facility will be presented.

The first one, in planar geometry, aimed at characterizing strong shock propagation and HE generation. Time-resolved radiographies were performed to study the hydrodynamic evolution. The HE source was characterized using x-rays spectrometers. Obtained values were used in simulations to reproduce radiography results, further constraining HE parameters. We found ~10% energy conversion into HE with T~27 keV. HE produced an increase of the pressure around the shock front. The low temperature found in this experiment could be advantageous for SI.

The second one, in spherical geometry in the so-called “40+20 configuration” used a laser temporal shaping typical of SI including a final spike. Neutron yield and areal density were measured for implosion performed changing the launching time of the final spike. The detrimental effect of HE on areal density and neutron yield was shown for an early spike launch. For a later spike launch, this effect is minimized because of higher target compression.

Keywords

Inertial fusion, hot electron, shock ignition, shocks, implosion.

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P1D2 ABSTRACT-2247
K. Inertial Confinement Fusion Studies and Technologies

Integrated simulations of ion fast ignition of inertial fusion targets

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Ion fast ignition is an alternative scheme to ignite thermonuclear fuels with lower energy and symmetry drive requirements [1,2]. Many ion beam requirements estimated so far for fast ignition are based on strong assumptions about beam focusing and ideal beam-plasma interaction given only by the standard stopping power models. However, some new effects have been reported, such as: i) divergence of laser-driven protons generated in hollow cones [3] and its consequences on ion energy deposition [4]; ii) anomalous energy deposition of intense ion beams in resistive plasmas [5]; and iii) improved modelling of stopping power. Dedicated experiments [6] have shown that the BPS stopping model [7] fits the measurements of ion stopping, while the standard stopping theories used so far show substantial differences.

We aim to evaluate the laser energy requirements of the IFI scheme for realistic fuel configurations. We have conducted integrated simulations of IFI, from ion generation to fuel ignition, combining PIC, hybrid, and radiation-hydrodynamic simulations. We will analyze the relevance of the effects mentioned above and will assess ion fast ignition as an alternative scheme for inertial fusion targets.

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Keywords

Inertial fusion targets, fast ignition, ultra-intense lasers.

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P1D3 ABSTRACT-68de
K. Inertial Confinement Fusion Studies and Technologies

Determining the spatial structure of inertial fusion energy plasmas

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Efficient burning of nuclear fuel in the hot plasmas of inertial fusion energy (IFE) schemes depends on establishing specific spatial distributions of plasma temperature and density conditions. Assessing the degree to which these conditions have been achieved in the IFE experiment presents a significant challenge to both diagnostics and measurement techniques. We discuss an approach to meet this challenge that relies on multi-objective data analysis and the fundamental physics properties of the x-ray emission from suitable tracers added to the fuel. The idea of multi-objective data analysis is that there is unique information that can be extracted from the simultaneous and self-consistent analysis of multiple measurements that is not possible to obtain from the separate and independent analysis of individual pieces of data. Furthermore, we have found that smart searches in multi-dimensional parameter space driven by Pareto genetic algorithms, which combine the idea of Pareto domination with the mechanics of natural selection, represent an efficient and practical implementation of multi-objective data analysis. Results are illustrated with the extraction of the three-dimensional temperature and density spatial distributions in inertial confinement fusion (ICF) implosion cores driven by the OMEGA 60-beam high-power laser, and the two-dimensional temperature spatial profiles of the laser-heating phase in magnetized-liner-inertial-fusion-energy (MagLIF) experiments performed at the Z pulsed-power facility. The ideas are general and can be applied to other IFE schemes as well.

Keywords

Spatial structure of inertial fusion energy plasmas, multi-objective data analysis, ICF, MagLIF.

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P1D4 ABSTRACT-8f47
K. Inertial Confinement Fusion Studies and Technologies

The HiPER+ Initiative for Inertial Fusion Energy studies

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The recent achievements at the National Ignition Facility (NIF) in the USA consisting in reaching ignition of fusion reactions with laser-driven technologies [1] set a historic milestone in fusion energy research by demonstrating inertial fusion using lasers as a viable approach for future energy production. Europe has a unique opportunity to empower research in this field at the international level, and the scientific community is eager to engage in this journey. We propose establishing a roadmap to Inertial Fusion Energy (IFE) in Europe, with the missions to demonstrate ignition of fusion reactions with a laser and to develop pathway technologies to the commercial fusion reactor. This roadmap takes origin in the HiPER project [2,3] and is comprised of five complementary axes: i) the physics of laser plasma interaction and burning plasmas, ii) the high energy high repetition rate laser technology, iii) the fusion reactor technology and materials and iv) the reinforcement of the laser-fusion community by international education, training programs, and collaboration with research centers, industry and private companies, v) the establishment of joined activities with the private sector involved in laser- fusion. Along with high level of societal importance this project aims to stimulate a broad range of high profile industrial developments in laser and plasma technologies [4,5].

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Keywords

HIPER+ Initiative.

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P1D5 ABSTRACT.-5fc1
K. Inertial Confinement Fusion Studies and Technologies

Time-resolved spectroscopy of proton-heated targets relevant to proton fast ignition

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The recent demonstration of ignition at the National Ignition Facility has generated significant interest worldwide in the use of Inertial Confinement Fusion (ICF) for future fusion power plants. However, to make thermonuclear fusion a reality for energy applications, one needs to significantly increase the gain of the target. Leveraging the intense energy deposition of fast particles (electrons, ions) into the compressed fuel has been proposed to increase target gains, a scheme referred to as Fast Ignition (FI). FI is also attractive due to less stringent symmetry requirements because of separate compression and heating phases. Heating by protons is interesting due to their heavier mass, which provides a more localized energy deposition and makes them less prone to instabilities during transport to the compressed fuel. We have carried out a series of experiments at the OMEGA-EP facility to understand the focusing, transport, and energy deposition of laser-generated proton beams. We will present our previous findings and a recent experiment where we resolved for the first time the x-ray fluorescence emission of a solid sample promptly heated by a laser-generated proton beam. In this experiment, the EP laser (450-900 J, 5-10 ps) was focused onto a cone-enclosed partial hemisphere to generate and focus an intense proton beam into 10 and 25 µm-thick solid copper samples. The measured line emission shifts reveal heating of the sample to temperatures up to ~50 eV within ~35 ps. Our data and simulations are consistent with the energy deposition of a proton beam focused to a ~100 µm spot at ~430 µm from the source, with a measured characteristic temperature of ~4 MeV and total energy of ~15 J for > 1 MeV (conversion efficiency from laser of ~2%). We will conclude by discussing plans for integrated experiments on OMEGA to advance the study of proton-FI scheme.

Keywords

Inertial confinement fusion, fast ignition, proton heating, warm dense matter, x-ray spectroscopy.

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P2A1 ABSTRACT-059c
H. Models and Experiments for FNT

Breeding blanket challenges and needs for technology qualification: ongoing R&D efforts and open fields on relevant nuclear testing data

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Undisputedly, the Breeding Blanket is one of the most important and novel components of DEMO reactor and the key component for the realisation of fusion power plant because of its main functions: (i) tritium self-sufficiency, (ii) power extraction and (iii) neutron shielding.

DEMO or any other nuclear fusion device after ITER would need to operate from the first day with a full coverage breeding blanket able to produce and recover reliably its own fuel. Despite its criticality to the development of fusion power, no breeding blanket has ever been built or tested and some technical issues remain unsolved (i.e., heat transfer across breeder/multiplier/structural materials, tritium generation and extraction in Li-based solid/ liquid breeders, n-irradiation damage, etc.).

Because of several remaining uncertainties and feasibility concerns, a selection of the DEMO breeding blanket is now premature, and a sustained programme of technology R&D is required and is being implemented by EUROfusion.

Several valuable studies were conducted in the 1980's – 1990's to identify the main knowledge gaps and the required R&D and testing for the development of the breeding blanket (see for example the FINESSE Study). These work needs revisiting to take into account, the breeding blanket concepts being considered and that relevant facilities and or test that have been built or are being constructed (e.g., TBM Programme in ITER) together with those being planned (such as IFMIF DONES).

In light of the above, fusion nuclear technology testing issues are critically reviewed in this paper, considering the different aspect of the breeding blanket such as materials science, structural mechanics, tritium generation and extraction, MHD, thermal hydraulics, etc. The resolution of these issues requires a vigorous program of integrated experiments, modelling, and theory. The definition of the needs to bridge the remaining gaps is discussed.

Keywords

Breeding Blanket, technology qualification, testing requirements.

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P2A2 ABSTRACT-dd23
H. Models and Experiments for FNT

Needs and Options for Qualifying Fusion Nuclear Technologies

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The qualification of fusion nuclear technologies to be deployed in future fusion reactors is an unquestionable prerequisite for the development of fusion power. In general, this is also a requirement for licencing fusion systems.

The resolution of the challenges connected to the qualification of the Breeding Blanket technologies requires extensive tests both in non-fusion and fusion facilities. In the non-fusion facilities single and separate effects can be studied and the experimental results can be compared with the outcome of numerical models. However, these data are limited and, sometimes, not fully representative of the operating conditions of the breeding blanket.

Irradiation in fission reactors and IFMIF-DONES are important, but we need much more. In particular, DONES will concentrate on lifetime neutron radiation effects in a small, limited volume (and limited time). There is a need for testing relevant blanket technologies in an actual DT-plasma based device to fulfil the remaining gaps. These should include the results of the Test Blanket Module programme in ITER, which remains an important confirmation experiments albeit at very low fluence.

Since the beginning of fusion research proposals were made to conduct extensive nuclear testing of key fusion nuclear technology such as the breeding blanket, including the construction of dedicated volumetric neutron sources with beam-heated plasmas of very modest energy gain and far less stringent plasma performance than in conventional high-Q energy schemes. This facility would be extremely needed to minimise the risks associated with validating robust breeding blanket and fuel cycle that remain a big challenge to allow deployment times of fusion energy. Also, aspects connected to the tritium extraction technologies as well as tritium management and relative safety concerns should be addressed.

This work summarises the testing requirements of a fusion facility for the qualification of the BB technologies identifying the strategy for testing.

Keywords

Breeding Blanket, technology qualification, testing requirements.

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P2A3 ABSTRACT-d5b4
H. Models and Experiments for FNT

Overview of Fusion Technology Programs at Oak Ridge National Laboratory

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Motivated by the US plan to rapidly develop a fusion pilot plant, ORNL is aggressively pursuing a Fusion Nuclear Science, Technology, and Engineering program to establish the technical basis and provide cost-attractive solutions for the fusion subsystems needed to turn burning plasma into a viable commercial power plant. These are divided into four primary groups:

The Blanket and Fuel Cycle group develops solutions for closing the tritium fuel cycle and efficient power extraction in future fusion systems. In addition to the development and deployment of pellet injection systems on fusion devices worldwide, the blanket program performs integrated design simulations incorporating neutronics, fluid flow (including liquid metal MHD), structural mechanics, and tritium transport. Helium and PbLi loops provide critical experiment data on heat transfer and material compatibility to inform these designs.

The Fusion Technology group develops innovative technology approaches for heating, fueling, and controlling plasmas required for efficient operation of future fusion systems.

The Fusion Engineering group develops advanced designs and engineering solutions for both individual components and integrated systems for future fusion systems. A recent focus of this group has been the design of the Material Plasma Exposure eXperiment (MPEX)—a next-generation linear plasma device under construction at the ORNL fusion energy campus. MPEX will support the study of the way plasma will interact long term with materials within future fusion reactors.

The Remote Systems group specializes in robotics, remote handling and process equipment R&D designed for use in hazardous environments, and has expertise ranging from full-scale prototyping and demonstrations, to field deployment on projects related to nuclear fuel reprocessing, remote maintenance, and environmental remediation.

This presentation will give an overview of these activities, as well as describing plans to develop an integrated fusion technology and physics framework to facilitate the design and optimization of pilot plant concepts.

Keywords

Fusion Technology, Fusion Engineering, Blanket and Fuel Cycle, Remote Handling.

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P2A4 ABSTRACT-10 a3
H. Models and Experiments for FNT

Measurement of tritium production in the HCPB TBM mock-up at JET during DTE2

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Quite often, detectors for measuring nuclear performance and radiation quantities of relevance in fusion experiments are requested to withstand harsh working conditions due to intense neutron and gamma radiation fields. High temperature constitutes a further harsh element in some locations of the machine where it is necessary to perform some on-line measurements, as expected in the breeding blanket. This is an essential component in future fusion power plants to provide tritium self-sufficiency and its performance must be continuously monitored. Some Test Blanket Modules (TBMs) will be installed in ITER to provide the first experimental data to validate the predictions on tritium production and recovery. In the meantime, within EUROfusion program, the mock-up of the Helium Cooled Pebble Bed Test Blanket Module (HCPB TBM), previoulsy used for the TBM experiment at the Frascati Nuetron Generator (FNG), had been installed at JET to test some detectors and for benchmarking numerical codes used for breeding blanket assessment during DTE2 campaign. A diamond detector, calibrated to measure the tritium production through neutron detection inside the HCPB TBM mock-up, was tested during some plasma pulses of the DTE2 campaign at JET. The main outcome is that, as far as neutron emission rate is below 10^{15} s^{-1}, neutrons are properly detected along the plasma discharge evolution by TBM diamond, consistently with the JET neutron monitor (KN1). Moreover, the amount of tritium measured (E) is 1.40×10^{-12} tritons per source neutron and the comparison with MCNP radiation transport simulation (C) gives a ratio C/E=0.77. Such measurements, considered promising, and their comparison with calculations are discussed in the present work. Criticalities emerged are analyzed and some improvements proposed with the main purpose of speeding up signal processing to make the system capable of working at higher neutron emission rates.

Keywords

HCPB TBM mock-up, Diamond detector, Tritium production measurement, Tritium production prediction.

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P2A5 ABSTRACT-927a
H. Models and Experiments for FNT

High-fidelity tritium transport modeling of retention and permeation experiments

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Idaho National Laboratory (INL) developed the Tritium Migration Analysis Program (TMAP) code to analyze tritium retention and loss in fusion reactor structures and systems during normal operation and accident scenarios. TMAP4 (released in 1992) and TMAP7 (released in 2006) have been used for safety analyses in licensing ITER and design studies for many U.S. and international future fusion reactors. However, the previous versions of TMAP were limited to *one-dimensional* thermal- and mass-diffusive transport and trapping calculations (with a single trap for TMAP4 and up to three traps for TMAP7) through structures and zero-dimensional fluid transport between enclosures and across the interface between enclosures and structures. This limitation prohibits us from conducting high-fidelity tritium transport modeling of complex 2D and 3D geometry experiments.
The (Multiphysics Object-Oriented Simulation Environment) MOOSE-based TMAP (TMAP8, available at https://mooseframework.inl.gov/TMAP8) was developed in 2019 to utilize the multiphysics coupling and massively parallel computational capabilities of MOOSE to conduct high-fidelity tritium transport simulations in complex 2D and 3D geometry to couple with other MOOSE-based tools in mind. MOOSE is an open-source framework for developing multiphysics simulation software that allows rapid development of new simulation tools and is developed to a Nuclear Quality Assurance, Level 1 (NQA-1) standard.
This paper discusses the development and applications of TMAP8, highlights the similarities and key differences between TMAP4/TMAP7 and TMAP8, and shows examples of high-fidelity tritium transport modeling of retention and permeation experiments.

Keywords

Tritium permeation, tritium retention, modeling, TMAP.

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P2B1 ABSTRACT-fcf9
A. Plasma-Facing High Heat Flux Components

Additive manufacturing techniques for the fabrication of tungsten based plasma-facing components

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In future deuterium-tritium magnetic confinement fusion devices, plasma-facing components (PFCs) will have to sustain intense particle, heat and neutron fluxes over a considerable operational lifetime. Such harsh operating conditions ask for the development of advanced and resilient components capable of exhausting high heat fluxes reliably under the aforementioned conditions. Present-day developments regarding additive manufacturing (AM) can be considered as a technological leap that can leverage unprecedented potential regarding the design of components, including highly loaded PFCs. AM is a term that describes processes in which material is deposited layerwise enabling substantial design freedom, meaning that parts with more or less arbitrary shape can be fabricated straightforwardly. Although W is a challenging material for fabrication processes in general due to its intrinsic properties, like high melting point and high ductile-to-brittle transition temperature, the AM of W has made substantial progress in the more recent past making such a fabrication approach an increasingly viable way for the realisation of high-quality W parts. Against this backdrop, the present contribution will summarise the status quo of AM of W in the context of different AM processes that are currently being investigated by research institutions and industries. Such processes mainly utilise laser or electron beams as energy source for selective material melting and subsequent consolidation. Based on that, the contribution will highlight recent and topical investigations of the authors regarding the AM of W based PFCs, including laser based W/Cu alloy multi-material AM. Furthermore, an approach to improve the performance of highly heat loaded W and Cu based divertor PFCs by means of tailored composite structures will be presented and discussed. Such W-Cu composite structures forming the heat sink of a PFC are realised by means of liquid Cu infiltration of open porous and additively manufactured W preforms.

Keywords

tungsten, additive manufacturing, plasma-facing component, high heat flux, tungsten-copper composite, divertor.

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P2B2 ABSTRACT-ed5d
A. Plasma-Facing High Heat Flux Components

Improvements of the Tungsten Guard Limiters of 4.6 GHz Lower Hybrid Wave Antenna of the EAST and its Service Status under Operation

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It has been observed since 2018 that severe ablations happened on the tungsten guard limiter (GL) for the 4.6GHz hybrid waveguide antennas (LHWA) in the Experimental Advanced Superconducting Tokamak (EAST) after every campaigns. The ablations of the GL located ON the top and edges near the LHWA . The emergence of ablations is partly due to flaws in structure design of the GL, and partly due to poor heat-exhaust capability of the GL. Thus, a new guard limiter for the 4.6GHz LHWA must been designed to remedy flaws and enhance heat-exhaust capability. This paper will focus on the improvements of the new GL and its service status in the 2022 winter experimental of the EAST device and service. First, the reason for ablations is deduced: ablations on the top of the GL was deduced to be caused by the plasma; ablations on the edges of the GL near the LHWA was deduced to be caused by fast electrons due to LHWA heating driving. Then the design flaws are corrected in the new design of the GL: the edge of the GL near the LHWA is set behind the edge of LHWA in the radial direction to avoid being attacked by the fast electrons at a small bevel angle, and the former highest corner of the GL formed by two mono-blocks was replace by the ridgeline of one mono-block. Preliminary analysis of temperature field of the GL is conducted using roughly estimated heat flux from the plasma and fast electrons by finite element method (FEM) for estimation and decision. The new GL has gone through 2022 experimental campaigns of the EAST device, and no ablation was observed on the GL again, but hotspot was observer on the edges of the LHWA. These new phenomena may be worthy being studied and discussed.

Keywords

Experimental Advanced Superconducting Tokamak (EAST), hot spots, lower hydrid current drive (LHCD), tungsten guard limiter (GL).

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P2B3 ABSTRACT-280e
A. Plasma-Facing High Heat Flux Components

Divertor cooling technology, fabrication process, and armor materials comparison studies

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During the last decade, the design and performance of the European DEMO divertor target has been optimized for high heat-flux performance. It is based on the so-called "monoblock concept", in which tungsten monoblocks are hot radially pressed on a CuCrZr pipe with an interlayer of pure copper. The interlayer is produced by casting, which requires quality assurance of each monoblock due to its susceptibility to porosity formation.

Our contribution contains four interrelated studies:

(1) Galvanic deposition of copper interlayers on the monoblock boreholes proved a reliable and reproducible alternative to casting. The process can also be used to add activation layers to enhance the adhesive strength, which has been demonstrated by low-temperature solid-state bonding, followed by ultra-sonic and high-heat-flux tests (HHFT).

(2) This process was then applied to the monoblock concept for sintered and rolled W, K-doped W, and WC. The analysis of the mockups after HHFT in the GLADIS (Garching LArge DIvertor Sample) facility indicated an enhanced recrystallization and grain growth resistance of K-doped W. WC suffered from significant volumetric swelling, and most sintered tungsten monoblocks showed cracks, which gives a clear lower margin for the materials selection.

(3) We developed three alternative concepts (copper and CuCrZr infiltrated tungsten flat tile featuring swirl, hypervapotron, or turbulence promoters) by rapid prototyping, i.e., by additive mockup manufacturing via electron beam melting and direct metal laser sintering. The turbulence promoter geometries were derived from computational fluid dynamics simulations, the flat tile shapes were based on thermal stress analyses, and the swirl based monoblock dimensions are taken from the ITER/DEMO concept.

(4) The HHFT of these new concepts in the GLADIS facility confirmed the expected cooling performance and armor stability. The turbulence promoter solution provides even more relaxed operation conditions with higher tolerances.

Keywords

Plasma facing unit, water cooled mockup, rapid prototyping, additive manufacturing, materials technology, high heat flux test, alternative designs, CFD simulation.

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P2B4 ABSTRACT-026f

A. Plasma-Facing High Heat Flux Components

Progress of JA-DEMO Divertor Conceptual Design: Coolant Distribution and Thermal Stress Analysis

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A conceptual design of JA-DEMO divertor has been developed for handling large thermal power (250-300 MW) and neutron nuclear heat (NH: ~120 MW) in the ITER-like divertor geometry, where radiative cooling is enhanced in the divertor leg (1.6 m) longer than ITER. W-monoblock (MB) target units with CuCrZr pipe are used only near the strike-points (high plasma heat load area), and high temperature coolant (~200˚C, 5MPa) is used to minimize DEMO-specific risks such as radiation embrittlement of Cu inter-layer and Cu-alloy pipe. W-MB target units with F82H steel pipe are used with the PWR coolant (~290°C, 15MPa) for the high neutron load areas such as baffles, reflectors and dome. Conceptual design of the W-MB units, cassette body (CB) and coolant pipe routing has been recently renewed, based on thermal stress analyses (Ansys) on the W-MB units and CB, and computational fluid dynamics (CFD) analysis on coolant distribution from the main pipes to the W-MB units.

Analyses of heat transport and elasto-plastic stress were performed for the fish-scale shaping W-MBs with high plasma heat load and NH. Elastic plus plastic strain was increased in the CuCrZr pipe under repeating heat load (15MW/m^2) higher than that during normal steady-state operation (<10MW/m^2). Stress-strain trace at the maximum-stress location showed similar trajectory. Change in the strain was ~0.25%, which may not be critical for the long-term.

New water-cooling concept for NH on CB was proposed with making numbers of toroidal coolant passages, and the PWR coolant was provided from a main side route of CB to another side route. Coolant velocity of 1.5 m/s was enough to remove NH of 0.7 MW/CB. The coolant flow distribution and analyses of heat transport and thermal stress were performed over the whole CB. The dome/reflector design and coolant distribution were also developed.

Keywords

JA-DEMO, Divertor, Stress analysis, Coolant distribution, Plasma Facing Units, Cassette body.

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P2B5 ABSTRACT-5933
A. Plasma-Facing High Heat Flux Components

Plasma Facing Component technologies and test facilities developments in India

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Institute for Plasma Research

Development of technologies and establishment of test facilities related to Plasma Facing Components (PFC) for tokamak applications are being carried out at the Institute for Plasma Research (IPR) in India. Recent activities include: (a) Thermal cyclic testing of water cooled PFCs: Brazed Tungsten Mono-Block test mock-up with smooth coolant tube is tested for 1200 Thermal Cycles at 9.5MW/m^2 heat flux; (b) Critical Heat Flux (CHF) of water cooled PFCs: New CHF Correlation is developed for one-sided heating condition of PFC, Experimental indications obtained on marginal reduction in the CHF value due to rapid increase of incident heat flux on PFC; (c) Non-Destructive testing of PFC: Ultrasonic testing is performed for flaw detection of PFC, Ultrasonic wave propagation studies through tungsten materials is performed to estimate bulk elastic properties, Digital Image Correlation is performed to study stress distribution on surface of materials up to elevated temperatures; (d) High Heat Flux Test Facility (HHFTF): New Target Handling System is designed and fabricated for HHFTF to test Helium Gas Cooled PFCs; (e) X-ray imaging: X-ray imaging is being developed to detect position of high energy electron beam incident on surface of tungsten based PFC during high heat flux testing in HHFTF; (f) Tungsten composites & tungsten coatings for plasma facing material applications: Specimens of Tungsten Composite materials using tungsten fibers/foils are produced and their thermo-physical properties are studied; (g) Small specimen testing: Shear Punch testing on small tungsten specimen are performed for estimation of its mechanical strength at elevated temperatures; (h) Helium Gas Cooled PFCs: Engineering design and fabrication of helium gas cooled PFC test mock-up is performed.

Present paper gives current status of the progress made in the above mentioned areas related to technologies and test facilities for PFCs.

Keywords

Tokamak, Plasma Facing Components, Divertor, Firstwall, High Heat Flux Testing, Critical Heat Flux, Small Specimen Testing.

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P2C1 ABSTRACT-ad92
G. Safety Issues and Waste Management

Studies on Fusion Nuclear Technology, Materials and Safety at FDS

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The fusion studies at FDS are focused on fusion concept design, neutronics and nuclear technology, materials and blanket technology, safety, environment & socio-economics, etc., with support from various national and international programs.

A series of fusion reactor and blanket concepts had been designed, including the Multi-Functional eXperimental Reactor FDS-MFX, Liquid Lead Lithium Blanket for CFETR, Fusion-fission Hybrid Reactor FDS-I/SFB, Fusion Power Reactor FDS-II, High Temperature Fusion Reactor FDS-III, etc. Based on these designs, shared common nuclear technologies are under developing.

Series High Intensity Steady Neutron Generators (HINEG) are being developed for the missions including neutronics design validation, materials & components irradiation test, nuclear waste burning and nuclear technology applications, etc. The D-T fusion neutron source HINEG-Ia had been built and operated with yield of 6.4×10^{12} n/s. New sources including HINEG-Ib, HINEG-IIa, HINEG-IIb, etc. are under design and construction.

A new version of the neutronics analysis software
existing Monte-Carlo code SuperMC, has been developed with new functions including the coupling transport calculation of neutron, photon and electron, the improved D1S for shutdown dose rate calculation, the physical parameter visualization, etc. TopMC has been verified and validated by more than 2000 benchmark models and experiments, including ICSBEP, SINBAD, IRPhEP, etc.

For material technology, the CLAM steel, candidate structural material for Chinese ITER-TBM and DEMO, has been developed. The oxide dispersion strengthened ODS-CLAM for better irradiation resistance and higher application temperature is under developing. Besides, a series of liquid metal loops have also been built for experimental studies of material corrosion and heat exchange of blankets and advanced nuclear systems.

R&D in fusion safety, environment & socio-economics has also been performed. RiskA, a software for reliability and safety assessment, has been developed. RAMI and reliability analyses of fusion reactors have been performed.

Keywords

Fusion reactor, Nuclear technology, Fusion Materials, Fusion safety.

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P2C2 ABSTRACT-971e
G. Safety Issues and Waste Management

Waste management strategy for EU DEMO: status, challenges and perspectives

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One of the top-level safety objectives for EU DEMO design and operation is to protect workers, the public and the environment from harm and thus to minimize radioactive waste hazards and volumes and ensure that the legacy to the future generation is limited.

The objectives of the waste management strategy studies are to identify any showstoppers arising from the waste management that could impact the design and operation of the facility and identify the R&D needed to ensure a safe management of these materials and waste.

The following aspects are considered at this stage to anticipate and facilitate the waste management in particular with regards to waste that would require deep final disposal: limitation at source by selection of materials and impurities limiting the amount and toxicity of the waste, characterization of the fluxes, definition of management routes and interim storage of waste.

Studies are currently performed to ensure that the design choices will not lead to the generation of waste with decay heat preventing their management as Intermediate Level Waste (in particular tritium breeder materials and neutron multipliers such as KALOS, Be12Ti and LiPb are scrutinized), otherwise a catalogue of prohibited materials will have to be established. In parallel, the various expected streams of waste that will be produced by the operation and decommissioning and the associated management routes including processing are being analysed. Unlike waste management strategies of current fusion facilities, for DEMO and future fusion power plants, it will become essential to develop realistic and economic strategies for the reuse and recycling of rare and expensive materials such as W, Be or LiPb. On the basis of these various challenges, an experimental R&D programme focussed on the management of waste without existing management routes (such as mercury waste) and waste management optimization needs to be developed.

Keywords

Waste, EU DEMO, R&D.

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P2C3 ABSTRACT-2b6f
G. Safety Issues and Waste Management

Lessons Learned from the Public Acceptance of Fission Reactor in China for Fusion Energy Development

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Fusion energy has the potential to be a safe, environmentally friendly and sustainable power source, with many substantial advantages in terms of sufficient fuel supply, little long-lived radioactive waste, no criticality accident, and no proliferation risk, and it seems to be more easily accepted by the general public than the fission energy. However, fusion is not born to be safe and the public acceptance of fusion energy may also be a big concern. For example, the potential hazards from numerous radioactive materials may also bring huge challenges. Moreover, bad impressions of fission reactor may have a negative influence, even a stereotyped image, in shaping the public perception of fusion.

In this contribution, the preliminary studies on the public acceptance of fission reactor in China will be introduced as well as some findings including: 1) the public sensitivity on the potential danger of radiation is much higher than other environmental pollution; 2) the public knowledge level about nuclear energy is generally low and many people are easy to be incited by some negative comments; 3) the nuclear power plant (NPP) operating company is widely conceived as an untrustworthy information source; 4) the ratio of inland NPPs opponents reaches a quite high level and the syndrome of "not-in-my-back-yard" (NIMBY) is very obvious. Suggestions for public acceptance of fusion energy will be also presented in this work.

Keywords

Public Acceptance, Nuclear Power Plants, Fusion Energy, Fission Energy, NIMBY.

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P2C4 ABSTRACT-b96e
G. Safety Issues and Waste Management

Nuclear Integrated Engineering approach to enhance nuclear integration following ALARA principle in ITER project

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The main ITER goal is to produce 500 MW of DT fusion power, targeting the production of 3×10^{27} neutrons in the current research plan. The massive production of neutrons leads to nuclear integration challenges shared with any nuclear fusion machine to be connected to the electricity grid.

Here we focus on radiation protection of workers, with the Occupational Radiological Exposure (ORE) as a challenge. During the outages, maintenance and assembly, workers will intervene in environments activated by the neutrons. The ITER Preliminary Safety Report emphasised the importance of considering ALARA principles in the design to demonstrate a collective dose of 500 man.mSv/year on average. As the design maturity level permitted detailed analysis, the need for further study and optimization following the application of ALARA approach across all project design activities became evident. Given the ITER particularities, the need to implement an integrated approach by an entity playing the role of the future nuclear operator was identified as well.

The Nuclear Integrated Engineering (NIE) was set, a project wide transverse study with support from industry, to enhance nuclear integration optimization of the plant design, and application of industrial best-practices in the management of complex maintenance in a nuclear environment. The aim was to assess working scenarios via a holistic, multi-disciplinary approach, considering external radiation exposure, contamination hazards, influence of human and organizational factors, maintainability, tools & equipment, personal protection, and occupational health & safety risks. Iterations were undertaken for different scenarios to optimize the plant configuration and working scenario. Amongst the outputs of this process were quantifiable ORE assessment based on integrated maintenance scenario, Health & Safety, Human and Organizational Factors recommendations and Dose Reduction Measures leading to plant optimization.

We give an overview of the implemented integrated approach and the outputs; lessons learned can be useful for other fusion reactors.

Keywords

ALARA, radiation, ITER, Occupational Radiation Exposure.

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P2C5 ABSTRACT-4541
G. Safety Issues and Waste Management

Experimental Investigation on Tungsten Dust Explosion Behavior for Fusion Reactor

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In a Tokamak vacuum vessel, the micro-dust is produced by the interaction between energetic plasma with first wall. The dust inventory in the existing tokamak devices is quite small, in grams (e.g. JET, JT-60) or kilograms (e.g. TFTR). However, in future fusion reactors, more dust will be produced because of the plasma is more energetic and the material erodes faster. In the case of loss of vacuum accident (LOVA) or loss of coolant accident (LOCA), The dust will be re-suspended, dispersed, and even triggered to explode by hydrogen, then poses a great threat to the safety and operational performance of fusion reactor. Therefore, understanding the dust explosion behavior is very important. In this work, a series of experiments were carried out to study the explosion characteristics and mechanism of the pure tungsten dust and hydrogen/dust mixtures with a 20 L spherical vessel and a 5 L closed combustion tube. The explosion intensity parameters, flame propagation speed and flame structure were obtained. The effects of the initial conditions including temperature, vacuum degree, ignition energy, ignition delay time, dust particle size were analyzed. By means of scanning electron microscopy (SEM) and X-ray diffraction (XRD), the structure and composition of explosive residues were identified. Based on the experimental data, the physical picture of the reaction mechanism of pure tungsten and hydrogen/dust mixtures was futrther established. The study can provide a reference for the accident prevention of the future fusion reactor.

Keywords

Dust explosion, Explosion characteristics, Reaction mechanism, flame structure.

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P2D1 ABSTRACT-9638
C. Fuel Cycle and Tritium Processing

Tritium related challenges to be overcome in order to deliver fusion power plants.

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The fusion landscape has changed immensely over the last few years, notably following the arrival of many privately funded fusion companies. In the main, these companies tend to focus on the core fusion problem of reaching ignition and scale up of their core technologies towards establishing fusion power plants. Most of the commercial initiatives, plus the non-commercial activities such as EU DEMO and UK STEP, presume the use of the deuterium-tritium fusion reaction. It is therefore an appropriate opportunity to summarise the challenges and solutions encountered in fusion power plant design relating to the use of tritium, both in terms of tritium availability, the fusion fuel cycle, and the migration and recovery of tritium throughout power plants.

Tritium topics to be considered include:

- Scarcity
- Safety
- Inventory minimisation
- Permeation
- Breeding and tritium extraction
- Recovery and processing
- Tritium management and control
- Waste management
- Decommissioning
- Regulation and licensing

Without attention being paid to the above issues, such that solutions are found in a timely manner, deployment of fusion power plants will be delayed. This presentation is intended to seed interest in developing solutions for tritium challenges to facilitate power plant roll-out.

Keywords

Overview, tritium, fuel cycle, fusion power plant, challenges.

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P2D2 ABSTRACT-fa5a
C. Fuel Cycle and Tritium Processing

Construction and Technological Tests of Storage and Delivery Demo-system (SDS) for Deuterium-tritium Internal Fuel Circulation in Tritium Plant

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Deuterium-Tritium (D-T) storage and delivery system (SDS) is a crucial part in the tritium plant of nuclear fusion, and its foremost function is to rapidly provide D-T fuel for fusion reactor and to store D-T fuel from other sub-systems. However, the integrated SDS matching the actual engineering scale has not been reported yet, and the technological parameters have no way to acquire and verify. Herein, we built a full-scale SDS demo-system to predictably meet the requirements of future nuclear fusion reactor (200-1500 MW). In the SDS, self-developed double-thin wall storage bed with a secondary containment structure, rapid heat/mass transfer, and in-situ precise metering technology were integrated. Highly sophisticated distributed control system (DCS) and safety interlocking system (SIS) are introduced to monitor, control and interlock protection of the SDS demo-system. Basically, leak rate < 10^{-9} pa.m^3/s, pressure resistance > 0.6 Mpa, vacuum degree < 5 Pa, operated temperature > 500 ℃, and automated operation have been achieved. Large scale storage and delivery experiments show that the rates can reach to 5.09 m^3/h (141.4 Pa.m^3/s, 39.5 min) for storage and 7.24 m^3/h (201.1 Pa.m^3/s, 41.5 min) for delivery respectively. In addition, the SDS demo-system has also been integrated with TEP, ISS, and ANS to establish a steady-state operation (2.4 m^3/h, ~500 MW) and further evaluate the engineering feasibility of the whole D-T internal fuel circulation.

Keywords

Internal fuel, Storage and delivery system, Deuterium-tritium, Full scale.

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P2D3 ABSTRACT-ef32
C. Fuel Cycle and Tritium Processing

Tritium infrastructure to support an Accelerator-based DT Neutron Generator

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SHINE Technologies has commissioned a 1012 n/cm2/s neutron source with a DT fusion spectrum. The facility delivers a 50-mA deuteron beam accelerated to 300 kV into a low-pressure tritium/deuterium gas target. Continuous outputs on the order of 4.6*1013 neutrons/s for 15 minutes by delivering deuterons into a pure tritium gas targets have been achieved. Steady-state operations to increase fluences have been demonstrated by delivering the 50-mA beam into a gas target containing a 40/60 tritium/deuterium mixture for 8-hours. A 1.7-gram tritium facility supports the accelerator operations and is licensed by the state of Wisconsin. Tritium is delivered to the gas target via a mass flow meter and extracted from the chamber with a cryo-sorption pump to maintain the desired gas pressure within the target chamber. The effluent is passed over a permeator to remove non-hydrogenic species from the stream before the gas is sent to an isotope separator. Isotope separation is based on the Thermal Cycling Absorption Process (TCAP). Pure tritium leaving the isotope separator is returned to the target chamber to maintain the required tritium-to-deuterium ratio. Preparations are underway to operate the facility continuously over multi-day periods. This presentation will describe the systems used to deliver and process the tritium, discuss planned upgrades to increase the fluence and present performance data for the 8-hour run.

Keywords

Tritium, fuel cycle, tritium management.

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P2D4 ABSTRACT-fd65
C. Fuel Cycle and Tritium Processing

Regression models for generating thermodynamic and transport properties of all phases of hydrogen isotopologues

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Extensive research has been conducted on the fuel cycle of deuterium and tritium in fusion reactors to ensure successful operation and efficient use of tritium. The properties of hydrogen isotopologues (H2, D2, T2, HD, DT, and HT) play an important role in designing and optimizing fuel cycles in tritium-related facilities. These isotopologues exist in gas, solid, and liquid phases during the fuel cycle, and there is still a lack of experimental data, particularly for tritium-containing species and for solid and liquid phases, limiting the accuracy of the parameters required for the optimization process. To address this issue, we have developed and utilized theoretical and computational approaches, such as vibration analysis and path-integral molecular dynamics, and collected thermodynamic and transport properties, such as enthalpy, heat capacity, density, and viscosity, of hydrogen isotopologues. The calculation results showed a good agreement with available experimental data.

In the present study, based on the data obtained from the theoretical and computational methods and available experimental data, we developed a code to generate thermodynamic and transport properties of hydrogen isotopologues for all phases using regression models. The code provides properties data at a given temperature, pressure, and isotopic composition, along with predicted errors, in a data format that can be easily used in design codes such as ANSYS, and thus is expected to help improve the reliability of optimization process of the tritium fuel cycle. In the presentation, we will introduce the functionality and performance of this code as well as the theoretical models and regression methods implemented.

Keywords

Thermodynamic properties, transport properties, hydrogen isotopologues, regression model.

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P2D5 ABSTRACT-e9fb
C. Fuel Cycle and Tritium Processing

Graphene-based electrochemical system for radioactive hydrogen isotope enrichment

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Isotopic effects in the electrochemical processes of water had been reported broadly: starting from the early era of nuclear technologies up today. Effect is based on the differences in the kinetics of the hydrogen evolution reaction of different isotopes and can be used for heavy hydrogen isotope separation. Electrolysis had been the first widely used method for deuterium enrichment and currently is used in combination with catalytic isotope. However, this technique involves large energy consumption. Use of the solid polymer electrolyte membrane PEM instead of electrolytic solutions has number of advantages: lower specific energy consumption, minimal amount of electrolyte in the device, absence of any impurities and no risk of explosive gas mixture formation since hydrogen and oxygen can be emitted by separate outputs. In order to enhance separation in electrolysis process, the use of graphene coated polymer membrane electrolyte has been proposed by Hidalgo et al, where the role of graphene is based on quantum effects. However, hydrogen gas produced during the electrolysis still contains a considerable fraction of the heavier isotopes, therefore introduction of the fuel cell into the systems allows to continuously reprocess this gas into the water and enriching it with deuterium or tritium. The present study reports of the development of a set-up for separation of tritium in a water phase based on the combination of concepts described above. Additionally, special attention is dedicated to evaluation of the radiation stability of the developed system, thus choice of radiation stabile materials and test under potential exposure conditions is also discussed in this study

Keywords

Tritium, graphene, polymer electrolyte membrane.

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P3A1 ABSTRACT-216d
B. Blanket Technology

Overview of Progress on Water Cooled Ceramic Breeder Blanket in Japan

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National institutes for Quantum Science and Technology (QST) are the implementing body of ITER project and takes a central role in fusion DEMO reactor development in Japan. The primary concept of the breeding blanket for JA DEMO is Water Cooled Ceramic Breeder (WCCB) . The other concept such as Helium cooled liquid breeder is going to test at the later stage of JA DEMO. ITER Test Blanket Module (TBM) program is an opportunity for the demonstration of DEMO breeding blanket concept under the real fusion environment. QST is leading WCCB concept in the program since TBM arrangement signature in 2014. After the conceptual design approval in 2016, QST changed TBM shape from box to cylindrical to enhance tritium breeding capability with maintaining pressure resistance. The preliminary design of test blanket system (TBS) is on-going. TBS consists with TBM-set and its ancillary systems. WCCB TBM is made of Reduced Activation Ferritic/Martensitic steel, F82H. Since that is ferromagnetic, electromagnetic analysis is important to show structural integrity of TBM. TBM is cooled by pressurized water at around 573 K and 15.5 MPa and provided by water cooling system. Bred tritium in the TBM is going to purge out by Helium gas flow and separated from it at tritium extraction system. To get tritium production rate, neutron environment in TBM should be known. Neutron activation system is a pneumatic system to send a capsule including activation foils to the TBM. These ancillary systems will be operated according to ITER plasma operation. The instruments and controlling system design are progressed. Recently, the status of the preliminary design of TBS was assessed between QST and the ITER Organization. So, the latest design status of TBS and its DEMO relevancy will be introduced.

Keywords

ITER, DEMO, Test Blanket Module, Water Cooled Ceramic Breeder, F82H.

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P3A2 ABSTRACT-879b
B. Blanket Technology

Alternative water-cooled breeding blanket concepts for the EU DEMO: Overview on studies and perspectives

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During 2014 to 2020, the pre-Conceptual Design Phase (PCDP) of the EU DEMO took place. Two breeding blanket (BB) concepts, namely the Helium Cooled Pebble Bed (HCPB) and the Water Cooled Lithium Lead (WCLL) BB concepts have been developed and proposed as reference BB concept candidates for the EU DEMO. During the final design review of the PDCP, risks have been identified for both BB concepts, some of them deemed as critical, and a new system design and R&D program have been defined in the Work Package Breeding Blanket (WPBB) within the EUROfusion Consortium for the Conceptual Design Phase (CDP) to address those.

In parallel to these design and R&D activities in the WPBB, an action was started in the DEMO Central Team (DCT) in EUROfusion to find alternative architectures for a water-cooled BB concept that may mitigate the present risks by proposing innovative designs. The studies conducted have led to two possible solutions, the so-called WCLL "double bundle" and the Water-cooled Lead and Ceramic Breeder (WLCB). Although the former is a derivation of the reference WCLL, analyses have shown a significant mitigation of the aforementioned risks while having a limited impact on the current R&D program. In contrast, the WLCB is more ambitious and tries to bring the best technological features of both HCPB and WCLL concepts, offering a good trade-off. The outcome of the different design studies and the perspectives are reported in this paper.

Keywords

HCPB, WCLL-DWT, WCLL-DB, WLCB, Breeding Blanket.

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P3A3 ABSTRACT-10f6
B. Blanket Technology

Deuterium retention properties of beryllium intermetallic compounds

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Beryllium intermetallic compounds, known as beryllides, such as $Be_{12}Ti$ and $Be_{12}V$, are highly promising advanced neutron multipliers for use in demonstration (DEMO) power reactors due to their low swelling and high stability at high temperatures. Advanced neutron multipliers are being developed by Japan and the EU as part of their Broader Approach (BA) activities within the International Fusion Energy Research Center (IFERC) project.

The release and retention of hydrogen in beryllium and beryllides are extremely important properties since tritium can be generated in beryllium/beryllides by reactions between beryllium and neutrons. However, few data on tritium release and retention in beryllides have been reported. Therefore, the authors' group has conducted research and development on deuterium release and retention in beryllium and beryllides using deuterium ion implantation at room temperature. It was clear that the desorption properties and retention of hydrogen isotopes (deuterium) using the newly developed beryllide are better than those of beryllium. This is evidenced by lower starting-up temperature, total retention, and activation energy for deuterium desorption. Moreover, through comparison with transmission electron microscopy, it has been clarified that no bubbles exist at high temperatures due to the complex structure of beryllides.

This study provides an overview of research and development on deuterium desorption and retention properties of beryllides, focusing on temperature dependence, and compares them with tritium desorption and retention properties of beryllium and beryllides conducted in the past.

Keywords

Beryllides, deuterium, implantation, retention, multipliers.

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P3A4 ABSTRACT-f75b
B. Blanket Technology

Challenges of the High Heat Flux loaded Helium Cooled First Wall, Contributions of Numerical Flow Simulations

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Several blanket concepts for DEMO fusion reactors employ high-pressure (8 MPa) helium gas as coolant for the plasma facing first wall (FW). For current EU-DEMO concepts, stationary heat loads from the plasma to the first wall are expected to typically be in the range of 0.3 MW/m2, where the additional loads by thermal charged particles estimated up to 0.45 MW/m2 approx. 0.65 MW/m2 can occur. The definition of the peak values is ongoing. Higher short-term transient loads are expected.

Efficient cooling (acceptable pressure loss, aimed temperature range for structural material kept) has already been demonstrated for heat flux densities of 1 MW/m2 with helium channels equipped with transverse rectangular ribs on one side. 60°-V-shaped ribs show even 35% higher heat transfer and are thus the subject of the present study.

Numerical simulations of ribbed FW cooling channels/ FW Blanket components aim to provide: temperature fields as basis for thermomechanical analyses, technical correlations that can be used in system codes to balance the pressure drop in the parallel channels, fluid outlet temperatures and the overall pressure drops i.e. as input for the Balance of plant (BOP).

Although Scale-Resolving Simulation (SRS) techniques such as LES are able to calculate heat transfer and pressure drop precisely in a time-resolved manner, their application is limited to individual ribs or a few mm channel segment due to the required high mesh resolution and the associated high demand for hardware resources and engineering time. Nevertheless, SRS techniques can be used to compare different ribs and to evaluate the performance of RANS and Reynolds Stress Models (RSM).

RSM requires higher computational effort compared to two-equation turbulence models, but offer higher accuracy predicting flow separation and reattachment and small flow details. In comparison to LES, tested RS models underestimated temperature fields and heat transfer compared to LES simulations.

Keywords

EU-DEMO, Breeding Blanket, CFD, Heat Transfer.

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P3A5 ABSTRACT-d21b
B. Blanket Technology

Multiphysics tritium transport modelling in WCLL breeding blankets: Influence of MHD effects and neutron damage

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The replenishment of tritium fuel in a breeding blanket is fundamental for the successful operation of commercial DT fusion reactors. Accurate modelling of hydrogen transport and inventories within the breeding blanket will be essential for safety issues and economic sustainability. One of the proposed breeding blanket concepts for DEMO, the Water Cooled Lithium-Lead (WCLL) concept is modelled using the open-source hydrogen transport code FESTIM [1]. 3D Multi-material and multi-physics simulations of the WCLL design have been conducted to investigate the significance of tritium inventories and permeation into cooling channels. Trapping effects are considered in the solid domains, in addition to how trapping properties alter as a results of neutron damage over time, subsequently affecting tritium inventories. A fluid dynamics model is implemented to simulate the flow of the liquid metal LiPb in the blanket, accounting for MHD effects. The resulting velocity field was coupled with FESTIM to accurately simulate hydrogen transport in both the liquid and structural domains of the model.

Simulations have been conducted assuming DEMO operates at steady-state, for a full power year. The presence of a magnetic field is shown to severely disrupts the flow regime of the liquid metal breeder. The inclusion of trapping mechanisms has been shown to increase tritium inventories by 15% [2]. However, when considering the neutron damage effects, inventories can increase up to 4 orders of magnitude. Additionally, the difference in trap creation rate vs trapping rate for tritium results in localised build ups closest to the sources of tritium.

[1] - R.Delaporte-Mathurin et al., NME, 21 (2019)

[2] - J.Dark et al., Nuc Fusion, 61, 11, (2021)

Keywords

DEMO, WCLL, modeling, tritium transport, MHD, neutron damage.

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P3B1 ABSTRACT-1276
F. Nuclear System Design

A feasibility analysis of D-based fusion reactions based on classical thermodynamics

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In classical thermodynamics, the spontaneity of a process is established through the assessment of the change in Gibbs free energy (ΔG). So far, the feasibility of nuclear reactions has been characterized in terms of cross section and Q-value while the entropic term (T ΔS) has been neglected. Such an assumption is always justified for fission reactions where the term ΔS is positive. In the case of fusion reactions that operate at very high temperatures and where ΔS is negative, the change in Gibbs free energy may result positive making the reaction non spontaneous.

This work describes a classical thermodynamic analysis of D-based reactions of interest for the magnetic-confinement fusion applications. The entropy contribution has been evaluated via the Sackur-Tetrode equation while the change in enthalpy has been considered constant and corresponding to the Q-value of the fusion reaction. The results of the thermodynamic analysis are compared with the nuclear reactions feasibility criteria relied on the reaction reactivity.

The reactions DT and D^3He show a high degree of spontaneity although the second one presents a lower reactivity. The increase of the temperature could enhance the reactivity of the reaction D^3He at the cost of decreasing its thermodynamic spontaneity. Both branches of the DD reaction are characterized by a much lower thermodynamic spontaneity than that of the DT and D^3He reactions. Furthermore, at the temperature of their maximum cross section the DD reactions exhibit a largely positive change in Gibbs free energy and, therefore, are not spontaneous.

At the temperature of magnetic-confinement fusion machines (1.5×10^8 K), among the D-based reactions studied the DT one exhibits the highest degrees of spontaneity and reactivity.

Keywords

Deuterium-based fusion reactions, magnetic confinement fusion, statistical thermodynamics.

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P3B2 ABSTRACT-9 a57

F. Nuclear System Design

Physics and Engineering Considerations for Compact Fusion Pilot Plants

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The purpose of a fusion Pilot Plant (PP) is to bridge the gap between ITER and a *net*-electricity producing demonstration power plant (DEMO). While ITER is primarily a science and technology experiment, DEMO will need to have advanced systems that are capable of delivering electricity to the grid in an un-interrupted way. Thus, a PP will have to demonstrate the electricity production with Q_E < 1 in a cost-effective way with moderate tritium consumption (as TBR <1). This calls for compact machine configurations, addressing key issues such as several hours-long-pulsed operations, tritium production and heat extraction from the blankets, heat removal from the α-heated plasmas and power conversion with ease of maintenance and a scalability option for DEMO. Taking this into account, we have explored the parameter-space for PP using a multi-physics code SARAS [1] and identified three configurations with moderate β, f_{BS} < 0.65 for 1 FPY operation-time: (1) An ST-based configuration R350 with R=3.5m, A=1.9, P_f =300 MW, Q = 5; (2) R300 – Heat Extraction Test Reactors (HxTR) with R=3m, A = 2.7, P_f = 200 MW, Q = 4 [2] and (3) R440 with R=4.4m, A=3; P_f =300 MW, Q = 7. We have considered HTS magnets with joints for the R350, joint-free HTS/LTS magnets for HxTR with port-maintenance and LTS magnets for R440. These configurations consider the potential scalability to DEMO, either ST or conventional route [3]. In-board shielding thickness and magnet current density (J_{wp}) were important for the machine size. The details of the analysis along with the options for current-drive, radiation, blanket and maintenance for these configurations will be discussed.

References:

1. P. Deshpande and P.N. Maya IPR/RR- RR-1408
2. Prajapati et al., in this conference
3. P.N. Maya and S.P. Deshpande IAEA DEMO 8th IAEA DEMO Workshop, Vienna, 30 Aug.- 02 Sept. 2022

Keywords

Fusion Pilot Plants, Nuclear Design, Shielding, Magnet, Current Drive.

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P3B3 ABSTRACT-2622
F. Nuclear System Design

Radiological protection design considerations for DEMO

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Lessons learnt from fusion technology R&D as well as design and operation of DT fusion devices are essential input for nuclear design integration efforts by improving nuclear safety culture and accounting for radiological protection objectives. Experiences at ITER are highlighting the challenges in considering safety, design and regulatory objectives for implementing the radiological protection requirements, and in particular with regard to the design of shielding provisions when the global design of the whole facility is still evolving. These elements are demonstrating the issues in implementing measures to limit exposure to ionizing radiation, i.e. radiological protection, in a nuclear fusion plant layout. Whereas a respective comprehensive Radiological Protection Programme would need to encompass the plant's full life-cycle, current efforts for the EU DEMO are focusing on conceptual design considerations.

The paper summarizes current understanding and achievements in this area within the design efforts for EU DEMO. The perception of plant maintenance strategy, nuclear integration and safety engineering as transversal functions in a regulatory environment enables a coherent nuclear design process with ALARA integration compatible with DEMO's plant availability requirement. Main elements of a radiological protection design approach will be presented with shielding requirements and studies, identification and control of radiation source terms, minimization of activation, defininition of the main objectives of nuclear maintenance policy and preliminary ORE assessments. A preliminary radiological protection design guideline taking into account shielding design, safety and regulatory constraints and challenges at the DEMO tokamak and within the nuclear building is presented.

Keywords

Nuclear design, radiological protection, neutronics, ORE, maintenance, DEMO.

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P3B4 ABSTRACT-8e4d
F. Nuclear System Design

European Efforts and Advances in Stellarator Power Plant Studies

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Stellarator research in Europe concentrates on the exploitation and extension of the advanced stellarator Wendelstein 7-X (W7-X) located in Greifswald, Germany. W7-X provides unprecedented insights into complex 3D plasma physics. To name only few highlights, the optimization for neoclassical transport could be proven, record stellarator triple product was achieved, and recently steady-state operation with an actively cooled divertor and 1 GJ energy turnover was carried out.

With the success of W7-X, there is an increasing interest in Stellarators as promising candidates for fusion power plants. However, this field of research received very little attention over the last decades. In 2021, within the new EUROfusion framework program, a small Task was established reinvigorating the field of Stellarator Power Plant Studies (SPPS) within the Work Package Prospective Research & Development.

The goal of this Task is to bring the stellarator concept to maturity in terms of its reactor capability: i.e. catching up with tokamak developments, demonstrating the viability of the stellarator concept as a reactor, and delivering attractive options for such a device.

Stellarators have inherently a complex 3D geometry, providing a unique stellarator-specific engineering challenge towards achieving these goals. To overcome this difficulties, a strategy is employed that focuses strongly on the development of flexible modelling tools, such as systems codes and parametric CAD/CAE models, etc. Ultimately, this allows fast design iterations and optimization within minimal time and resources. This is further enabled by establishing a linked collaboration across nine different institutes with experts from relevant fields, leveraging existing expertise where possible, but also aiming to develop more stellarator competences in the EU.

I will discuss this strategy, give an overview over ongoing activities, and present initial highlights, as well as challenges encountered on the way to developing a stellarator power plant concept.

Keywords

Stellarator, modelling, system design, strategy.

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P3B5 ABSTRACT-5c72
F. Nuclear System Design

Overview of recent advancements in IFMIF-DONES neutronics activities

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IFMIF-DONES (International Fusion Materials Irradiation Facility – Demo Oriented NEutron Source) is a neutron irradiation facility currently under development within the Early Neutron Source (ENS) project of EUROfusion. It aims to provide crucial data for the construction and safe operation of a DEMO fusion power plant, and data for material modelling by providing high-intensity neutrons equivalent to fluxes at the first wall. The facility utilizes a high-current deuteron accelerator operating at 40 MeV and 125 mA to strike a high-speed (15 m/s) flowing Li curtain. This and produces high-intensity neutrons up to 55 MeV energy and 10^{15} n/cm^2/s fluxes through stripping reactions. The neutrons will be utilized for irradiating material samples housed in the test module behind the target, performing complementary nuclear experiments. Shielding will be provided by steel and concrete structures.

In recent years, within the EUROfusion Framework Programme FP9, significant progress has been made in IFMIF-DONES neutronics activities on nuclear analyses, tool and data development, and benchmark experiments. The nuclear analyses have been thoroughly conducted on beam-on and beam-off radiation doses, nuclear responses and activation inventories taking into account existing radiation source terms, including neutrons and photons produced from deuteron beam losses, beam depositions in the beam scraper, collimator, beam dump and Li target, as well as activation produced by deuteron and neutron impinging on the structures, lithium, cooling water, corrosion products, air and atmospheric gases. These nuclear analyses provide crucial data for the design optimizations, licensing, construction, commissioning and operation of the Accelerator System, Test System, Li System, Building and Plant System, as well as necessary inputs for the Remote Handling, Logistics and Maintenance, Safety and Licensing, and beyond. In addition, this paper will summarize the recent developments in the tools, nuclear data and benchmark experiments, which play crucial roles in supporting, accelerating and validating nuclear analyses.

Keywords

IFMIF-DONES, Neutronics, doses, shielding.

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P3C1 ABSTRACT-d4c0

D. Material Engineering for FNT

RAFM materials database, model data inputs and future developments toward DEMO

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Reduced activation ferritic/martensitic (RAFM) steel, e.g., Japanese F82H, is the leading structural material of the DEMO blanket and many efforts have been devoted to data-base toward DEMO design. For instance, to apply F82H to the structure of ITER-TBM, the reference procurement specification and the reference standard strength have been defined, following the example of RCC-MRx in France [1]. To establish the neutron irradiation database toward the DEMO design, tensile and toughness data up to 80 dpa were updated, and more recently, the evaluation of helium effects using Ni additives is progressing. In this paper, the status of the RAFM high-dose irradiation database will be reviewed first.

In parallel to the continued acquisition of the reactor irradiation data, it is important to identify the critical condition to explicitly show the 14 MeV fusion neutron irradiation effects by modeling simulations and experiments under the fusion-relevant environment using accelerators. In this approach, multi-scale structural analysis analytically and experimentally is important to understand the irradiation behavior at the atomic level and to reflect it in the macroscopic behavior at the bulk structural level. Recent R&D more emphasized micro- and mesoscale microstructure to bridge the atomic scale to bulk. This paper will present recent R&D progress about the microstructure vs. strength correlation as model data inputs. For instance, block structure and PAG boundaries were evaluated by the micro-scale tensile technique.

The last part of this paper is in-service properties, e.g., 1) high-temperature and high-pressure water corrosion behavior with consideration of the impacts of the water radiolysis and magnetic field, etc., and 2) mechanical properties under a high magnetic field. They are the essential information to demonstrate the macroscopic structural integrity under the DEMO-relevant environment. The progress of ongoing experiments implemented in QST will be introduced.

[1] T. Nozawa et al., Nuclear Fusion 61 (2021) 116054.

Keywords

Reduced activation ferritic/martensitic (RAFM) steel, materials database, neutron irradiation, multi-scale structural analysis, corrosion, magnetic field.

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P3C2 ABSTRACT-c073
D. Material Engineering for FNT

Croatian contribution to the development of fusion materials technology

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Croatian contribution to the development of fusion materials technology, realised thorugh EUROfusion Workpackage Early Neutron Source and Workpackage Materials, is focused on preparation of construction and operation of the IFMIF-DONES facility, as well as on operation and upgrades of the DiFU dual-beam facility for ion irradiation of fusion materials.

Design of components for the IFMIF-DONES facility involves six research institutions assembled in Croatian DONES.HR Consortium and various industrial partners. The DiFU dual-beam facility, developed jointly by EUROfusion, IAEA and Croatian Ministry of Science and Education is one of just four such facilities in Europe, with distinct design characteristics.

Keywords

DiFU dual-beam facility, ion irradiation, IFMIF-DONES.

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P3C3 ABSTRACT-0594
D. Material Engineering for FNT

Coatings: challenges of Tritium Permeation Barriers in fusion reactors context

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The development and application of robust tritium permeation barriers (TPBs) are crucial for a safe and economic fusion reactor operation because tritium permeation could cause serious safety problems, such as brittleness of the structural material, fuel loss, and radioactive contamination. Coatings effectively inhibit the permeation of tritium fuel generated inside the breeding blanket (BB) through the structural materials reducing the contamination of the ancillary systems and the exposure of workers as well as of the population. The general characteristics of TPBs must include: the ability to prevent or reduce hydrogen-isotopes adsorption; high permeation reduction factors (PRFs); and high performances about resistance to irradiation, low activation, high thermal mechanical integrity, PbLi compatibility and low electrical conductivity.

During the last years, big efforts have been made for the selection of the best TPB candidate. The last step has been the study of their compatibility in PbLi environment under reactor conditions. So far, Al2O3 has been selected as the EU candidate. However, because of the chemical reactivity of this compound with lithium, several authors report the formation of Lithium aluminate compounds meaning that Li penetrates into the coating. In this contribution, we will show recent results about the Li diffusion quantification as a function of depth in the coating and discuss the possible effects that may happen under neutron irradiation on the bases of the light species production (T and He) due to the nuclear reactions (n, T) and (n, α) with 6Li and 7Li.

Since the problem arose, a parallel selection of new candidates is started, taking into account several restrictions as the aforementioned. All the challenges that this important material has to overcome to fulfil its function will be presented, together with a general overview of coating application and functionality in the different breeder blanket designs.

Keywords

Coatings, Tritium Permeation Barrier, Lead Lithium corrosion, Materials Characterization, Safety, Breeder Blanket.

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P3C4 ABSTRACT-7a6e
D. Material Engineering for FNT

Compatibility assessments for fusion applications: Sn, Li, Pb-Li and FLiBe

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Several liquids are being considered for blanket and plasma-facing component (PFC) applications. For PFCs, Sn showed very poor compatibility with F82H (Fe-8wt.%Cr-2W) specimens in static tests at 400°C. Promising results were obtained with pre-oxidized FeCrAl at 400°C-500°C so a thermal convection loop (TCL) was built from APMT (Fe-20Cr-5Al-3Mo) and operated with flowing Sn for 1000 h with a peak temperature of 400°C. Large mass losses for pre-oxidized FeCrAl and APMT suggest that Sn will be challenging to use.

For Li, liquid metal embrittlement (LME) of 4340 and F82H steels was evaluated using hollow tensile specimens at 200°C. Results for 4340 steel showed LME but similar testing for F82H did not show similar evidence of LME.

For Pb-Li, six TCL experiments have been conducted for 1000 h each with commercial Pb-17at.%Li and APMT tubing. The first four monometallic TCLs had increasing peak temperatures from 550° to 700°C. The fifth TCL with ODS Fe-10Cr-6Al and high-purity SiC specimens with a peak temperature of 700°C. Severe attack was observed with large dissolution of FeCrAl and the formation of a thick (>100 μm) carbide-silicide reaction product on SiC specimens. The sixth TCL lowered the peak temperature to 650°C and pack aluminized F82H showed only small mass changes.

For FLiBe molten salt, initial static experiments are underway to test the compatibility of beryllide neutron multipliers (e.g. $TiBe_{12}$) and F82H. Experiments in Isothermal 500 h Mo capsules are being conducted at 550°-750°C to measure the amount of dissolution and to determine if F82H reacts with Be if $TiBe_{12}$ also is present in the salt.

Research sponsored by the U.S. Department of Energy, Office of Fusion Energy Sciences, the FRONTIER collaboration with Japan and the Laboratory Directed Research and Development Program of Oak Ridge National Laboratory.

Keywords

Compatibility, liquid metals, molten salts, liquid metal embrittlement.

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P3C5 ABSTRACT-3723
D. Material Engineering for FNT

SMART materials for DEMO: towards industrial production

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Self-passivating Metal Alloys with Reduced Thermo-oxidation (SMART) are under development for plasma-facing components of the future fusion power plant, such as DEMO. SMART materials containing chromium, yttrium and tungsten exhibit similar sputtering resistance as that of pure tungsten during regular plasma operation. In case of an accident combining loss-of-coolant and air ingress into the vacuum vessel, SMART demonstrate the suppression of oxidation outperforming tungsten.

The viability of SMART concept has been shown at the laboratory scale. SMART materials and pure tungsten have demonstrated identical sputtering resistance during deuterium plasma exposures under conditions corresponding to 20 days of continuous plasma operation of the power plant. Under accident conditions, SMART features remarkable 10^4-fold suppression of oxidation and more than 40-fold mitigation of sublimation of tungsten oxide as compared to those of pure tungsten.

Presently, the scale-up of SMART technology is underway involving industrial partners. This activity comprises the industrially supplied feedstock materials for SMART, mechanical alloying of SMART powder at industry and field-assisted sintering of SMART materials using industrial equipment.

Industrial mechanical alloying already now can be accomplished within 20 hours of milling, instead of 60 required earlier at laboratory scale, providing at least four kilograms of fully alloyed powder per session.

At the same time, the synergy of using the industrial know-how and research experience allowed to sinter SMART rectangular samples with linear dimensions of 10 cm, thickness of 0.5 cm and a weight slightly below 1 kg. Density evolution of industrial-grade SMART materials demonstrated an outstanding progress, presently exceeding 97%.

Finally, the transition to industrial suppliers of the feedstock allowed SMART to outperform pure tungsten in powder procurement costs needed for plasma-facing components. Current status of industrial SMART technology will be presented along with outlook to future activities.

Keywords

DEMO safety, SMART materials, oxidation resistance, first wall materials, tungsten alloys.

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P3D1 ABSTRACT-8e2d
I. Repair and Maintenance

Overview of progress towards maintainable architectures for fusion devices

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The maintenance system for a fusion power plant will be significantly more complex than for any previous fusion device. A plant architecture is needed that permits efficient maintenance, but even so, it will be challenging to ensure an acceptable availability.

In this talk I will discuss how progress is being made towards maintainable architectures in the EU DEMO through:

1. Collaboration with the DEMO Central Team and with plant designers to ensure the maintainability of the plant architecture
2. Definition of candidate remote maintenance equipment
3. Identification of gaps and development of remote maintenance technologies

I will describe how we are adopting a bottom-up design approach to compliment the top-down approach, grouping our designs into a catalogue of Remote Maintenance Equipment, organised by equipment type, to ensure the same item of equipment or components can be used for as many applications as possible, rather than the geographic approach previously taken. This allows us to concentrate on the design driving, novel equipment, and we can determine its maximum capability, and this design data is vital to inform the top-down plant architecture development. The top-down and bottom-up work inform each other so that as they come together, they form a feasible system from both a plant and maintenance perspective.

And I will also briefly show the key design driving maintenance equipment and some of the main areas of technology development and the proposed new Remote Maintenance Test Facility to be built in the UK starting in 2024.

Keywords

Maintenance, Availability, Remote Handling, Robotics, Technology.

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P3D2 ABSTRACT-c378
I. Repair and Maintenance

Breeding Blanket Remote Handling System for CFETR and EU-DEMO

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The demonstrated ability to remotely maintain, both as corrective or planned actions, the tritium breeding blanket of fusion nuclear devices represents an essential requirement for the licensing as well as the proof of economical viability. This challenge is commonly shared by CEFTR and EU-DEMO. Breeding blankets of both devices are made of banana-shaped vertical segments toroidally organized in groups of 5 per vacuum vessel sector. A collaborative effort involving Chinese and EU institutions is currently underway aiming at conducting a joint development of a viable technical solution for the remote maintenance of these breeding blanket segments. Significant progress is jointly being made in completing the design of a blanket segment remote handling system compatible with the requirements of both CFETR and EU-DEMO including, in particular, the characteristics of full automation. A maintenance scheme through vertical ports has been adopted and the design work covers all the necessary tools and steps from the machine all the way to the hot cell. These include, in particular, pipe handling tools (gripping, cutting, welding), functional casks, movers and transporters, automatic connectors and contamination protection, etc. Moreover, a detailed definition of test facilities to be used to verify and validate the proposed design as well as maintenance scheme and to be implemented within the Comprehensive Research fAcility for Fusion Technology (CRAFT) at ASIPP in Hefei, China has been completed. As the procurement of the first set of small facilities to test the feasibility of critical design components currently progresses, the definition of a test programme is maturing with the following aims: (i). design verification, (ii). investigation of tolerances to e.g. misalignment, sticking of blanket segments and other issues, and (iii) investigation of reliability growth through repetitive testing. This presentation aims at providing an overview of the work carried out so far and future plans.

Keywords

Breeding blanket, Remote handling, CFETR, DEMO.

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P3D3 ABSTRACT-c4ad
I. Repair and Maintenance

Application of optimal control in remote maintenance of tokamak breeding blankets

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Model-based optimal control uses knowledge of system properties in determining actuator commands to achieve a chosen objective. In remote handling (RH) robotics, a typical control objective is to minimise payload tracking error throughout a desired trajectory. A model-based optimal control approach employs real-time numerical optimisation to calculate and execute a best-case robot trajectory that accounts for obstacles, actuator limitations and system dynamics. With recent advances in computing technology, this approach has seen increasing use within robotics, especially for control of complicated, non-linear systems with many degrees of freedom.

Robotic handling of Breeding Blankets (BB) is a significant challenge in future tokamaks such as EU-DEMO. An individual Breeding Blanket segment could be upwards of 80 tonnes in mass and 10 metres in length; these components require efficient manoeuvring during in-vessel maintenance. With extremely tight positional constraints, any structural flexibility of the combined BB and RH system must be carefully considered during manoeuvring. This highlights the need for a model-based control system that can predict and counteract oscillatory robotic behaviour. The evolving nature of concepts for future tokamaks motivates us to develop control techniques that are also agnostic to robot configuration.

We discuss the application of model-based optimal control for two relevant BB RH arrangements: firstly, a single manipulator system that exclusively handles BB segments from above. Secondly, we present and analyse a dual manipulator system that uses a lower manipulator to improve BB control using load sharing. For each scenario, we use simulation and experimental robot test-rigs to prove the advantages of model-based optimal control over classical control techniques.

Keywords

Remote Handling, In-Vessel Maintenance, Robotics, Optimal Control, model-based control, Breeding Blankets.

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P3D4 ABSTRACT-b560
I. Repair and Maintenance

In-Vessel Inspection System: Development and Testing activities of high vacuum and temperature technologies for Fusion Remote Handling

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The In-Vessel Inspection System (IVIS) carry out in-service visual inspection of SST-1 like tokamak under vacuum in between the plasma shots. The IVIS system is ~4m long with 06- DOF, comprising of five rotary joints & one linear motion for deployment within the tokamak. The IVIS is designed to handle a cantilevered payload of ~1kg with a positional accuracy of <2mm. IVIS is initially stowed in a ~4.5 m long Ultra-High Vacuum (UHV) Storage Vacuum Chamber (SVC) isolated from the Vacuum Vessel (VV) by an UHV gate valve. During viewing, the gate valve will open so that IVIS can be deployed inside the VV, complete the viewing procedure & return back to its initial position outside the VV. IVIS structural components are five links of arms made of aluminium alloy 6061-T6 and base link, lugs, stiffeners and shafts made of SS 304L. Each module can provide rotation (±90◦in the horizontal plane). With varying lengths, simulations were performed in virtual reality model to arrive at the present configuration. Based on the VR simulation, it was found that IVIS with a total of 05 links, 01 fixed link & maximum link lengths of 0.7m (joint to joint) is sufficient to encompass the 180° sector inside SST-1 like machine.

A feasibility assessment of suitable technologies to operate the IVIS under the SST-1 like conditions of vacuum and temperature were performed. The IVIS components sustains ~100°C during VV inspection and UHV (<1×10^{-7} mbar) for its conditioning prior to entering the VV. Limits on out-gassing inside the VV impose serious constraints for the design. The COTS components selection and availability (actuators, bearings, encoders, vacuum feed-throughs & cable tray) to match the performances compatible with high temperature, UHV requirements and to overcome pollution issues of the tokamak environment are discussed in this paper.

Keywords

Fusion Remote Handling, Vacuum, High temperature, Maintenance.

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P3D5 ABSTRACT-a9f2

I. Repair and Maintenance

Balanced-risk analysis in the engineering design of complex systems with extreme conditions

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The engineering design of components and systems in a fusion power plant, an example of a complex system with extreme conditions, requires careful consideration and balance of the numerous associated risks. Several key risks are observed from several perspectives and are categorized as failure modes, situational and quality risks. A failure mode is a way a product's component, assemblies or subsystems might fail and is typically derived through relevant historical data. Failure modes could be avoided through design optimization and material selection. Situational risks come from unaware scenarios during plant operation and maintenance. Situational risks could be mitigated by improving the product's precision, payload, or control algorithms. Quality risks are related to the effectiveness of the operation or maintenance. It reflects how the component or the system impacts the plant's production performance. A good illustration is when a remote maintenance system takes too long to complete maintenance tasks, resulting in poor key indicators such as Mean Time Between Failures (MTBF), Mean Time to Repair (MTTR), or Machine Availability (MA). Several methods exist for the separate consideration of these risks at the various design phases. However, an optimum approach is to consider all risks and balance the effects during the conceptual design phase. A single methodology is developed to consider all associated risks at the conceptual design phase by combining the balance of plant (BOP) methodology and a generalized Π-theorem and dimensional analysis. The method allows for concurrent consideration and analysis of all risks for risk-balanced evaluation with the intent of modelling the imbalanced risk analysis problems at the conceptual stage.

Keywords

Risk, failure mode, situational risk, quality risk, fusion, balance of risk (BOR).

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P4A1 ABSTRACT-ce5d
B. Blanket Technology

Plan on Breeding Blanket Test Facility for DEMO in Korea

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

Korea Institute of Fusion Energy (KFE) is conducting a pre-conceptual study on the R&D infrastructure for the development of fusion fuel system technology for DEMO, called Korea Fusion Engineering Advanced Test Complex (KFEAT), with collaborators. This mainly consists of the Integrated Breeding Test Facility, the Blanket System Test Facility and the Fuel Cycle Pilot Facility. The Integrated Breeding Test Facility, whose primary purpose is for the blanket component test, is based on a 40 MeV deuteron accelerator-driven system with maximum 10 mA for fusion-like neutron generation and testing, possibly, one-to-one scale breeding unit of DEMO blankets. The Blanket System Test Facility is to demonstrate the reliability and safety of the blanket and its ancillary systems for DEMO-relevant long-term operation in a non-nuclear environment. The Fuel Cycle Pilot Facility aims to verify the continuous operation performance of a few kg tritium treatment envisaged in DEMO using hydrogen and deuterium with 1/10 pilot scale. In this study, overall plans for the KFEAT and current design status are addressed.

Keywords

Breeding blanket, blanket component test, fuel system, fuel cycle, DEMO.

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P4A2 ABSTRACT-f485
B. Blanket Technology

Perspectives of a DCLL blanket for a future strong magnetic field fusion device

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Selection of a right breeder and a blanket concept is vitally important to the successful development of future fusion devices, in particular to the US fusion pilot plant (FPP). While the FPP concept is being evolved, an anticipated new feature of the FPP is a stronger plasma-confining magnetic field. The strong magnetic field puts in questions feasibility of conventional liquid metal (LM) blankets. The paper reports results of the exploratory studies for a Dual Coolant Lead Lithium (DCLL) blanket for a strong magnetic field of B=10-20 T, whose impact on the LM breeder has not been studied before. In the present analyses for the DCLL blanket, the geometry, materials and thermal loads are adopted from the existing blanket design for the US Fusion Nuclear Science Facility (FNSF). The obtained numerical results for MHD flows, heat transfer, and corrosion of structural RAFM steel for the reference DCLL blanket at PbLi Tin/Tout=350°C/550°C suggest the maximum allowable magnetic field around 16 T. In such conditions, the requirements on the MHD pressure drop < 2 MPa and corrosion of RAFM walls < 20 μm/year are both met. High temperature DCLL blanket at Tin/Tout=350°C/700°C was shown to exhibit acceptable MHD pressure drop even at B=20 T but the corrosion losses were found to be unacceptably high regardless the magnetic field strength.

Keywords

DCLL blanket, lead lithium, MHD, heat transfer, corrosion, fusion pilot plant.

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P4A3 ABSTRACT-b187
B. Blanket Technology

Progress in the Conceptual Design of the Supercritical CO2 Cooled Lithium-Lead Blanket and the Power Conversion System for CFETR

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A design of the supercritical CO_2 cOoled Lithium-Lead (COOL) blanket has been proposed as one advanced blanket candidate for the Chinese Fusion Engineering Testing Reactor (CFETR). It features the single-module segment of RAFM/ODS steel box with several sub-breeding zones. Self-cooled PbLi with an outlet temperature of 600–700 °C flows in the breeding zones and the supercritical CO_2 of 350–400 °C cools the FW and the structures. Besides, the electric and thermal insulating SiC_f/SiC composites are utilized as the Flow Channel Inserts (FCIs) to isolate the high-temperature corrosive PbLi and mitigate the MHD effect.

In the contribution, progress in the conceptual design of the COOL blanket and its power conversion system (PCS) will be reported. The irradiation damage and the radioactive waste has been assessed by a 3D neutronic model. Maximum irradiation damage is 14.5 DPA/FPY at the first wall and PbLi contributes over 60% of the total radioactive waste. And, the segment inlet has been optimized to simplify the structure and decrease the MHD effect. Besides, the accidental analysis has been performed and results show that the plasma should be shut down in time to prevent the blanket overheating under the Loss of Flow Accident. Finally, a preliminary scheme of the PCS based on the S-CO_2 recompressing cycle is proposed for the COOL blanket. And, an Intermediate Heat Transfer System (IHTS) equipped with the Energy Storage System (ESS) is used to connect the Primary Heat Transfer System (PHTS) and the PCS. The gross thermal efficiency is estimated to be 39%–46% at a turbine outlet temperature of 550–650 °C.

Keywords

CFETR, COOL Blanket, Conceptual design progress, Power conversion system.

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P4A4 ABSTRACT-eb7f
B. Blanket Technology

Embrittlement of WCLL Blanket and Its Fracture Mechanical Assessment

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In the European fusion programme, the Water Cooled Lithium Lead Breeding Blanket (WCLL-BB) uses EUROFER as structural material cooled with water at temperatures comprised between 295-328°C and a pressure of 155 bar. The WCLL-BB will be significantly irradiated while some parts will not receive significant heat loads, e.g. the sidewalls or the back-supporting structures. The irradiation together with an irradiation temperature of EUROFER below 350°C produces a shift of the ductile-to-brittle-transition temperature (DBTT) to levels above room temperature at neutron doses causing material damage as low as 2-3 dpa. Even though the DBTT does not reach the operating temperature level, there is a concern that the re-pressurization of the WCLL-BB cooling loop or deadweight loads during maintenance would cause its fracture, since it can experience temperatures below the DBTT. The embrittlement of the WCLL-BB was investigated by quantifying the local DBTT shift in its parts based on nowadays knowledge of the embrittlement behaviour of EUROFER under neutron irradiation. Therefore, a sufficiently, not overly conservative procedure was derived considering dpa damage and transmuted helium effects. The results demonstrate the ability to identify the 3D-spread of the severely embrittled zones in the structure whose impact on the structural integrity was assessed considering the risk of brittle/non-ductile fracture. Thereby, the fracture mechanics approach established in nuclear codes was applied assuming its applicability to EUROFER. The embrittled zones in the First Wall (FW) and its sidewalls pass the criteria when assessing the relatively small stresses resulting from the coolant pressure. The assessment was then continued considering stresses appearing in the FW during maintenance, in particular when lifting the WCLL-BB segment and transporting it out of the Vacuum Vessel. In this context, the maximum tolerable flaw sizes were determined in a parameter study in which FW designs with different cooling channel wall thicknesses were considered.

Keywords

Neutron irradiation, embrittlement, WCLL, breeding blanket, brittle/non-ductile fracture, fracture mechanical assessment.

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P4A5 ABSTRACT-4474
B. Blanket Technology

An integral methodological framework for the thermo-mechanical analysis and structural integrity assessment of the European TBM Sets

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Two European Test Blanket Modules (TBM) concepts are being developed by F4E, the Water-Cooled Lithium Lead (WCLL) and the Helium-Cooled Ceramic Pebbles concept (HCCP, in collaboration with ITER KOREA). For both concepts, the ESTEYCO mechanics team is involved in the design and analysis of the TBM-box, which is the actual plasma-facing module for tritium breeding, as well as the TBM shield in the WCLL concept. The design-by-analysis activities of the TBM-Sets must be performed in a very challenging context due to: (i) the variety and characteristics of the loads, including complex evolutive thermal loads, fast transient EM events, high pressure loads or seismic loads, (ii) the amount and complexity of the failure modes to be assessed, (iii) the use of a new structural material under development (EUROFER 97) with currently some limitations in the information available for material properties and design rules and (iv) the magnitude of the loads in a restricted design space, resulting in tight compliance margins that often prevent the use of simplified approaches. In the past years, Esteyco has developed advanced methodologies to virtually test alternative design solutions in a consistent, integral and agile manner and linked to the applicable nuclear design code (RCC-MRx). This paper presents an overview of the most relevant developments for the simulation of the TBM-Set under the complete set of loads and the subsequent structural integrity assessment with an unprecedented level of comprehensiveness and detail. Novel and automatic tools have been developed covering the complete process, including the consistent distribution of neutronic thermal loads, the transient simulation of the cooling system in operation conditions, the characterization and distribution of EM loads, the complete implementation of the code assessment rules for a great number of evaluation points and the development of powerful visualization tools to analyse and validate the results computed.

Keywords

Test Blanket Modules, Structural Integrity Assessment, F4E, ITER.

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P4B1 ABSTRACT-31aa
C. Fuel Cycle and Tritium Processing

UNITY: The integrated testing facility for commercial breeder blankets and tritium fuel cycle

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The UNITY facility is currently being fabricated in Japan. It will support integrated testing of components required for the primary and secondary thermal cycles for power generation and fuel cycle of early fusion power plants. The thermal section provides heating capacity for blanket modules up to 0.1 m^2 plasma-facing surface area. Of the three liquid coolants supported (Li, LiPb, FLiBe), a LiPb loop will be connected first. A uniform magnetic field of up to 4 T can be generated with a pair of NbTi magnets for liquid metal MHD testing. A plasma exhaust pumping system for the inner fuel cycle, direct internal recycling, a fuel cleanup system, tritium extraction from the coolant, and storage will be integrated into one system, using deuterium as a proxy for tritium. At the same time, a second generation facility using actual tritium will be built at and in collaboration with the Canadian Nuclear Laboratories. This new facility is intended to be the first demonstration of a closed fuel cycle for application in a commercial DT fusion reactor independent of the confinement method.

This presentation will detail the feasibility due to heating, cooling, and power constraints, the current fabrication status, the roadmap, and ongoing experimental activities.

Keywords

Testing facility, tritium extraction, closed fuel cycle, vacuum pumping.

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P4B2 ABSTRACT-1fb0
C. Fuel Cycle and Tritium Processing

A quantitative case for direct internal recycle of deuterium and tritium

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Nuclear fusion is a potential source of carbon-free energy that uses hydrogen isotopes deuterium and tritium as fuel. About 5% of the fuel injected into a fusion reactor undergoes fusion, leaving a massive fraction that can be reinjected into the reactor. The ability to recycle this fuel efficiently is imperative for limiting tritium inventory, and therefore cost, safety, and feasibility of a fusion reactor. Reducing processing time for any portion of the fuel will reduce the required inventory by increasing the portion of fuel available for continuous operations. Non-hydrogen gas removal, bound hydrogen catalysis, and isotope separation could be a time-consuming process. Direct recycling greatly reduces the tritium processed through isotope separation, which reduces the size, and hence, inventory of tritium in this system, and it reduces the amount of tritium on storage beds. As a result, recycle systems that reduce residence time in these potentially avoidable steps is of particular interest for fuel cycle studies. The use of hydrogen-permeable membranes allows for complete removal of any other gasses from deuterium and tritium. The use of permeators containing these membranes would facilitate the reintroduction of fuel into the injection system with reduced processing (to include removal of non-hydrogen streams and potential doping with purified streams), while some portion of the reactor exhaust goes through additional processing steps to recover and separate the remaining traces of deuterium and tritium. A reduced-order permeation model, validated by a permeation experimental campaign, will define the advantages of this approach by way of systems analysis. A thorough understanding of the processing steps needed for the fuel cycle is the cornerstone of eventual fuel cycle design, which will need to encompass inventory and scheduling considerations.

Keywords

Direct internal recycle, tritium separation, permeation.

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P4B3 ABSTRACT-89ce
C. Fuel Cycle and Tritium Processing

Admixed pellets for fast and efficient delivery of plasma enhancement gases: investigations at AUG exploring the option for EU-DEMO

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Gas and pellet injection is envisaged for particle fuelling in EU-DEMO. The gas system will provide edge and divertor fuelling and any further gas species required for operation. At reactor scale, a gas-system response time of several seconds is expected, too slow to respond to events such as transient divertor reattachment. The pellet system can respond on much faster time scales (0.1s). Pellets, mm-sized bodies formed from solid hydrogen fuel, are designed for efficient and fast core fuelling. However, they can also be employed for a more efficient delivery of plasma enhancement gases, by admixing them with the fuelling pellets. To check this option for EU-DEMO, explorative investigations have been performed at ASDEX Upgrade (AUG).

The AUG system produces ice in a batch process sufficient for about 100 pellets, initially designed for operation with pure H_2 or D_2. On a trial basis, pellet formation was tested using an H_2/D_2=1:1 mixture and admixtures containing small amounts (up to 2 mol%) of N_2, Ar, Kr or Xe in the D_2 host. A homogeneous and reproducible ice composition was found for the H_2/D_2=1:1 case. For all the admixed gases, a depletion of dopant in the ice with increasing atomic number is observed. Nevertheless, the fast and efficient delivery of admixed pellets was clearly demonstrated in dedicated plasma experiments. Detailed investigations showed that the Ar supplied via admixed pellets has a higher radiation efficiency and a faster radiation rise than an Ar/ D_2 gas puff. Investigations performed at the Oak Ridge laboratory with a large batch extruder using up to 2 mol% Ne in D_2 confirmed that production of much larger ice quantities can be achieved.

The initial explorative investigations clearly reveal the great potential of admixed pellets, although they also demonstrate that further technology efforts are required before their benefits can be utilized.

Keywords

ASDEX Upgrade, EU-DEMO, Fuelling, Pellet.

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P4B4 ABSTRACT-4c71
C. Fuel Cycle and Tritium Processing

Process simulation of Tritium Extraction System in CN HCCB TBS

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Helium Coolant Ceramic Breeder Test Blanket Module System (HCCB TBS) is a Chinese contribution to the ITER project. This paper is an overview of system process simulation of Tritium Extraction System (TES), which is one of the important auxiliary systems in HCCB TBS. The simulation consists of several components, such as chemical reaction beds, pumps, filters, and various other components. The process simulation of TES is a complex procedure to accurately replicate real-world conditions. It involves a series of sophisticated calculations and the set up of a suitable environment for the process. To simulate this process, in the first step, basic properties of the system are identified such as temperature, pressure, electric field etc. Then these parameters are adjusted according to the requirements. In the second step, the values of each variable have to be determined in order to obtain the right conditions for a successful simulation. Once the conditions are set, the simulations can then start and be monitored instantly. Finally, different scenarios of the extraction process are simulated to evaluate the performance of TES and analyze the various outcomes. Simulated results of the tritium extraction system provide invaluable information regarding its efficiency and effectiveness which is then used to make further improvements in the design and operation of the system for optimization.

Keywords

HCCB TBS,Tritium Extraction System,Simulation.

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P4B5 ABSTRACT-ce91

C. Fuel Cycle and Tritium Processing

Research progress of tritium related system in CN HCCB TBS

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This paper presents the research and development progress of Tritium related system in China Helium Coolant Ceramic Breeder Test Blanket Module System, CN HCCB TBS), which is one of the blanket concepts to be tested in ITER facility. Tritium related system mainly includes Tritium Extraction System (TES), Tritium Accountancy System (TAS), and Coolant Purification System (CPS). The design process of the system was discussed, from initial concept to final realization. The progress made in the design and verification phase of the tritium related system was presented. The emphasis was put on the choice of materials and components for the construction and installation of the system, as well as the system's control and monitoring functions. The performance and reliability of the system were tested and verified in some respects based on the dedicated test platforms and facilities to ensure its safe operation. The results of research and analysis showed that the system design is reliable and effective for CN HCCB TBS tritium related system. But there are still some key issues that need to be addressed in the future. The knowledge gained from the study will benefit future work of the projects involving the design, verification and construction.

Keywords

HCCB TBS, Tritium process, R&D, Preliminary Design.

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P4C1 ABSTRACT-7697
K. Inertial Confinement Fusion Studies and Technologies

Asymmetric growth of peak and valleys in the linear phase of the Rayleigh-Taylor instability in solid media

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Rayleigh-Taylor instability is a key issue in many experiments on high-energy density physics (HEDP) and in inertial confinement fusion. Some implosion driven experiments like LAPLAS (Laboratory Planetary Science), being designed in the framework of the international collaboration on HEDP developed around the heavy ions accelerator FAIR (Facility for Ion and Antiproton Research, presently under construction at the GSI Darmstad (Germany), relay on the mechanical properties of the pusher for the control of the hydrodynamic instabilities. The generation of spikes and bubbles, a typical characteristic of the non linear regime in the Rayleigh-Taylor instability, is found to occur as well during the linear regime in an elastic-plastic solid medium caused, however, by a very different mechanism. This singular feature originates in the differential loads at different locations of the interface, which makes that the transition from the elastic to the plastic regime takes place at different times, thus producing an asymmetric growth of peaks and valleys that rapidly evolves in exponentially growing spikes, while bubbles can also grow exponentially at a lower rate.

Keywords

High energy density physics; Hydrodynamic instabilities; Rayleigh-Taylor instability.

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P4C2 ABSTRACT-ca12

K. Inertial Confinement Fusion Studies and Technologies

Xcimer Energy Aspen-HYLIFE3 Inertial Fusion Power Plant Concept

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The Xcimer HYLIFE-3 Inertial Fusion Energy (IFE) concept combines a well-studied IFE liquid first wall protection chamber concept with a novel new Krypton Fluoride (KrF) laser to enable a promising fusion power plant concept. The Xcimer laser presents a unique large energy (~12 MJ) laser architecture that substantially reduces the cost of the IFE driver by using Stimulated Raman Scattering (SRS) and Stimulated Brillouin Scattering (SBS) technologies for beam combining and pulse compression in lieu of glass resulting in about $30/J. This paper summarizes the key enabling features of the Xcimer laser towards a fusion chamber that uses molten salt to protect the plasma facing components using two laser beamlines at a repetition rate 1 Hz. We describe the unique jet array/formation required to protect the lifetime steel first wall. We describe the unique target irradiation approach which employs so-called 'hybrid' drive using both indirect and direct drive and provide predicted gain results. This paper focuses on the nth of a kind mature power plant with a fusion yield of ~2.3 GJ and a 1 Hz repetition rate for an overall approximately 960 MWe of net electrical power. We describe plans to experiment to validate the lifetime of the first structural wall using a pulsed fission reactor. We discuss the integration of the two laser beam lines with the chamber where gas at an atmosphere is kept separated from the chamber under vacuum, and the approach to inject, track, and irradiate the IFE target. Ongoing work on target design and fabrication with an outer thick solid FLiBe cylinder and an interior high Z layer and capsule is presented. We describe our methods to extract tritium and impurities from the FLiBe. We conclude by summarizing current costing that predicts ~ 4 cents per kilowatt-hour.

Keywords

Krypton fluoride, thick liquid wall, FLiBe, Hybrid drive, ASPEN, HYLIFE3.

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P4C3 ABSTRACT-17ae
K. Inertial Confinement Fusion Studies and Technologies

Experimental studies of extended-MHD effects and confinement properties of magnetized cylindrical implosions

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The recent demonstration of fusion breakeven at the National Ignition Facility (NIF) is a major milestone towards fusion energy. The addition of a background magnetic field (B-field) is appealing to further increase fusion yields. The B-field compressed with the target acts in addition to inertia to confine the hot spot, resulting in a hotter fuel, allowing to ignite at lower areal densities than otherwise possible and with slower implosions that are less susceptible to hydrodynamic instabilities.

A cylindrical implosion platform facilitates less convoluted analysis of the magnetized transport of heat and magnetic flux and measurements of the imploding and stagnated plasma conditions, compared to spherical implosions. We present the design, numerical simulations (extended-MHD, atomic line shape and radiation transport), and experimental data of magnetized cylindrical implosions performed at the OMEGA laser. The cylindrical targets are filled with Ar-doped D_2 and imploded by a 14.5 kJ, 1.5 ns laser drive, with or not an axial B-field of 30 T. The compressed core conditions are obtained from highly reproducible Ar K-shell emission spectra showing distinct changes between the cases with and without the applied B-field. When magnetizing the implosions, the compressed B-field reaches ~10 kT and the subsequent reduction of the heat conduction along the radial direction enhanced the average temperature of the hot spot by ~70% (from ~1 keV to ~1.7 keV), a record in magnetized implosion experiments.

Proposals to scale this platform to 300 kJ of laser energy-drive have been accepted at both Laser Mega Joule (LMJ) and NIF, enlarging the level of achievable magnetizations and foreseeing diagnostic improvements, namely the use of dual dopant (Ar and Kr) spectroscopy to achieve effective spatial resolution of the core temperature, and B-field compressibility measurements from angularly-resolved spectra of secondary neutrons.

Keywords

Inertial fusion, magnetized implosions, extended-MHD, strongly magnetized HED plasmas.

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P4C4 ABSTRACT-e5e4
K. Inertial Confinement Fusion Studies and Technologies

Developments on inertial confinement power plants based on dry wall chambers for direct drive targets: bottlenecks and solutions

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In parallel to the recent sound research on ignition strategies, target fabrication and new laser concepts, the development of functional inertial fusion plants require to identify and solve a number of important problems. Both scenarios, direct and indirect drive illumination, have implications and unsolved issues regarding plant development. In this presentation, we focus on the development of plants for direct drive targets based on the work carried out by our group in the framework of the European HiPER project.

The HiPER demonstration plant scheme is based on an evacuated dry wall chamber and silica transmission lenses as the main components of the final optics. This scheme is compatible with target injection and beam propagation, however, we will discuss a number of problems regarding the chamber materials, final optics and breeding blanket.

- The chamber is exposed to detrimental conditions. In particular, the first wall materials, in this unprotected scheme, must withstand very intense irradiation conditions that lead to materials degradation and light species retention. No studied material can fulfil the strict operation conditions. We have studied a new material based on highly porous nanostructured tungsten with promising properties for the first wall.
- The final optics materials are unavoidably exposed (at least) to neutron irradiation that leads to a non-uniform temperature distribution resulting in unacceptable aberrations. We have designed a final optics system able to maintain the temperature of the final lenses constant and uniform as required for operation.
- Excess or deficient tritium production in the breeding blanket constitute a safety issue. We have developed a breeding blanket concept based on lithium ceramics able to modify the tritium production during operation with the aid of a water tank that changes its neutron reflection properties as the filling level changes.

Keywords

Inertial Confinement Fusion, Power Plant, Dry Wall Chamber, Plant systems, First Wall, Final Optics, Breeding Blanket.

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P4C5 ABSTRACT-b040
K. Inertial Confinement Fusion Studies and Technologies

The CLPU: a high repetition rate, high-intensity laser facility for ICF preparatory studies

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The CLPU is the Spanish Centre for Pulsed Ultra-Intense Lasers, located in Salamanca and part of the Spanish roadmap of strategic scientific infrastructures. The CLPU has been fostering scientific and technological activities aimed to the exploitation of high repetition rate Lasers that will certainly contribute to the development of future efficient schemes for Inertial Confinement Fusion (ICF). Several experiments have been investigating plasmas and nuclear reactions of interest for future schemes of ICF. More in detail, several classes of real-time detectors aimed to study laser-plasma interactions, as much as secondary sources have been tested and developed in house. The development of magnetic compact transport lines is an additional asset aimed at studying and employing secondary charged particle beams. The development of reliable high repetition secondary sources is a further ingredient necessary to extend the capabilities of these facilities.

Keywords

High Intensity, High Repetition, Secondary sources, Real time detectors.

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P4D1 ABSTRACT-4357
H. Models and Experiments for FNT

Liquid metal MHD research at KIT: fundamental phenomena and flows in complex blanket geometries

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Lead lithium (PbLi) is foreseen in fusion blankets as breeder material to produce tritium and as heat transfer medium for power extraction. In the European concept for a water-cooled lead-lithium blanket, the volumetric heat released by neutron energy deposition and absorption is removed by water flowing at high pressure through channels embedded in walls and in cooling pipes immersed in the eutectic alloy. The liquid metal is circulated via manifolds and pipes to ancillary systems for tritium extraction and purification. The PbLi flow is driven by applied pressure differences under the influence of buoyancy due to temperature differences between liquid metal bulk and water-cooled surfaces. Furthermore, the motion of the electrically conducting fluid within the plasma-confining magnetic field is affected by Lorentz forces that arise from flow-induced electric currents. They cause high pressure drop, peculiar velocity profiles, and electromagnetic coupling of neighboring fluid regions. To assess the impact of these effects on the performance of liquid breeder blankets, a deep understanding of complex magnetohydrodynamic (MHD) phenomena assisted by validated predictive capabilities is required.

Liquid metal MHD research at KIT supports blanket design activities through theoretical and experimental investigations. Predictive computational tools are developed and validated by empirical data obtained for fundamental problems such as flows in non-uniform magnetic fields, pressure drop reduction by insulating flow channel inserts, or magnetoconvection around cooling pipes. Two complementary facilities (MEKKA and MaPLE) available at KIT provide a unique and versatile platform for MHD investigations at fusion relevant parameters. Using NaK as a model fluid in MEKKA allows conducting experiments at high Hartmann numbers in large complex geometries such as scaled blanket mock-ups. For magnetoconvection and heat transfer studies, MaPLE is more suited since it enables experiments with the prototypical fluid (PbLi) in test sections inclined at various orientations with respect to gravity.

Keywords

Liquid metal blankets, magnetohydrodynamics, PbLi experiments and modeling.

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P4D2 ABSTRACT-0d07
H. Models and Experiments for FNT

Overview of MHD facilities, Experiments and Modelling in China

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Recently, both from engineering and fundamental aspects, China has made great efforts in designing a nuclear fusion reactor which has been regarded as a suitable energy source for the future. Among the components in the Tokamak reactor, the liquid metal plasma-facing components and the liquid metal blanket are two important ones that may face magnetohydrodynamics (MHD) problems. Here, we give a comprehensive review of the investigations of MHD from experiments and modeling in China. Regarding the experiments, the liquid metal loops, Magneto-Thermo-Hydrodynamic (MaTH) loop, utilize the Galinstan, lithium as working liquid have been built to study the MHD effects in liquid metal film flow, and liquid metal thermal convection. It is found that, under the influences of the transverse magnetic field, the free surface waves are strongly suppressed along the spanwise direction, leaving the surface waves to propagate in the streamwise direction. Furthermore, a scaling law $\delta_{en}*\sin\beta \sim N$ to quantitative identify the influence of magnetic field on the relative film thickening δ_{en} has been given by experiments and theoretical analysis. With respect to the MHD modeling, a highly accurate and efficient numerical simulation platform known as MHD@UCAS which has the capability to model the fluid dynamics and thermal characteristics of liquid metals under strong magnetic fields, even in complex configurations has been developed. Our study has revealed the order of magnitude of MHD pressure drop inside the blanket module (~2MPa), demonstrated the crucial impact of buoyancy on liquid metal flow and heat transfer inside the blanket, highlighted the safety risks associated with ignoring thermal effects in research, and discovered the coexistence of three-dimensional and quasi-two-dimensional turbulent structures inside the blanket. Additionally, based on a thorough flow analysis, we propose an optimal flow channel design strategy that involves increasing the length of the inlet baffle.

Keywords

Liquid metal, plasma facing components, liquid metal blanket, magnetohydrodynamics.

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P4D3 ABSTRACT-4 a18
H. Models and Experiments for FNT

Status and applications of the RSTM tool for coupled CFD-activation fluid simulation

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Accurate prediction of the activation of fluids flowing under irradiation is important for the timely development of fusion technology. One of the major current issues is the cooling water of ITER, which becomes activated (N-17, N-16, O-19, H-3, C-14) by plasma neutrons during nuclear operations, thus raising a number of concerns related to radiological implications such as radiation effects on sensitive equipment (e.g. magnets, insulators, sealants, electronics), compliance with radiological protection zoning, and compliance with radioactivity limits in regulations of pressure equipment and effluents. Further significant applications are the activation of other service fluids in ITER and of the LiPb in both ITER and DEMO breeding blanket modules.

To avoid systematic uncertainties, simulation of the activation of fluids flowing in arbitrarily complex 3D geometry, flow regimes and neutron fields requires full coupling of activation with fluid-dynamics physics models. No such tools were available until recently: the Radio-Species Transport Model (RSTM), based on the well-established ANSYS Fluent® UDS methodology, is one conceived and developed at F4E.

Here we review the methodology and capabilities of the RSTM, as well as earlier validation, benchmarking and application activities. We then report currently ongoing further applications and benchmarking being performed in collaboration with specific tasks of the EUROFusion programme Preparation for ITER Operation (PrIO): (i) the design and construction of the JSI water activation loop, and (ii) the dedicated water activation experiment during the planned JET DTE3 campaign. RSTM computations, results and comparison with other methodologies for several cases of interest are presented and discussed

Keywords

Radiation shielding, neutronics, fluid-dynamics, activation.

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P4D4 ABSTRACT-4cfd
H. Models and Experiments for FNT

Development of HINEG Series High Intensity Steady Neutron Generators

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Neutron sources are important experimental platforms for R&D of advanced nuclear energy systems, especially for fusion systems. Series High Intensity Steady Neutron Generators (HINEG) have been developed in China for missions including neutronics design validation, materials & components irradiation test, nuclear waste burning and nuclear technology applications, etc. HINEG includes three phases: HINEG-I, HINEG-II and HINEG-III.

HINEG-I includes two sources, HINEG-Ia and HINEG-Ib. HINEG-Ia, which has already operated, is a D-T fusion neutron source with yield of 6.4×10^{12} n/s, and coupled with the Lead-based Zero Power Reactor CLEAR-0. HINEG-Ib is a cyclotron-based neutron source with yield over 10^{14} n/s and is under construction now. HINEG-I is a platform for fusion neutronic design, and fission reactor shielding design validation. It also can be used for extended nuclear technology applications, such as radiotherapy, radiography, isotope production, etc. Series experiments have been carried out on HINEG-I, including neutronics performance test of TBM blanket, leakage spectra measurement with Pb and Pb-Bi, irradiation damage testing for laser crystal, etc.

HINEG-II includes two sources, HINEG-IIa and HINEG-IIb. HINEG-IIa is accelerator based D-T neutron source with yield over 10^{13}n/s. HINEG-IIb is a cyclotron-based spallation neutron source with yield over 10^{15}n/s. These two sources are aimed to provide multi-type of neutron spectra and high neutron intensity for radiation damage mechanism study, advanced reactor technology validation, and extended nuclear technology application research, etc. The design of HINEG-IIa have been finished, the construction and assembly are on-going. For HINEG-IIb, design and key technologies R&D are on-going.

HINEG-III is high flux steady state neutron source with intensity of 10^{17}-10^{18} n/s. It will be coupled with a subcritical reactor with neutron flux higher than 10^{15} n/cm^2/s. HINEG-III is a multi-purpose neutron irradiation platform for irradiation testing of fuel, material and components for advanced reactors. The conceptual design of HINEG-III is on-going.

Keywords

High Intensity, Neutron Sources, HINEG, Neutronics.

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P4D5 ABSTRACT-6d39
H. Models and Experiments for FNT

The scaling methodology applied for designing HELOKA-US facility, the EU-DEMO HCPB BOP mock-up

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HELOKA-Upgrade Storage facility will be the first pilot test plant to replicate the operational behavior of the EU-DEMO HCPB Indirect Coupling Design (ICD) Balance of Plant (BOP) providing the experimental insights of the Primary Heat Transfer System (PHTS) and the Intermediate Heat Transfer System (IHTS) under DEMO operation modes (pulse, dwell and corresponding transitions). The ICD is based on an Energy Storage System (ESS) in the IHTS leveling the effects of the tokamak thermal power source variation on the Power Conversion System (PCS). Currently, HELOKA-US is under design and construction at the Karlsruhe Institute of Technology.

HELOKA-US pilot plant is composed of a Molten Salt loop coupled to a Helium loop as heat source and to a Water Cooling System as heat sink. The scaling methodology used to design HELOKA-US is challenging due to the operational requirements of DEMO HCPB ICD BOP. The degrees of freedom in the scaling analysis are the thermal power level and physical dimensions provided that the similarity to the prototype is assured as well as the heat transfer performance is maximized while minimizing pumping power, amount of coolant volume, amount of structural materials, heat loss and temperature drop.

This paper presents the HELOKA-US scaling analysis including the applied scaling methodology, the figures of merit and the distortions between the prototype and the scaled facility and components. Furthermore, the final conceptual design of the test facility able to simulate the flow and heat transfer behavior of the prototypical system under investigation is presented.

Keywords

BOP, HCPB, HELOKA-US, ESS.

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P4E1 ABSTRACT-14ca
L. Fission-Fusion Synergy and Cross Cutting Technologies

Fusion Fission Hybrid as a Synergistic Step Between Fission and Fusion Energy Development

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Fusion-fission hybrid is a solution to overcome difficulties of fission energy development with the features of energy production, fuel breeding and waste transmutation. It is a more feasible approach to the early fusion energy application than pure fusion reactors considering of its low fusion parameters. Meanwhile, fusion-fission hybrid shares lots of vital technologies with fission and fusion reactors. For example, lead-based technology could be employed in both Generation IV fast reactor system and liquid lithium–lead blanket. So, fusion-fission hybrid is a highly promising concept as a bridge between fission and fusion energy development, by using lead-based technology as the intermediate stage.

The contribution will analysis why fusion fission hybrid can be used as a ‘bridge’ between fission and fusion development. From the perspective of fusion, the requirements of fusion driver plasma parameters, materials selection, blanket energy multiplication and tritium breeding performance will be comprehensively analyzed. From the perspective of fission, the role of hybrid reactor in improving its safety and sustainability performance is given, and the feasible range of fuels and materials selection for fission blanket is studied for hybrid. At last, the contribution summarizes typical designs and R&D for hybrid in China.

In conclusion, fusion-fission hybrid could be a synergistic step between generation IV reactor and fusion reactor with lead-based technology as a connection. China has got a series of achievements in the development of fusion-fission hybrids, including conceptual designs and related R&D activities.

Keywords

Fusion, Fission, Fusion Fission Hybrid.

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P4E2 ABSTRACT-82ce
L. Fission-Fusion Synergy and Cross Cutting Technologies

Selected fuel cycles for stellarator-mirror fusion-fission hybrid

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The stellarator-mirror (SM) fusion-fission hybrid consists of a fusion neutron source and a powerful sub-critical fast fission reactor core. The major specialization of the hybrid is transmutation of spent nuclear fuel and safe fission energy production. In its fusion part, a stellarator-type system with an embedded magnetic mirror is used. This hybrid has unique possibilities to burn a wide variety of fuels including fuels which suffer of small number of delayed neutrons. It can realize multiple recycle fuel cycles (MRFC), in which it is possible to turn fissile elements to fission products and generate maximum energy output. They use a chain of cycles in which, in each next cycle, all the elements from previous cycle that have fission properties are used. The only principal output of the MRFC are the fission products.

Here three MRFC for the SM hybrid are considered based on usage the transuranic elements extracted from the light water reactors spent nuclear fuel. For modeling, a simple 0D particle balance model is used which describes the nuclear transformations in the uranium-238 based isotope chain.

The first cycle uses pure plutonium. The MRFC with this fuel has the mostly high k-eff, the ratio of production to loss neutron rates, and for this reason can be used in small-size reactors. Since the highly radioactive americium remains not involved, this MFRC does not contribute much to the spent nuclear fuel transmutation, but could be used in research.

Calculations for the plutonium-americium fuel indicate substantial accumulation of the americium during the MRFC cycles and, for this reason, these MRFC are hard to realize in practice.

A possible solution is usage of the plutonium-americium-uranium-238 fuel in which the accumulation of americium is substantially mitigated and k-eff increases. This MRFC is prospective and is recommended for further theoretical investigations.

Keywords

Fusion, fission, transmutation, fuel cycle.

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P4E3 ABSTRACT-40f0
L. Fission-Fusion Synergy and Cross Cutting Technologies

IAEA Activities in Support of Nuclear Fusion Research and Technology Development

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The main objective of fusion research and technology development is to deliver a safe, economical, essentially unlimited and emission-free energy source. While considerable progress has been made during the last decade, there is still a great deal of research and development needed. The International Atomic Energy Agency (IAEA) fosters international collaboration, coordination and exchange of scientific and technical results to help close the existing gaps in physics, technology and safety with the objective of encouraging further progress in science and technology development as well as facilitating the coordination of the global effort for the fusion energy to become a reality. IAEA's technical activities in support of fusion research are planned and implemented under the guidance of the International Fusion Research Council and include but are not limited to: (i) the series of Fusion Energy Conferences and DEMO Programme Workshops; (ii) coordinated research activities on magnetic fusion and inertial fusion confinement; (iii) technical meetings and workshops on plasma-engineering aspects, divertor design and fusion machine operation, (iv) management and updates of the global Fusion Device Information System (FusDIS), and (v) schools and training courses targeting new generation fusion researchers and engineers. In this talk, these activities will be presented and discussed in detail. In addition, as fusion research and development activities increasingly require cutting-edge technologies, expanding beyond the horizon of present-day know-how, the IAEA aims to broaden its focus in a way to encourage technological and safety-related developments that need to keep up with the pace of scientific knowledge advances. Future strategies and priorities in this new area will be presented and placed in context as part of the wider global fusion research and development effort.

Keywords

Nuclear fusion research, technology development, synergies with other nuclear technologies, international cooperation.

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P4E4 ABSTRACT-f768
L. Fission-Fusion Synergy and Cross Cutting Technologies

A study of compact tokamak hybrid reactors

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The expansion of nuclear power would involve a solution to burn the long half-life transuranics (TRU) in the spent nuclear fuel discharged from LWRs. D-T fusion neutron sources sufficient to drive sub-critical advanced reactors can be an answer for these needs. A sub-critical advanced burner reactor (or breeder reactor) with a fusion neutron source (a "fusion-fission hybrid") will have the advantage of a variable strength neutron source, achieving deeper TRU fuel burnup (fuel residence time limited by materials damage rather than criticality).

Most neutrons in a subcritical transmutation reactor would come from the fission process, the role of the fusion neutron source being to provide a modest number of neutrons to maintain the neutron fission chain reaction. Compact high-field tokamaks can be a candidate for being the neutron source in a fission-fusion hybrid, essentialy due to their design characteristcs, such as compact dimensions, high magnetic field, flexibility of operation, etc. This study addresses the development of a tokamak neutron source for a hybrid reactor using compact ARC-like designs. This tokamak source operates at lower magnetic field values, and does not reach ignition: however, its neutron production is estimated to be fully sufficient for an experimental hybrid reactor. The project could be divided into two main phases:

- Phase 1 – A compact tokamak neutron source for materials irradiation
- Phase 2 – The same neutron source with a blanket component where TRU transmutation takes place and it is demonstrated

Results from neutronics simulations we have carried out show excellent TRU transmutation potential. In this way, fission should explore the possibility of Burning TRU and developing alternatives to GEN III-IV mainstream, such as hybrids, while fusion could shortcut its deployment as an energy source

Keywords

TRU transmutation, fission-fusion hybrid, high-field tokamaks.

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P4E5 ABSTRACT-1872
L. Fission-Fusion Synergy and Cross Cutting Technologies

Laser-driven proton-boron reaction for alpha particles and radioisotopes production

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The fusion reaction P + ^{11}B ==> 3x ^{4}He + 8.7 MeV generates 3 alpha-particles with a total energy of 8.7 MeV opens the possibility of developing a novel generation of high brightness alpha-particle source, with potential interest in astrophysics [1, 2], space propulsion [3] or fusion energy [4] and medical treatment [5, 6] in particular for the production of radioisotopes for either diagnostics (PET imaging) or medical treatment. The alpha particle beam interaction with Ca samples produces Scandium radioisotopes ^{43}Sc Beta+ emitters with a gamma emissions line at 373 keV [5] and ^{47}Sc Beta- emitter, considered as one of the "radioisotopes of the futures", and used for theranostic treatment when combined together.

This work aims at studying different schemes of laser-driven p-^{11}B reaction exploiting the high-power and the high repetition rate of VEGAIII laser to improve the alpha production detection and Scandium radioisotope production. Different setups are proposed: (i) Pitcher-catcher configuration: TNSA protons produced from thin aluminium foil target (Pitcher) interact on a solid B type target (catcher) and (ii) Directly irradiation of solid samples of Boron-nitride (BN) target.

We present in this work the experimental setup and the diagnostics that were used during the campaign, as well as the first preliminary results of the experiment.

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Keywords

Proton-boron fusion reaction; Laser ; Radioisotopes.

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P5A1 ABSTRACT-73e1
B. Blanket Technology

Progress on Blanket Technology Development in China

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Tritium breeding blanket is one of the key components in fusion reactor with main functions of tritium production and heat extraction. Based on national development strategy of magnetic confinement fusion energy, China will test helium-cooled ceramic breeder test blanket module (HCCB TBM) in ITER, and also develop other tritium breeding blanket concepts that could be used for DEMO, i.e., water-cooled ceramic breeder (WCCB) blanket, water/helium-cooled lithium-lead (WCLL/HCLL) blanket, supercritical-CO2-cooled lithium-lead (CCLL) blanket, etc.

In recent years, the preliminary design of HCCB TBM have been developed and passed through the status assessment organized by ITER Organization. Also the HCCB and WCCB blanket concepts have been preliminarily designed for China Fusion Engineering Test Reactor (CFETR) that is one proposed DEMO reactor. At same time the liquid metal blanket concepts, such as WCLL, HCLL and CCLL, are studied as the advanced blanket concepts to enhance performance and economical efficiency of blanket.

Meanwhile, the corresponding R&D have been performed based on the development plan. Currently, the structural and functional materials used for HCCB TBM are being produced and tested for engineering qualification. The other advanced materials are under development based on the general requirements of DEMO blanket. Semi-prototype of HCCB TBM for ITER, prototype of HCCB and WCCB blanket for CFETR were fabricated, which will give big advance for the future DEMO blanket technology. At same time, the test platforms are gradually established, including helium cooling experiment loops, water thermo-hydraulic test platform, liquid-metal test loop and tritium test platforms. Also some new test platforms are under consideration.

In conclusion, under the support by domestic projects and China TBM project, lots of design and R&D activities for different blanket concepts have been implemented, also many testing facilities are constructed to support the design, which will provide indispensable experience for the design of DEMO.

Keywords

Tritium breeding blanket, Design, Material, Fabrication, Testing Platform.

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P5A2 ABSTRACT-f647
B. Blanket Technology

Study on the tritium breeding characteristics of solid blankets of fusion reactors

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Solid Tritium breeding blanket (TBB) is an essential candidate to achieve the tritium breeding in the magnetic confinement fusion reactor. The internal structure, geometric parameters, material properties, configurations between modules and the contour line of the solid TBB will directly cause a fluctuation in the spatial and energy distributions of neutronic performances, and make a significant impact on the tritium breeding characteristics. It is also a challenge for the fusion reactors to realize tritium self-sufficiency with the tritium breeding ratio (TBR) >1. The analysis of the influencing factors on TBR from a macroscopic perspective towards a typical TBB module has been implemented in most of the current related studies. To overcome this challenge, the characteristics of tritium breeding and efficient neutronic optimization mechanism for solid TBB were performed in this paper, respectively. Firstly, a theoretical model for the neutronic characteristics analysis based on Monte-Carlo transport and conjugate calculations was systemically established. Also, the dynamic response of tritium breeding performance under the multi-factor synergy of geometric and material parameters was comprehensively explained. Then, an innovative optimization method for the tritium breeding performance enhancement was put forward by combining the advantages of the simulated annealing algorithm and the hill climbing algorithm. In this way, the efficient and intelligent dynamic tuning of geometric and material parameters was realized. Finally, a **M**ulti-physics **C**oupling and **I**ntelligent **N**eutronic **O**ptimization (MCINO) code for solid TBB was developed. Demonstration applications of MCINO was also performed for code verification. Numerical results indicate that MCINO shows a higher figure of merit (FOM) a better optimization effect and a higher optimization efficiency, which can provide technical support for the neutronic analysis and nuclear optimization for China fusion demonstration reactor or commercial fusion reactors in the future.

Keywords

Fusion reactor, tritium breeding blanket, intelligent neutronic optimization, multi-physics coupling.

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P5A3 ABSTRACT-f7a5
B. Blanket Technology

Electrical properties of ceramic coatings after heavy-ion irradiation and lithium implantation

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Tritium permeation barrier has been investigated using ceramic coatings for nearly a half century to ensure fuel efficiency and radiological safety in fusion reactor blankets. Our recent studies elucidated irradiation and corrosion effects on microstructure of the coating, electrical conductivity, and hydrogen isotope permeation toward the practical application to the blankets. In a liquid lithium-lead blanket concept, recoil atoms of lithium flowing in the blanket channels and iron from the structural materials by collision with high-energy neutrons will be implanted in the coating, resulting in change in the coating properties. However, lithium implantation effects on the properties of the ceramic coatings have not reported yet. Therefore, this study focuses on electrical conductivity for the ceramic coatings after heavy-ion irradiation and lithium implantation.

Yttrium oxide and zirconium oxide coatings with approximately 700 nm in thickness were fabricated on reduced activation ferritic/martensitic steel F82H plates by magnetron sputtering and metal organic decomposition, respectively. After deposition of 4-mm^2 platinum electrodes, 250-keV lithium ions were implanted into the coatings with an averaged concentration of 1200–12000 appm. Some of the coatings were damaged by 6-MeV nickel ions with a damage concentration of 1 dpa before lithium implantation. Electrical conductivity before and after the irradiation and implantation were measured at room temperature by electrical impedance spectroscopy.

The as-coated samples showed the electrical conductivities of 10^{-8}–10^{-6} S m^{-1} for the yttrium oxide coating and 10^{-9}–10^{-7} S m^{-1} for the zirconium oxide coating. After lithium implantation, the samples showed an increase in the conductivity by 1–2 orders of magnitude and no dependence on lithium concentration. On the other hand, the lithium-implanted samples after heavy-ion irradiation showed similar conductivities of 10^{-4}–10^{-2} S m^{-1}, indicating that the conductivity reached a saturation. In the presentation, the conductivities at elevated temperatures and other coating properties will be discussed.

Keywords

Coating, Irradiation, Implantation, Electrical conductivity.

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P5A4 ABSTRACT-5676
B. Blanket Technology

A Liquid Metal Blanket Concept with the First Wall Covered by the Liquid Metal Surface Flow for the Helical Fusion Reactor HESTIA

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In a fusion reactor, the fuel particles, including tritium, head to the divertor. On the other hand, a nonnegligible part of the fuel particles hits the FW (First-Wall) after the charge-exchange reaction at the plasma edge region. To keep the tritium economy, controlling the tritium inventories on both divertor and FW is necessary. In the case of a helical fusion reactor HESTIA being developed by Helical Fusion Co., Ltd., this issue is solved by covering both the divertor and FW with LMSF (Liquid Metal Surface Flow). HESTIA is a small helical fusion reactor designed based on public knowledge accumulated in LHD. The major radius of the helical coil is 7.8 m and the magnetic field strength at the plasma center is ~7 T. The net electricity of 100 MW will be generated from the fusion output of 340 MW. HESTIA is characterized by a modular-type liquid metal blanket. A multi-metal including Li, Sn, and Pb is being developed as a functional liquid metal. The blanket modules are equipped with the first wall covered with the LMSF. The liquid metal inside the blanket cartridges leaks from the porous material on the FW to form the LMSF. The helical divertor region on each blanket cartridge is also covered by the LMSF. The heat flux on the helical divertor is expected to be ~100 MW per square meter. This will be manageable with the LMSF, although it depends on the flow velocity of the LMSF. The LMSF finally drops to the pool inside the vacuum vessel, together with the liquid metal exhausted from the blanket cartridges. Tritium atoms included in the liquid metal pool are extracted and refueled to the plasma.

Keywords

Heliotron, LHD, liquid metal, divertor, first wall.

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P5A5 ABSTRACT-30da
B. Blanket Technology

Design and construction of a helium cooling experimental loop for tritium blanket

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In Chinese Fusion Engineering Test Reactor (CFETR) project, helium cooled blanket is an important candidate tritium blanket concept. Helium cooling circuit is the cooling system of helium cooling blanket. In order to improve the heat carrying capacity and thermoelectric efficiency of helium and reduce the energy consumption of the circulator, the working pressure of helium is set to 12 MPa and the maximum temperature is set to 550 °C. In order to verify the construction technology of the cooling system and provide testing conditions for the thermal experiment of the blanket components, a high temperature and high pressure helium experimental loop was designed and constructed. A helium circulator with electromagnetic bearing and two compact Printed Circuit Heat Exchangers (PCHE) are used in the loop. The ' 8' shaped design process similar to Test Blanket Module (TBM) helium cooling loop is adopted, which reduces the requirement of the working temperature of the circulator, reduces the preheating power, and reduces the investment too. In the later stage, it will be connected to the existing electron beam device to expand the thermal experimental ability. In commissioning, the parameters of the experimental section of the loop reached flow rate of 2.5 kg / s, temperature of 550 °C, and pressure of 12 MPa.

Keywords

CFETR, Tritium blanket, Helium loop.

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P5B1 ABSTRACT-3587
H. Models and Experiments for FNT

Fusion Science and Technology Studies on the Basic Plasma Science Facility

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The Basic Plasma Science Facility (BaPSF) at UCLA is a US national collaborative research facility for studies of fundamental processes in magnetized plasmas sponsored by US DOE and US NSF. BaPSF is utilized for a number of fusion-relevant studies including ion-cyclotron range of frequencies (ICRF) wave excitation and plasma-materials interaction. The ICRF Campaign on LAPD focuses on a number of fundamental issues related to ICRF wave excitation for heating and current-drive in fusion plasmas. A high-power (~150 kW) tube-based RF amplifier is available, along with three antennas: a single strap, insertable/rotatable fast wave antenna, a large 4-strap fast wave antenna, and a 10-module helicon antenna (low power antenna previously used on DIII-D). During fast wave excitation using the high-power RF source and the single-strap antenna, evidence of rectified RF sheaths is observed along with copper deposition on plasma facing components due to sputtering at the antenna. Significant reduction or elimination of RF rectified sheaths and associated sputtering is observed by replacing the original copper sidewalls with insulators (e.g. MACOR). Low-power (~100 W) helicon wave experiments were conducted to study wave propagation and coupling properties with a 10-element comb-line traveling wave antenna, verifying coupling to both the fast and slow wave branches and control of directionality and parallel wavenumber. Plasma-Material Interactions (PMI) studies have recently been conducted as part of the DOE/ARPA-E funded Advanced Materials for Plasma-Exposed Robust Electrodes (AMPERE) project. The AMPERE team designed and tested new volumetrically complex (VCM) tungsten targets that were compared to flat tungsten for helium plasma exposure at target bias voltages up to 600 V. The VCMs significantly suppressed arcing likelihood and allowed operation over 200 V higher than flat targets, while high-speed videography and spectroscopy of tungsten lines showed a reduction in arc-induced sputterants for the VCM targets.

Keywords

Basic Plasma Science Facility, RF Heating and Current drive, Plasma Materials Interaction.

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P5B2 ABSTRACT-4eaa
H. Models and Experiments for FNT

IFMIF/EVEDA Achievements Overview

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The plasma of the future fusion power plants based on a deuterium-tritium reaction will generate high energy neutron flux interacting with the materials of the fusion plasma facing components. In order to study the irradiated behavior of these materials, the International Fusion Materials Irradiation Facility (IFMIF) was conceived to generate these fusion relevant neutrons at 14 MeV through Li(d,xn) nuclear reactions. Indeed, IFMIF will enable high intensity neutrons to reach 20 dpa/fpy (full power year) in a volume around 500 cm^3 in its high flux test module. IFMIF is presently in its Engineering Validation and Engineering Design Activities (EVEDA) phase under the Broader Approach agreement signed between EURATOM and Japanese Government in 2007. This agreement mandates to validate the design of the different systems of IFMIF and to produce an integrated engineering design of IFMIF, together with the data necessary for future decisions on the construction and operation of the plant. While the Engineering Validation Activity (EVA) of the Lithium Target Facility and the Test Facility was completed by constructing prototypes, the EVA of the Accelerator Prototype Facility with the Linear IFMIF Prototype Accelerator (LIPAc) is still on-going at the Rokkasho Fusion Institute, Japan. This paper overviews the achievements in the previous phase ended on 31 March 2020 and the progress in the current phase to complete the commissioning of the LIPAc and to enhance the sub-systems in order to prepare for the design and operation of the IFIMIF-like future Fusion Neutron Source facilities like A-FNS and DONES.

Keywords

IFMIF-EVEDA, LIPAc, Fusion Neutron Source, Broader Approach.

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P5B3 ABSTRACT-4d4a
H. Models and Experiments for FNT

Post-irradiation analysis of ITER materials following JET DTE2

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The record-breaking fusion results obtained at the Joint European Torus (JET) during deuterium-tritium experimental campaign in 2021, and the following tritium plasma campaign in 2022 produced approximately 1E21 neutrons. As part of EUROfusion JET3 and PrIO (Preparation of ITER Operation) projects – aimed at using the JET experiments to improve knowledge on neutronics, nuclear technology and safety issues, to develop and validate nuclear codes, neutronics tools and experimental techniques – the results of post-irradiation analysis of ITER materials installed into the JET nuclear environment during these experimental campaigns are presented. Recently, a total of 68 ITER material foils and 13 dosimetry foils were irradiated in a long-term irradiation station assembly close to the JET vacuum vessel. The ITER materials were samples from those used in the manufacturing of the main tokamak components and include SS316L steels from various manufacturers, SS304B, Alloy 660, W, CuCrZr, XM-19, Al bronze, and EUROFER among others. This paper details the recent results from experimental analysis activities, and future plans within this project that are associated with the forthcoming JET DTE3 campaign. Initial observations of the experimental results that have been derived are made along with comparisons against nuclear simulations using detailed models, radiation transport, inventory codes and fusion-relevant nuclear data libraries.

Keywords

Neutronics, activation, JET, ITER.

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P5B4 ABSTRACT-8266
H. Models and Experiments for FNT

TRL analysis of IFMIF-DONES and Overview of the required validation needs

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IFMIF-DONES is a powerful neutron source, generated by the interaction of a high current (125 mA) deuteron beam accelerated to 40 MeV and a liquid lithium target, which is being designed to qualify DEMO structural materials.

This work presents an analysis of the IFMIF-DONES design maturity and the necessary activities to increase it up to the level required to launch the procurement phase.

The present maturity of the IFMIF-DONES subsystems and components has been analyzed using a Technology Readiness Level (TRL) methodology. The TRL scale and definitions employed in EU Horizon programs have been found to be appropriate for this assessment, with some modifications to consider the IFMIF-DONES peculiarities. The most recent information about the design and R&D performed has been used for this analysis.

The level of Technology Readiness needed for launching the procurement of each subsystem or component ("target TRL") has been established. New technologies developed for DONES shall in general reach TRL7 (full scale prototype tested under the foreseen operational conditions) although in some specific cases, tests at engineering scales or even laboratory tests (TRL6-TRL5) are considered sufficient. It is recognized that for the components located close or into the irradiation cell, full validation under operational radiation conditions will not be possible. In addition, some requirements regarding RAMI or beam-target interaction cannot be experimentally tested before the facility commissioning phase.

From the comparison between the present TRL and the target TRL, the elements requiring further development and qualification have been identified. The experimental activities needed to increase their maturity have been defined. Other validation activities, motivated by issues related to project transversal activities such as neutronics, safety or control have also been included. A summary of the most relevant prototyping, testing and qualification needs identified will be presented.

Keywords

IFMIF-DONES, neutron source, technology validation, technology readiness level.

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P5B5 ABSTRACT - df9c
H. Models and Experiments for FNT

Fusion neutronics experiments utilizing the intense DT neutron generator of Technical University of Dresden

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In the early 2000s the Technical University of Dresden has set up a new fusion neutron laboratory with an accelerator-based intense DT neutron generator, TUD-NG. It provides a typical neutron yield at DT (deuterium-tritium) energies up to several 1011 s-1 and at DD (deuterium-deuterium) energies up to several 109 s-1. Operation can be in continuous and pulsed mode, the machine control provides microsecond and nanosecond pulsing. Accelerated deuterons with energies up to 325 keV bombard water-cooled solid targets containing tritium or deuterium. The maximum beam current achieved so far is approximately 8 mA at the target, the laboratory is licensed for a maximum DT neutron yield of 1012 s-1. The setup of experiments is very flexible, and irradiations of materials and small samples at controlled temperatures up to several hundred degrees Celsius are possible.
The laboratory has been involved in the European fusion research since its commissioning through close collaboration with Karlsruhe Institute of Technology. In the past, measurements were done on mock-ups of the European Helium-Cooled Lithium-Lead Test Blanket Module (TBM) and the Helium-Cooled Pebble-Bed TBM for ITER as well as activation experiments on fusion reactor relevant materials to validate evaluated nuclear data files.
Starting in 2010, focus shifted to R&D of nuclear instrumentation. Extensive work was done with novel self-powered neutron detector designs, radiation detectors based on silicon carbide diodes at high temperatures, and a neutron activation test system for the European ITER TBMs. Currently upgrades of the neutron generator and the measurement capabilities are underway. Two mobile high-purity germanium detectors were added as well as a compact electron spin resonance spectrometer suitable e,g, for dosimetry applications. New activity standards for dosimetry purposes with low uncertainty and traceable to national standards are available as well. Future plans for exploiting the TUD-NG capabilities in fusion neutronics research are presented.

Keywords

Neutron generator, fusion neutronics, breeding blanket, nuclear instrumentation development.

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P5C1 ABSTRACT-fc1b
D. Material Engineering for FNT

Exploit existing Material Test Reactors to support the aggressive timeline for deployment of commercial fusion power

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Structural materials are used to construct the reactor vessel, the divertor, and other critical components of Fusion Power Plants (FPPs), and are exposed to high temperatures, radiation, and high-energy neutrons. They must withstand extreme conditions over long periods of time without degrading or failing to ensure the safety and reliability of FPPs. One major challenge is the impact of high-energy neutrons. Neutron damage can cause degradation and potential failure of materials, as well as the accumulation of hydrogen and helium that can lead to embrittlement. Researchers use various testing methods, including ion and neutron irradiation tests, to address these challenges.

The U.S. Department of Energy recently announced a bold decadal vision for commercial fusion energy, which includes establishing private-public partnerships to develop and demonstrate integrated fusion energy systems that can generate electricity on a commercial scale within 10 years. This aggressive timeline is not compatible with the completion of large neutron source facilities for structural materials qualification. This work assesses the possibility to exploit existing Material Test Reactors (MTRs) to complement long term qualification plans with near term neutron irradiation data to support design efforts.

The Advanced Test Reactor (ATR) at the Idaho National Laboratory is a water-cooled MTR that has been in operation since 1967 and has a long history of capsule and instrumented lead-out irradiation tests. ATR currently utilizes a cadmium-lined basket to filter thermal neutrons for higher energies to test Sodium Fast Reactor (SFR) fuel pins, reaching SFR prototypic heating rates of about 3 at% burnup per year in fuel and about 5 dpa per year in cladding. This work presents a preliminary analysis supporting the design of the Boosted Energy Advanced Spectrum Test (BEAST), which leverages the current cadmium basket test rig to achieve conditions relevant to fusion structural materials.

Keywords

Fusion, structural materials, material test reactors.

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P5C2 ABSTRACT-817e
D. Material Engineering for FNT

Selective adsorption properties of layered titanate for tritiated water

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Inorganic materials, such as concrete used in many fusion reactor facilities, adsorb tritiated water, which makes decontamination difficult and causes the generation of a large amount of nonflammable radioactive waste. Water strongly adsorbed by surface with inorganic materials causes tritiated water to be adsorbed more strongly than light water as an isotope effect, resulting in the accumulation of tritium within inorganic materials. In response to this well-known phenomenon, we found that the composition controlled titanate, an inorganic material, can selectively inhibit the adsorption of tritiated water onto the material, a characteristic that is the exact opposite of common knowledge. Focusing on "titanate," a general-purpose inorganic material that can be given various characteristics by controlling its shape and composition, we examined adsorption properties of tritiated water on the material experimentally by the method of indirectly contacting inorganic materials with tritiated water as vapor in an airtight container. Molecular sieve as an inorganic material that selectively adsorbs tritium concentrates tritiated water within itself, resulting in a lower concentration of tritiated water in the container. In contrast, it was found that titanate with composition control showed the exact opposite tendency of tritiated water concentration. This result is considered as a selective inhibition of tritiated water adsorption on the material. It was shown that it is possible to control the composition of titanate to selectively inhibit the adsorption of tritiated water onto materials based on a new concept.

Keywords

Tritium, tritiated water, adsorption, titanate, isotopes separation.

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P5C3 ABSTRACT-d0de
D. Material Engineering for FNT

Towards the down-selection of ceramic materials for the European high temperature DCLL BB concept based on Single Module Segments (SMS)

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The previous architecture of a DCLL breeding blanket based on multi-module segments has required the development of insulating flow channel inserts included in the PbLi flowing channels. These sandwich-like, steel-ceramic-steel layered structures were incorporated in the design to electrically decouple the liquid metal with the magnetic field, minimizing the flow pressure drop. However, identified issues point out the design towards a simpler and more efficient model, the single module segment (SMS). Here, the bare ceramic channels contain the flowing liquid metal, reinforced by an EUROFER steel envelope. The SMS concept of insulating ceramic channels allows the temperature to increase over the nominal 550ºC, enhancing the plant net efficiency but relying on materials of limited knowledge under fusion environmental conditions. The CIEMAT-LNF is leading the development of the high-temperature design, emphasizing on the down- selection of ceramic insulators compatible with the high demanding application. Therefore, the experimental evaluation of the dielectric and mechanical behavior of selected ceramics in flowing PbLi and their neutron radiation assessment are the key issues initially addressed. To simulate the neutrons effect, ion beams are accelerated towards selected ceramic targets, allowing the implantation of light species and the induce of structural damage. Besides, the numerical evaluation of the electrical conductivity threshold and the candidates' capability to withstand characteristics loads of the breeding circuit walls (e.g. weight of PbLi) are presented in this work. According to neutronics, selected ceramic compositions have been analyzed in term of their impact in the Tritium Breeding Ratio and their classification as Low/Intermediate/High Level Waste and as Simple/Complex Remote Maintenance. For such categorizations, the total beta-gamma activity, specific activity for different nuclides, decay heat and surface gamma dose rate have been evaluated with reference to the IAEA and SEAFP-2 standards, and to the specific regulations of the near-surface repository El Cabril (Spain).

Keywords
DCLL SMS, insulating ceramic, PbLi channels.

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P5C4 ABSTRACT-1fb2
D. Material Engineering for FNT

Effect of recrystallization on fatigue crack growth characteristics of a pure tungsten

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Tungsten (W) is used for plasma-facing components (PFCs) such as ITER divertor owing to its desirable physical and plasma properties. However, when exposed to high heat flux (HHF), W armor of divertor target components would experience recrystallization. Recrystallization can result in the formation of fatigue cracks on the recrystallized surface of PFCs. Repetitive thermal fatigue can cause plastic deformation accumulation or crack initiation, leading to functional failure when the crack approaches the Cu cooling tube. High-temperature fatigue evaluation data are necessary for structural integrity in a fusion environment. However, fatigue crack growth data for W materials are limited due to inherent brittleness and difficulty in introducing a sharp pre-crack needed for fatigue crack growth testing. In this work, fatigue crack growth rate (FCGR) tests for a ITER-grade W were performed, focusing on the effect of recrystallization. We conducted FCRG tests at 500 to 700°C for as-received and recrystallized W (3 h at 1300°C) using a small-sized disk-type compact tension (DCT) specimen with a sharp pre-crack. To overcome inherent brittleness and poor oxidation resistance, the processes of pre-crack and fatigue test have been conducted in Ar gas environment. The relevant deformation mechanism depending on test temperature and recrystallization were discussed in view of fracture resistance and microstructure.

Keywords

Tungsten, Recrystallization, Fatigue crack growth, Pre-crack.

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P5C5 ABSTRACT – 54db
D. Material Engineering for FNT

Additive Manufacturing of copper alloys and refractory metals for the DTT Project

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Nuclear fusion has many challenges, one of which is the production of reactors components. Due to the extreme operating requirements, the geometrical aspect is of great importance.

Some important components are, for example, the accelerating grids of the Injection system. In order to guarantee the most appropriate beam optics and optimized cooling, the grids must be characterized by a shape that would be impossible to obtain with traditional manufacturing approaches.

Additive manufacturing technologies allow producing very geometrically-complex components, with high dimensional accuracy and an optimized shape considering also the design-for-assembly point of view. However, these innovative technologies are not free from challenges. A key point is the optimization of the process parameters, that are specific for each material and powder, this preliminary research is fundamental to obtain the best quality and performance from a material.

Additive manufacturing represents a highly promising production method also for the fabrication of components made of refractory metals. Indeed, this special class of materials is extremely difficult to form due to their unique characteristics; moreover, they are usually very expensive, so wastage should be reduced as much as possible. In nuclear fusion, refractory metals are good candidates for plasma-facing and divertor applications, for example, where the operative conditions are prohibitive.

In this work, the studies related to the characterization of materials processed with LPBF technique are presented, in particular for what concerns copper alloys and refractory metals. The latest updates on the innovative design specially developed for additive manufacturing of the accelerating grids for the NBI system of DTT are also described.

Keywords

Additive manufacturing technologies allow producing very geometrically-complex components, with high dimensional accuracy and an optimized shape considering also the design-for-assembly point of view.

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P5D1 ABSTRACT-2c80
G. Safety Issues and Waste Management

Fusion waste requirements for tritium control: perspectives and current Research

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The successful realisation of energy production through the fusion of deuterium and tritium will necessarily lead to the generation of waste contaminated with tritium. Not only will some of the tritium fuel permeate into components of fusion reactors and their wider fuel cycle, but tritium will also be generated directly in the materials exposed to the high neutron fluxes via nuclear break-up reactions. Current estimates predict that this second, inherent source of tritium will be sufficient on its own to pose a significant hazard and decommissioning challenge, requiring long-term, tritium-compatible storage solutions and improved detritiation processes. UKAEA is exploring different solutions for detritiation, including acid etching and thermal treatments, leveraging the experience gained from handling the waste materials generated during the operation of JET. The high mobility of tritium poses a challenge for waste storage and containment. Knowledge of how tritium permeates through different materials under different conditions is critical for ensuring the environmental safety and regulatory compliance in the design and operation of waste-disposal containers, waste storage units and waste transportation systems. UKAEA, supported by Horizon 2020 and the STEP Programme, designed a lowtemperature permeation rig to measure the permeation of tritium at low temperatures through different materials; primarily investigating candidate materials for waste boxes, but which could also be deployed for tritium containment in low temperature regions of a fusion power plant, such as on the walls of a reactor hall. The experience of establishing and operating experimental tritium rigs and handling tritiated waste from JET is invaluable for the future fusion industry, while there is optimism that the experimental research trials will steer fusion waste strategy by designing waste containers and barriers that minimise the spread of tritium contamination or provide efficient detritiation solutions.

Keywords

Tritium permeation, waste processing, decommissioning.

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P5D2 ABSTRACT-5418
G. Safety Issues and Waste Management

Overview of French R&D studies for the development of tritium confining packages

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CEA

During its operating and dismantling phases, the ITER facility will generate different types of tritiated radiological waste for which a waste management scenario implementing best available techniques is mandatory.

Depending on national waste management strategy and waste acceptance criteria of final repositories, the management of this tritiated waste combines a cooling phase, characterization, waste cutting, sorting, treatments, conditioning steps and storage (with potential interim storage for tritium decay prior transfer to a final repository) or possible reuse for waste with an activity below a clearance level.

One of the major objectives is to guarantee a safe management strategy with a limited environmental impact in particular regarding the tritium release. Confining packages are thus a key option to contribute to prevent tritium outgassing from the waste to the environment.

This paper presents the strategy supported at CEA for the development of tritium confining packages compliant with French regulation, transportation constraints and final repositories waste acceptance specifications. This strategy relies on multipurpose research and development studies in view to reach a technical readiness level of at least 6 within 4 years.

To prevent tritium release, the package should be enough confining so that tritium diffusion characteristic time is higher than the natural tritium decay one. For that, the confining package design relies on an optimization of package design (size, material and thickness), an efficient sealing joint, a confining cement matrix and the use of a tritium trapping metal oxide mixture. Each of the key elements is assessed and developed under specific R&D program and an integrated modelling tool is under development to simulate the overall behavior of tritium in such confining package. This paper will give an overview of the identified issues, the different milestones already achieved in each field (modeling, cement matrix…) and the ones foreseen with their associated schedules.

Keywords

Tritiated waste, confining package, waste management.

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P5D3 ABSTRACT-71da
G. Safety Issues and Waste Management

Numerical Simulation of Tritium Behavior under Transient Conditions for CFETR

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China Fusion Engineering Test Reactor (CFETR) is an important milestone project in the development of nuclear fusion in China and the world, and it is a new generation of large-scale fusion engineering experimental reactor. Tritium is recognized as one of the key nuclear safety issues, and must be handled under multiple barriers served by detritiation system. This concept has been successfully adopted in tritium facilities of the world and it is important to ensure that the design of the containment system is appropriate. The tritium plant for CFETR is under detailed design, and it is of great important to simulate 3D tritium transport behavior in CFETR tritium safety containment system. As the member responsible for tritium transport behavior study in the CFETR tritium safety confinement system of the National Key R&D Program of China launched in 2017, our tritium technology team are carrying out numerical simulation and analysis for CFETR tritium plant safety. In this paper, Tokamak Exhaust Processing (TEP) system of CFETR is selected as the typical system and then the tritium migration and removal behavior study under postulated incident and accident conditions is described. Quantitative results of the transport behavior of tritium in the 2nd confinement system (glove box) and 3rd confinement system (process room) during the whole sequences (including tritium release, alarm, isolation, tritium removal, etc.) have been introduced. The results could support the detailed design and engineering demonstration related research of CFETR tritium plant.

Keywords

China Fusion Engineering Test Reactor (CFETR), Tokamak Exhaust Processing (TEP) system, Numerical simulation, Tritium transport behavior, Incident and accident conditions.

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P5D4 ABSTRACT.- b6d8
G. Safety Issues and Waste Management

Parametric assessment of the Activated Corrosion Products on the ITER Water Cooled Lithium Lead Test Blanket System

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Activated Corrosion Products (ACP) represent a significant radiological source term for occupational dose assessments in high-performance fusion facilities. ACP are produced and transported in the water-cooling loop via a combination of multiple complex interacting phenomena: i.e., corrosion-release, dissolution, precipitation, erosion, deposition, advection, purification, activation, and radioactive decay. The quantification of the ACP requires a multi-physics approach taking into account the design of the system, operational scenario, thermal-hydraulic conditions, water chemistry, neutron irradiation, and materials properties.

The present work aims at presenting a sensitivity study on some relevant design parameters subject to change in the generation of ACP, which contribute to the dose rate fields, and thus to the radiation exposure of workers, in the Water Cooling Systems (WCS) and Coolant Purification System (CPS) in the ITER Water Cooled Lithium Lead (WCLL) Test Blanket System (TBS).

To provide more realistic estimations, the latest design information and data of the WCLL TBS WCS and CPS are implemented in the OSCAR-Fusion v1.4 simulation code. Such a code is based on the physical models made available by CEA Cadarache and validated so far mostly under operating conditions relevant to pressurized water reactors. Locations of potential hot spots based on the reference data available on the design and working conditions of the system are provided, to feed potential radiological design improvements. The analysis then concentrates on evaluating the impact of the variations on surface roughness, initial oxide thickness, hydrogen content, pH, and corrosion rate on the ACP inventory, which in turn results in dose field modifications. Appropriate discussion on the numerical sover stability over the range of applications in fusion neutronics is provided.

Keywords

ACP, ITER, WCLL, WCS, CPS, TBS, OSCAR, blanket.

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P5D5 ABSTRACT – e3eb
G. Safety Issues and Waste Management

CHICADE nuclear facility : a collaborative technological platform, dedicated to the expertise and characterisation of nuclear wastes

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CHICADE is one of the nuclear facilities of Energy Division of French Alternative Energies and Atomic Energy Commission. CHICADE is part of the Directorate for Nuclear Dismantling, Services and Waste Management.

CHICADE brings together both skills and means of characterization, using destructive methods on waste packages. It also carries out measurements on the whole waste package (gas release measurements, leaching tests)

To carry out these assessments, CHICADE is a large Nuclear Facility with more than 6,000 m^2 including 4,000 m^2 of hot surfaces including a wide range of equipment from the bench, to the fume hood, Glove Boxes, shielded cell and measurement casemates allowing the expertise of wastes from few mg to several tons!

The fifty or so employees of CHICADE contribute to the activities of the 6 laboratories, each with its own field of expertise: operation, characterisation, radiochemistry, imaging, gamma spectrometry and, beyond the activity on nuclear waste, the manufacture of fission chambers.

These skills, both technical and human, give CHICADE the opportunity to carry out research and technological development activities making possible supervision of research work of PhD students, teaching activities and meeting the needs of other waste producers such as ITER, ONDRAF, Orano, EdF, etc.

The objectives of the assessments carried out at CHICADE are therefore:

1. Expertise in order to transport wastes to the appropriate outlet,
2. Development of waste conditioning methods,
3. Sample controls of waste assigned to existing outlets.

In conclusion, CHICADE is a nuclear facility with unique equipment, allowing complete expertise to be carried out in a single location, for greater efficiency, benefiting from crossed, complementary and innovative methods.

This presentation will review the characterizations carried out on nuclear wastes.

Keywords

Nuclear waste – characterization.

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P6A1 ABSTRACT – 7892
A. Plasma-Facing High Heat Flux Components

Lithium Technology Readiness as a Fusion Energy Plasma-Facing Component

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The low-recycling regime for tokamak and stellarator operation offers an affordable path to fusion. Dramatically reducing the recycling leads to higher edge temperatures, higher confinement time, better plasma stability, and smaller machines to generate the same amount of fusion energy. The best way to reduce recycling dramatically is by having a lithium plasma-facing surface. To ensure that the lithium surface does not saturate, the lithium needs to flow across the strikepoint and exit the machine where it can be de-saturated quickly and returned. The flowing lithium surface also needs to be able to handle a high heat flux and not produce droplets or other debris.

We work with both industrial and government partners to study, develop, and test liquid metals and the associated technology. Previous work has developed the Liquid Metal Infused Trenches (LiMIT) PFC design, and ongoing work is aimed at integrating this free-surface PFC with a liquid lithium loop system. This system has the ability to expose the PFC to a deuterium plasma, move the lithium out of the magnetic field, and remove the entrained deuterium in real-time. Many safety and liquid control systems have been developed to allow for continuous safe operation of the pumped loop. A suite of diagnostics is being used to track the hydrogen inside the system.

Along with PFC development, studies into the control, stability, corrosion, and fabrication of liquid metal alloys is underway. Vacuum fabrication systems are developed to create liquid metal alloys, like SnLi and PbLi, and test basic properties, such as chemical composition and vapor pressure. Both static and dynamic liquid metal corrosion experiments are performed at CPMI regularly to continue to test novel materials in liquid metal environments. Wetting tests and associated surface modifications and coatings allow for enhanced control over liquid metal flows on PFC surfaces.

The low-recycling regime for tokamak and stellarator operation offers an affordable path to fusion. Dramatically reducing the recycling leads to higher edge temperatures, higher confinement time, better plasma stability, and smaller machines to generate the same amount of fusion energy. The best way to reduce recycling dramatically is by having a lithium plasma-facing surface. To ensure that the lithium surface does not saturate, the lithium needs to flow across the strikepoint and exit the machine where it can be de-saturated quickly and returned. The flowing lithium surface also needs to be able to handle a high heat flux and not produce droplets or other debris.

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Along with PFC development, studies into the control, stability, corrosion, and fabrication of liquid metal alloys is underway. Vacuum fabrication systems are developed to create liquid metal alloys, like SnLi and PbLi, and test basic properties, such as chemical composition and vapor pressure. Both static and dynamic liquid metal corrosion experiments are performed at CPMI regularly to continue to test novel materials in liquid metal environments. Wetting tests and associated surface modifications and coatings allow for enhanced control over liquid metal flows on PFC surfaces.

Keywords

Lithium, Recycling, Divertor, Corrosion.

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P6A2 ABSTRACT – 5528
A. Plasma-Facing High Heat Flux Components

Development of a liquid metal divertor solution for DEMO

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[9]University of Latvia
[10]UKAEA-CCFE

For fusion power plants plasma facing component (PFC) resilience and longevity under extremely high particle and neutron fluences is of increased significance compared to today's devices due to the desired power output and plant availability. Liquid-metal (LM) based divertors offer significant advantages for such a scenario, such as a self-healing surface, smaller impact from neutron loading and inherent vapour shielding response during off-normal loading events. A Sn-based divertor for the EU-DEMO project is therefore being developed. Two pre-conceptual designs have been produced and published which conform to the design criteria and constraints for incorporation into DEMO and are predicted to handle at least equal steady-state heat loads as the baseline W-armour design. Recent experiments in ASDEX-Upgrade suggest that further improvements of the design are necessary to prevent uncontrolled spread of the liquid Sn from the CPS. A number of different Capillary Porous Systems (CPSs) to confine the LM are under development and are tested in the Magnum-PSI, OLMAT and GLADIS test-beds. At the same time core-edge integrated modelling has been carried out using COREDIV with detailed boundary modelling using the TECXY and SOLPS-ITER SOL codes in order to evaluate the steady-state performance, identifying solutions with additional impurity seeding which lead to both acceptable target heat loads and core Sn concentrations. TOKES modelling and testing in the Kh-50 QSPA device meanwhile indicate the protective effect of vapour shielding by tin during high intensity transient loading such as disruptions, implying a more-resilient PFC under off-normal conditions. Designs for larger-scale mock-ups and plasma-facing units for testing in high- heat and particle flux devices and for the COMPASS-U device are under way as a stepping stone on the path to LM deployment in DEMO. An overview of recent activities and results and an outlook on the future of this activity will be presented.

Keywords

Liquid metals, divertor, tin, divertor design, SOL modelling, core-edge modelling, high heat flux devices.

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Estimation of the Thermal-Fluid and Thermal-Structural Performance of Helium-Cooled Modular Finger-Type Divertors

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Over the past decade, our group at the Georgia Institute of Technology (GT) has investigated the thermal-fluid performance of the helium-cooled modular divertor with multiple jets (HEMJ) originally proposed for the EU DEMO and a simplified “flat” design variant of the HEMJ for long-pulse magnetic fusion energy reactors. As part of this effort, experimental studies were carried out on a single “finger” of these designs in a helium (He) loop at the prototypical design pressure of 10 MPa, inlet temperatures as great as 420 °C, and He mass flow rates as great as 10 g/s using test sections made from stainless steel and tungsten (W) alloys that did not include the hexagonal plasma-facing tile.

Experimental measurements of cooled surface temperatures were used to develop correlations for average Nusselt numbers and pressure loss coefficients, and to validate numerical predictions of average Nusselt numbers from a corresponding computational fluid dynamics (CFD) model. This validated CFD model was then used to analyze the performance of these finger-type divertors at prototypical conditions, and the impact of using non-prototypical coolants as well as inner cartridge and outer shell materials. We have previously presented thermal-fluids performance curves that estimate the maximum steady-state heat flux that can be accommodated by the plasma-facing W tile and coolant pumping power requirements at prototypical operating conditions based upon material temperature limits.

This work will present significantly updated thermal-fluids performance curves based upon experimental studies using fully prototypical materials, and introduce performance curves developed from ITER thermal structural failure protection criteria, including protection against ductile and non-ductile failure, ratcheting fatigue, and creep fatigue. The performance curves for these finger-type divertors demonstrate that the “flat” design, with a significantly less complicated geometry than the HEMJ, has a thermal-fluid and thermal-structural performance comparable to the original HEMJ concept.

Keywords

Thermal-fluids, divertors.

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P6A4 ABSTRACT-84ac
A. Plasma-Facing High Heat Flux Components

The Integrated Engineering Design Concept of the Upper Limiter within the EU-DEMO LIMITER System

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The EU-DEMO first wall protection relies on a system of limiters. Although they are primarily designed for facing the energy released by a limited plasma during transients, their design should safely withstand a combination of loads relevant for in-vessel components (IVCs) during steady-state. They are not meant to breed tritium, nor provide plasma stability. However, sitting in place of blanket portions, they should ensure shielding function to vacuum vessel and magnets while withstanding both their dead weight and the electro-mechanical loads arising from the interaction between current induced in the conductive structure and magnetic field. During plasma disruptions they will be subjected to halo currents flowing from/to the plasma and the grounded structures, whose effects must be considered. Disruption-induced electro-mechanical loads are IVC design-driving, despite the uncertainties in both eddy and halo currents' magnitude and distribution, which depend on IVC design, electrical connectivity, plasma temperature and halo width.

The integrated design of the limiter is made of two actively water-cooled sub-components: the Plasma-Facing Wall (PFW) directly exposed to the plasma, and the Shield Block (SB) devoted to hold the PFW while providing neutronic shielding. The PFW design is driven by disruptive heat loads. Disruption-induced electro-magnetic loads are instead SB design-driving, meaning that the design details (i.e. geometry, electrical connections, attachments) affect the loads acting on it, which, in turn, are affected by the mechanical response of the structure.

This paper describes the design workflow and assessment of the Upper Limiter (UL), resulting from a close and iterative synergy among different fields. Built on static-structural and energy balance hand calculations based on, respectively, preliminary electro-magnetic and neutronic loads, the UL integrated design performance has then been verified against electro-magnetic, neutronic, thermal-hydraulic and structural assessment under the above-mentioned load combination. The outcome will be taken as reference for future limiter engineering designs.

Keywords

EU-DEMO Upper Limiter, Integrated design, EM, Neutronics, Thermal-hydraulics, Structural integrity.

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P6A5 ABSTRACT-4785
A. Plasma-Facing High Heat Flux Components

Innovative concepts of COMPASS-U tokamak divertor surviving plasma pulses

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A remaining challenge in thermonuclear fusion is survival of the heat shield against destructive plasma heat shocks, especially the edge localised modes (ELMs) and vertical displacement events. The EU DEMO path aims for new plasma modes with significant ELMs suppression and disruption mitigations. Since, however, it's still under investigation and limiting the operational scenarios (e.g. max. Ip), they might yield less fusion power than the ITER standard (Type-I) ELMy H-mode. Therefore, we keep studying two divertor target improvements.

We demonstrated survival of liquid metal divertor (LMD) under ITER-relevant divertor surface heat loads on COMPASS tokamak, interpreted by HeatLMD simulation [https://doi.org/10.1016/j.nme.2020.100860] and used for its predicted behaviour on COMPASS Upgrade full-toroidal LMD [https://doi.org/10.1088/1402-4896/ac1dc9] where damage of the best water-cooled tungsten divertor is expected. CoreDiv simulations quantify that the eroded Li Sn might, however, cause significant central plasma cooling.

Therefore we invented an additional concept of divertor sweeping of the plasma contact strike point fast and far enough in order to spread the ELM heat pulse. For EU DEMO we quantified [https://doi.org/10.1038/s41598-022-18748-x] the feasibility of a dedicated copper coil, including the induced currents and power electronics, yielding suppression of the ELM-induced heat shield surface temperature rise by an interesting factor of 3-5. Multiplied by other mitigation concepts (I, QCE, X-point radiation), ELM-like pulses might be mitigated enough.

First proof-of-concept experiment on the COMPASS-U tokamak is here presented, expecting to reach $q_{||}$=¼ GW/m2 on the divertor strike point with ELMs $\varepsilon_{||}$=⅓ MJ/m2. Even though it's much lower than the expected $\varepsilon_{||}$~15 MJ/m2 on ITER, infrared image of the ELM swept strike point should verify its mitigation effect, especially whether plasma indeed follows the kHz-fast sweeping strike point position. We'll present heat conduction simulation, 3D magnetic field tracking and the power electronics.

Keywords

Liquid metals, fast divertor sweeping, ELMs, disruptions.

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P6B1 ABSTRACT – e157
C. Fuel Cycle and Tritium Processing

Tritium Transport Modelling: current status, open points and perspectives

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The development of the capability to predict tritium retention and permeation through accurate and reliable tritium transport models is one of the most relevant objectives of the European Test Blanket Module (TBM) Programme, constituting a key aspect also for the development and licensing of next-future fusion systems and power plants, including the European DEMO. Tritium analyses can be performed using either system-level models to calculate global blanket and ancillary system performances or detailed 3D models for spatially limited areas and critical regions. System-level models consider at most 1D mass balance equations, allowing to take into account several physical phenomena such as diffusion, surface recombination and dissociation, chemical reactions, trapping, etc. An example of these codes is EcosimPro, developed in collaboration with CIEMAT (Spain) and Empresarios Agrupados (Spain), which is able to predict the transport of tritium through different components and materials of the two European Test Blanket Systems to be installed in the ITER machine. A limitation of these codes is the difficulty in describing complex geometries or complex effects, such as the ones generated by magneto-hydrodynamics (MHD), the surface effects or temperature field variable with the time. Several finite volume and finite element models have been developed so far for the simulation of tritium transport in complex geometries and physical conditions, which also a include customization capabilities. The aim of this paper is to provide an overview of the status of the art in the tritium transport modelling, focusing on three aspects: first, to highlight the status of system-level and detailed-level codes; second, to propose a material database to be used as a reference for tritium transport simulations; last but not least, to propose a gap analysis in order to contribute to the organization of a work plan for the next future.

Keywords

Tritium transport, modelling, material database, ITER, DEMO.

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P6B2 ABSTRACT – 39d4
C. Fuel Cycle and Tritium Processing

Study on the processing of highly tritiated water by the two-stage palladium membrane reactors

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A small amount of highly tritiated water (10^{13} Bq/L~10^{16} Bq/L) will be produced during the tritium-related operation of fusion reactor exhaust gas processing. Highly tritiated water with concentration higher than 10^{13} Bq/L will decompose to oxygen and tritium gas by self-radiolysis, which has explosion risk and needs to be processed. Traditionally, tritiated water is processed by dilution or reduction to tritium gas in hot metal bed, which has the problem of tritium-containing waste expansion. Based on hydrogen-water isotope exchange reaction, a two-stage palladium membrane reactor processing technique is proposed in this paper. First, tritium water vapor after gasification is mixed with carrier gas and then introduced into the catalyst side of the reactor. At the same time, pure hydrogen is introduced into the other side of palladium membrane. Pure hydrogen permeates to the catalyst side and undergoes hydrogen-water isotope exchange reaction with tritium water vapor, and the produced tritium permeates to the pure hydrogen side and leaves the reactor. The tritium experiment was carried out with tritiated water of 6.9×10^9 Bq/L. The results showed that the tritium decontamination factor was 431 ~ 22258 at 6 L/h ~ 24 L/h (gas flow rate), and the tritium concentration of water decreased to 1.6×10^7 Bq/L~3.1×10^5 Bq/L. Under the optimum conditions, tritiated water of 17.9 mCi was continuously processed, and 99.995% of tritium was converted to tritium gas, tritium in the exhaust gas was 3.4 μ Ci (0.018%), and the product tritium recovery ratio was 99.97%. According to different processing occasions, a tritiated water processing device with the character of high integration, flexible interface and easy amplification can be designed based on the processing technique in this work.

Keywords

Highly tritiated water, palladium membrane reactor, hydrogen-water isotope exchange, decontamination factor.

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P6B3 ABSTRACT – 1295
C. Fuel Cycle and Tritium Processing

Halogen Removal from Tritium Gas Streams

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Halogens are unlikely impurities in the fusion fuel cycle, but their presence can be extremely corrosive in tritium gas streams. Chlorides are the most common halogen present, from either impurities in the metals that make us the tubing and process vessels, or from process catalysts like zeolites. Fluorides are less common in standard tritium processing, but it is a concern for certain tritium breeding mechanisms that use molten fluorinated salts. Fluorides are extremely corrosive and are a high hazard for personnel safety. The standard method to remove halogens from a gas stream involves bubbling the gas through a KOH solution to precipitate the halogen as a potassium halogen salt. However, for tritium processing, that is untenable as it would create large amounts of tritiated water to dispose of. A method to remove the halogens using a solid adsorbent is needed to ensure safe and effective processing over the course of a fusion plant lifecycle. Savannah River National Laboratory has synthesized a variety of adsorbent materials using numerous metals supported on a variety of zeolites and alumina. The synthesized adsorbents were tested for HCl and HF removal from dry hydrogen gas streams. Results will be presented on the halogen trap effectiveness and future work.

Keywords

Halogen, Tritium, Impurity Processing.

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P6B4 ABSTRACT-22e4
C. Fuel Cycle and Tritium Processing

Development of a Steady-state Simulation Program considering Energy Balance for Water Detritiation System

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This study focuses on the development of a simulator for water detritiation systems. The simulation program is based on a three-fluid, phase equilibrium-based hypothetical-stage reactive distillate column model. The model takes into account phase isotopic exchange reactions, isotopic self-exchange, and catalytic isotope exchange, utilizing both equilibrium and kinetic reaction-rate-based mechanisms. Notably, this model stands apart from existing models in the field due to its incorporation of heat balance, feed position, and additional feed entering the middle of the column, which are essential elements for effectively simulating the water detritiation process. The accuracy of the model was validated by analyzing and comparing the outputs of the program to experimental data under the same conditions. The simulator is based on Pyomo, a python-based optimization modeling language. This research demonstrates the simulation program has the potential to model water detritiation systems successfully, resulting in further enhancements in accuracy of the detritiation processes model. The significance of accurately modeling isotopic exchange reactions and the importance of heat balance, feed position, and additional feed for successful water detritiation system simulations are emphasized in this research.

Keywords

Water detritiation system, Simulation program, Three-fluid, Reactive distillate column, Heat balance.

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P6B5 ABSTRACT-9935
C. Fuel Cycle and Tritium Processing

Tritium inventory evolution modelling for demonstration and future fusion power plants

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Tritium is potentially becoming a limiting factor in the commercialization and widespread deployment of fusion energy. Scale-up from DEMO projects world-wide points to required on-site tritium inventories in the range of several kilograms per gigawatt of electrical power. Any early prototype plant or demonstration reactor will therefore not only require tritium self-sufficient operation but also rapid doubling of supplied start-up inventories in order not to act as a tritium sink in the global fuel economy.

This doubling time is a function of not only the achievable tritium breeding ratio but also of any tritium sinks such as the plasma chamber first wall, the overall operational timeline & availability of the plant, as well as of the total inventory required to operate the fuel cycle. Based on these aspects a model for the tritium inventory balance of a generic fusion power plant has been developed. Integrating this balance over the lifetime of a plant then allows the assessment of the doubling time, excess start-up inventory required, and total surplus inventory generated over the lifetime.

This contribution will present the developed model as well as its application to EU-DEMO, confirming the achievement of tritium self-sufficiency and finding doubling times of approximately 6 years. These results however indicate that significant improvements are still required for the next generation of fusion power plants to have their deployment not be limited by tritium availability. The developed model is also used to define the operational space available to achieve tritium doubling in less than two years.

Keywords

Tritium, Tritium doubling, Tritium self-sufficiency.

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P6C1 ABSTRACT – edcd
F. Nuclear System Design

Relevance of a high magnetic field to the design of the EU DEMO

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The limitation considered in the definition of DEMO in 2016 to not exceed the magnetic field strength foreseen in the ITER machine is reviewed here. At the time, a Nb_3Sn superconductor operated in a magnetic field, B_{max}, no greater than 13 T was considered to rely on mature technology. Here we do not consider this limitation in recognition that a first model coil with Rare Earth Yttrium Barium Copper Oxid (REBCO) superconductor was recently operated at a field of >20 T.

DEMO must have a burning plasma. At higher magnetic field strength, this condition is reached in a significantly smaller plasma. We carried out a system code study to size the DEMO machine varying the magnetic field. The outcome confirmed previous observations that the magnetic field strength is not a free parameter but linearly dependent on the plasma aspect ratio, A, when we target a minimum machine size. It is notable that in a machine with a given aspect ratio the increase of the magnetic field raises the plasma performance but does not allow reducing its size. The reduced plasma elongation of plasmas with high aspect ratio is a caveat as it compensates partly the gain in plasma performance achieved in the high field. Furthermore, the higher magnetic field at high A challenges the engineering of the magnets and associated structures, and substantially increases the particle loads on the divertor. This article assesses the engineering of the large DEMO coils operated at high field considering also advanced i.e., non-ITER like, mechanical concepts. None of these advanced concepts appear suitable for a large device like DEMO. From these assessments a magnetic field limit of approximately 11-12 T is identified to ensure the engineering feasibility of the DEMO coils.

Keywords

DEMO, tokamak, magnet coils, superconductor, magnetic field.

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P6C2 ABSTRACT – e561
F. Nuclear System Design

Characterisation of the neutron field for streaming analyses in TT operations at JET

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Assessing radiation fields in the ITER biological shield penetrations is a demanding task. Neutronics experiments are regularly performed at the Joint European Torus (JET) for validating in a real fusion environment the neutronics codes and nuclear data applied in ITER nuclear analyses. In particular, the fluence of neutrons passing through the penetrations of the JET torus hall is measured and compared with calculations in order to assess the capability of state-of-the-art numerical tools to correctly predict the radiation streaming in large and complex geometries. JET operated in 2021-22 with a Tritium-Tritium plasma. During the recent TT campaign, neutron fluence was monitored at several locations inside the torus hall at larger distance from the tokamak in order to asses neutron streaming. The measurements were performed with activation foils (AF) and thermo-luminescent detectors (TLDs).

The paper presents calculations of the neutron field at JET with emphasis on locations of the AF and TLD detectors. The neutron fluence and spectra at detector positions and the corresponding reaction rates in AF materials were calculated with a combination of the Monte Carlo N-Particle (MCNP) code and the deterministic solver of the ADVANTG code. This type of variance reduction was used to achieve the necessary low statistical uncertainty in the very demanding geometry of JET. Sensitivity of results with respect to changes in the plasma shape, i.e. in the spatial distribution of the plasma neutron source, is studied. The changes in plasma shape are a consequence of different plasma scenarios during individual JET discharges. The Monte Carlo statistical uncertainty of results is small and the uncertainty in the plasma shape was also found not to be significant. The largest uncertainty on the results is attributed to the difficulty of modelling the complex geometry of the JET torus.

Keywords

JET, Neutron streaming, Monte Carlo, Activation foils, TLD, neutron transport.

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P6C3 ABSTRACT – e53f
F. Nuclear System Design

Mock-up for the double door of CFETR & EU-DEMO maintenance casks for vertical ports: design & manufacturing progress

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In both the CFETR and EU-DEMO cask system for remote maintenance through vertical ports, contamination control during maintenance operations is an essential requirement to be fulfilled. A solution involving a double door with adequate sealing at the bottom of the cask has been developed within the close collaboration between European laboratories and ASIPP in Hefei. The suitability and adequacy of this solution is planned to be verified with a mock-up embedded in a dedicated test facility currently being built within the Comprehensive Research fAcility for Fusion Technology (CRAFT) project. In particular, the testing programme aims at verifying the sealing performance and the reliability of opening/closing actions.

Unlike the ITER-like rectangular section for the equatorial port, the opening section of the vertical port is trapezoidal. For the mock-up two separated containment enclosures are designed with several seal rings to simulate the maintenance cask and the tokamak vacuum vessel. Both containment enclosures can achieve a micro-negative atmospheric pressure condition to ensure no contamination dust leakage. Leak testing will be implemented to evaluate the tightness of these containment enclosures based on their hourly leak rate (according to ISO 10648-2). The upper door and lower door are automatically connected/disconnected with a dedicated lock mechanism. Both the upper and lower door are tightly clamped with the connection ports with latch structures. The opening/closing mechanism are driven by electrical motors and cylinders and the door motion envelope must not collide with components inside the cask or the extracted blanket module. Kinematic analysis and driving force simulation are conducted to ensure the effectiveness of this approach.

The purpose of this paper is to present the design solution for the double door mock-up and test facility as well as the manufacturing progress, including lessons learned from the interaction with industry and outlook of the testing methodology and plans.

Keywords

CFETR, DEMO, remote maintenance, double door, test facility.

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P6C4 ABSTRACT-b136
F. Nuclear System Design

Yield Estimation of Neutrons, Gammas, and Charged Particles in d-Li Target by using JENDL/DEU-2020 for IFMIF and Similar Irradiation Facilities

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An accelerator-based neutron source using d-Li reactions is one of the most promising neutron sources for fusion material irradiation facilities such as IFMIF, DONES, and A-FNS, where 40 MeV deuterons bombard a liquid lithium target. The d-Li reaction is very complicated consisting of several nuclear reaction processes. Therefore, the McDeLicious Monte Carlo code has been developed by Karlsruhe Institute of Technology group and is used for the d-Li neutron yield estimation including angular neutron spectra. Recently, a deuteron-induced reaction library of JENDL/DEU-2020 is released by the Japan Atomic Energy Agency. In our previous work [1], we confirmed that Monte Carlo calculation (MCNP and PHITS) using JENDL/DEU-2020 reproduced the experimental data and McDeLicious calculations on the d-Li neutron yield. In the present study, we carried out the yield estimation of not only neutrons but also gammas, and charged particles such as alphas, tritons, and 7Be, by using the PHITS code with JENDL/DEU-2020. Of course, neutron yield is the most important characteristic for the irradiation of fusion structural materials. However, gammas are also important in the irradiation of the functional materials and sensors in the low flux irradiation modules, and in the multipurpose applications of the irradiation facility such as neutron radiography and medical treatments. We found that the gamma-ray emission is isotropic, and is increased with the incident deuteron energy. Radiation activities of tritium and 7Be after the continuous one-year operation with deuteron energy of 40 MeV and 125 mA are estimated to be 7.5×10^{14} Bq and 5.8×10^{14} Bq. In this paper, we will discuss the comparison with the experimental data and the predictions by the McDeLicious code.

[1] T. Nishitani et al., Plasma and Fusion Research 16, 1405104(2021).

Keywords

Yield Estimation of Neutrons, Gammas, and Charged Particles in d-Li Target by using JENDL/DEU-2020 for IFMIF and Similar Irradiation Facilities.

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P6C5 ABSTRACT-2 a7e
F. Nuclear System Design

The Lead Lithium Loop for the European Water Cooled Test Blanket System (WCLL-TBS)

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Among the seven sub-systems that are part of the ITER WCLL-TBS, one of the most complex to be integrated is the Pb-Li loop. This loop will circulate melted Pb-Li through the TBM. During this circulation, the lead acts as neutron multiplier while the lithium acts as breeder. The tritium solubilized in the liquid metal is then extracted in gas phase through the Tritium Extraction Unit (TEU). The TEU, as well as the major part of the PbLi loop components, is located in the Ancillary Equipment Unit (AEU) area of the Port Cell outside of the Bio-Shield but still subject to certain magnetic field and neutron flux from the machine.

Due to heavy space constraints and the harsh environment existing in the Port Cell the challenge is even bigger. The WCLL PbLi loop will be installed in the PC#16 of ITER and will share the space available in the AEU with other three WCLL sub-systems and with other two sub-systems from the neightbouring TBS.

The requirements regarding access for maintenance and dose rate limits are very strict. As example, in order to minimize the exposure of workers, the dose rate in Port Cells 24 hours after machine shut-down shall be below 10 µSv/hr. At the same time, the maintenance activities can only be developed in scheduled dates so the PbLi loop shall accommodate the needed maintenance activities in the defined periods.

The design of the PbLi loop has been carried-out at a conceptual level taking into account all the aspects mentioned above. This paper analyzes the main issues in terms of design, operation and maintenance solved so far as well as the open points to be addressed in the next design phase. This next design phase is the Preliminary Design that will conclude in the Preliminary Design Review in 2024.

Keywords

TBM, WCLL, PbLi, Maintenance, Integration.

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P6D1 ABSTRACT – 5fa7
E. Vacuum Vessel and Ex-vessel Systems

Fusion Engineering Aspects of LHD Deuterium Experiment

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Deuterium experiment had been conducted on the Large Helical Device (LHD) since March 2017 to December 2022. The aims of the experiment were (1) to achieve high performance plasmas with the aid of isotope effects, (2) to investigate the isotope effect on plasma confinement and to explore the physics related to the effect, (3) to examine the confinement properties of energetic particles in helical devices and (4) to investigate the wall-material interactions in a toroidal plasma device.

In addition to them, several engineering issues/challenges are conducted, e.g., mass balance study of hydrogen isotopes in toroidal plasma devices using D-D fusion born tritium, development of negative-ion based neutral beam injector (N-NBI) with deuterium operation, development of divertor pumping system to assist in realizing high performance plasma and etc.

Because an Exhaust Detritiation System (EDS) was installed on LHD under the agreement with local government bodies and all of the vacuum exhausts are designed to go through the system, we can evaluate the total amount of tritium exhausted from LHD using the system. On the other hand, the total amount of tritium production can be evaluated from the D-D neutron measurement. Thus, we can perform accurate mass balance studies of tritium on LHD Deuterium experiment. The total amount of tritium production and exhaust were estimated to be 26.3GBq and 12.9GBq, respectively. Considering the radioactive decay of tritium, the tritium inventory is estimated to be 11.1GBq. Thus, almost half of produced tritium is estimated to stay in the VV including the pumping system.

In the presentation, detailed analysis on the tritium retention studies, developments of N-NBI and divertor pumping system will be shown.

Keywords

LHD, Deuterium experiment, Tritium Mass Balance, N-NBI, Divertor Pump.

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P6D2 ABSTRACT – 84fc
E. Vacuum Vessel and Ex-vessel Systems

Design of the DTT divertor cryopumps

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The Divertor Tokamak Test Facility (DTT) is a new device presently under construction in Frascati, Italy. Its main purpose is to study regimes with power exhaust conditions as close as possible to those foreseen in DEMO. The pumping system is a strong actuator in the operation of a divertor and hence has to be properly taken into account for integrated divertor design. Special care must be taken to ensure such that sufficiently high effective pumping speed can be delivered to the sub-divertor area.

As a result of a technology screening, DTT has decided to use a distributed cryopump system which is composed of nine identical units, placed in the lower vertical ports of the machine. Each cryopump unit consists of two cryosorption panels cooled by 4.2 K supercritical helium, and a thermal radiation shield surrounding the cryopanels made of a base plate and an inlet chevron baffle cooled at about 80 K. To have sufficient helium and hydrogen pumping capacity at these temperatures, the cryopanels will be coated with activated charcoal. Daily regeneration of the cryopumps at 100 K is foreseen overnight.

The paper will present the design of the cryopumps and highlight the main results of the supporting design development. Thermo-mechanical analyses have covered a variety of load cases, reflecting operation and regeneration conditions, electromagnetic forces under plasma disruption conditions, and accelerations under seismic events. Particularly challenging was to achieve a design of the component in a given space, limited by the triangular cross-section of the divertor ports. In this regard, the presented cryopump system comes with the maximum achievable pumping speed. It will be shown how the requirements put on the cryopumps have been considered in the final design and where the limits of the operational space will lie.

Keywords

Divertor, vacuum, cryopump, DTT.

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P6D3 ABSTRACT – 3219

E. Vacuum Vessel and Ex-vessel Systems

Radiofrequency electromagnetic analysis in the engineering of ITER Electron Cyclotron Heating Upper Launcher.

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The ITER Electron Cyclotron Heating system provides 170 GHz milli-wave heating and current drive to the ITER plasma, thus contributing to keep the plasma burning for the required time to produce the fusion reaction and controlling the plasma instabilities by compensating local temperature fluctuations. Four Upper Launchers, together with one Equatorial Launcher, installed in the Tokamak ports, constitute the front-end systems which guide the EC power to the plasma through a set of mirrors. Each Upper Launcher injects up to 8 MW of power into the reactor.

The technical challenges for the development of the ITER Upper Launchers involve high power millimetric radiofrequency waveguides and optics, high intensity neutronic and photonic radiation shielding structures, cooling systems integration and high vacuum and radiation resistant mechanisms, among others.

A key point in the engineering of the Upper Launchers relies in the capability to perform robust, reliable and efficient electromagnetic analysis of the millimeter waves and their interaction with the optical elements, launcher structures and finally the plasma. Special focus has been placed on the integration of this electromagnetic analysis in the end-to-end analysis cycle of the Launcher. After a thorough assessment on the different Computational Electromagnetic Methods (CEM), the Full Wave Method has been selected, in particular, the formulation based on Electric Field Integral Equations supported on the Method of Moments. The integrated procedure allows coupling the electromagnetic analysis with the thermo-structural analysis, so the perturbation of the electromagnetic beams due to mirrors deformation can be accounted. This methodology is aimed to obtain the parameters of the electromagnetic beams injected into the plasma, but also to verify the structural integrity of the mirrors and structures against the ohmic heating and stray radiation.

These works have been performed by IDOM and ALSYMEX in collaboration with F4E, ITER Organization and specialized European Fusion Laboratories.

Keywords

ITER, Electron Cyclotron Heating, Upper Launchers, mm-wave electromagnetic análisis.

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P6D4 ABSTRACT-4d21
E. Vacuum Vessel and Ex-vessel Systems

Commissioning and initial operation of HL-2M vacuum system

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HL-2M is a new copper-conductor tokamak constructed in Southwestern Institue of Physics(SWIP) and achieved its first plasma in Dec. 2020. It can be operated with high performance plasma to support the operation of ITER and the design of future fusion devices.

The approach to HL-2M vacuum system commissioning makes a clear distinction between sub-system commissioning and integrated commissioning. When all of sub-systems of vacuum system including pumping, leak testing, wall conditioning and gas puffing had passed their tests respectively, the integrated commissioning was carried out to ensure the vacuum system work normally together. Firstly, all air had been evacuated out of vessel to bring the pressure inside to ~ 10^{-4} Pa by turbomolecular pumps sets, then leak testing was organized. Prior to the wall conditioning, the baking operation and glow discharge cleaning (GDC) procedures were carried out to verified that all is well. After that, the baking at 100 ◦C for 50 hours and intensive H2-GDC were performed to remove impurities from the first wall surface. In close coordination with central control, a series of injection tests with H2 on gas puffing system had been performed. After several weeks of commissioning, the HL-2M vacuum system is ready for the first plasma, with ultimate vacuum of 2.4×10^{-6} Pa.

During the initial phase of HL-2M, similar procedures is performed. In particularly, an alternative wall conditioning procedure, RF assisted GDC, was developed. Initial breakdown was produced by an induction coil at radio frequency of 2.1 MHz, 13.56 MHz or 40.68 MHz respectively, then DC bias was applied for coupled scenarios. Compare with DC GDC, a lower breakdown pressure ~10^{-1} Pa was obtained as well as higher impurities yield (80%).

Keywords

HL-2M, Vacuum system, Wall conditioning, RF assisted GDC.

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P6D5 ABSTRACT-a7f4
E. Vacuum Vessel and Ex-vessel Systems

Assessment of the stability of an optimized cooling configuration of Gyrotron resonant cavity for nuclear fusion machines through an analytical and numerical model

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Gyrotrons are a promising technology for electron cyclotron resonance external heating in nuclear fusion machines. As the demand for heating power continues to rise, there is an increasing need for higher tube beam power; however, one of the main challenges remains the high amount of energy released on the inner wall of the resonant cavity. This release leads to high temperatures, displacements, and subsequent frequency shifts in the electromagnetic power, reducing the efficiency of the tube. Therefore, efficient cooling of the cavity is crucial.

The heat load released is not uniform and can reach peaks of ~25 MW/m², causing thermal gradients in the cavity structure and, consequently, thermal stresses. Moreover, it is essential to maintain the pressure drop below a specific limit to reduce the power required by the pump used for the sub-cooled water flowing in the gyrotron cooling circuit. All these aspects were considered in a previous optimization study, which utilized a Biogeography-Based optimization algorithm to determine an optimized axial profile for the heat transfer coefficient (HTC) of the cavity coolant. A straightforward engineering solution was designed to achieve the identified HTC profile.

The study was conducted under steady-state conditions; however, even with the optimized HTC the cavity walls will still experience displacements, potentially leading to changes in the cooling system configuration and, consequently, a different HTC profile. The present study aims to verify the stability of the proposed engineering solution by developing an analytical and dynamical numerical model that accounts for the displacements in the system and the resulting changes in the HTC. Additionally, an initial computational fluid dynamics verification is performed, demonstrating that the displacements on the external wall lead to a reduction in temperatures and, therefore, a decrease in further displacements.

Keywords

Gyrotron cavity cooling, heat transfer enhancement, design optimization.

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POSTER SESSIONS

PS1-1 ABSTRACT-781e
A. Plasma-Facing High Heat Flux Components

The thermohydraulic parameters of copper foam based cooling channel.

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Power exhaust and high heat flux cooling are significant engineering challenges of the nuclear fusion reactor construction. Water cooled cooling channel filled with conductive copper foam is presented as an alternative solution to present cooling channel geometries (hypervapotron/swirl tube). The thermohydraulic parameters (pressure drop and heat transfer coefficient) are experimentally investigated. The results of experimental study are compared with numerical analysis to evaluate the reliability of CFD model. The application of copper foam based cooling channels in the divertor area is discussed.

Keywords

High Heat Flux Cooling, Copper Foam, Water Cooling, Divertor Targets, CFD.

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PS1-2 ABSTRACT-122f

A. Plasma-Facing High Heat Flux Components

Experimental Study of Boiling Heat Transfer Characteristics of Multi-Elbow Cooling System under One-sided Heating with Long Area for Fusion Divertor Cooling

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Divertor cooling is one of the most important issues in the realization of a fusion reactor. In the case of the ITER, an extremely high heat flux of 10 MW/ is assumed to be irradiated during steady-state operation, and swirl tubes with twisted tapes inserted are used as cooling channels. Although swirl tubes enhance the heat transfer performance by forming a swirling flow, they have engineering issues such as the difficulty of machining and the increase in pressure drop due to the insertion of the tape.

On the other hand, as an alternative to swirl tubes, a cooling system using swirling flow formed downstream of three-dimensionally connected multi-elbows has been proposed. This cooling system is expected to solve the engineering issues of swirl tubes because of its simpler structure and smaller pressure drop.

In this study, heat transfer experiments were conducted under one-sided heating conditions using 2 atm water as the coolant and an electron gun as the heat source. To evaluate the boiling heat transfer characteristics of the swirling flow formed by the multi-elbow, a 10 mm wide × 60 mm long section in the flow direction (0.5 D to 6.5 D from the elbow outlet) of a copper specimen (diameter of flow channel, D, =10 mm) was irradiated with heat flux by the electron gun and temperature distributions in the flow, circumferential, and radial directions were measured. The experimental results showed that the wall temperature cooled by the water flow in the case of swirling flow by the multi-elbow was lower than that of the straight pipe system when irradiated with the same value of heat flux. Thereby, the heat flux at the boiling start and DNB points was larger than that of the straight pipe.

Keywords

Boiling heat transfer, multi-elbow, divertor cooling, swirling flow.

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PS1-3 ABSTRACT-1e2d
A. Plasma-Facing High Heat Flux Components

Numerical simulation of the erosion process in W lattice armor for sacrificial limiters under high heat flux

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One of the greatest challenges in the demonstration of fusion energy stands in the need of strongly mitigating the degradation of the breeding blanket first wall (FW) plasma-facing components (PFCs) during plasma instabilities and disruptions. For this reason, discrete sacrificial limiters are foreseen in the EU-DEMO reactor to protect the un-shadowed reactor wall during plasma transients. Such components are easily replaceable and must be capable of sustaining several disruptive events. Above all, their engineering must comply with conflicting requirements, i.e. adequate heat exhaust during normal operation but thermal insulation of the heat sink during transients. In this regard, W lattices have been preferred to a dense armor, as the former provide a larger design flexibility, high surface-to-volume ratio, and superior insulation of the heat sink.

In this contribution, we present the results of finite element simulations implemented to assess the erosion of the lattice armor material under extremely high heat fluxes. Preliminary analyses were carried out in Ansys Mechanical in presence of strong non-linearities linked to modelling the phase change of W including the latent heat. In second step, an existing MAPDL routine was integrated in the solid model to deactivate the "vaporized" mesh elements at each time step. The routine gauges the vaporized volume at different combinations of exposure time (1-4ms) and heat flux magnitude (50-300GW/m^2). Two lattice geometries and a dense W armor were analyzed. The results suggest a lower rate of erosion (mass/time) in the lattice structure compared to the dense W, although, given the low relative density of the lattice, this equates to a faster degradation of the component (depth of erosion). Further refinement of the model (i.e., accounting for vapor shielding effect, accounting for the increased wetting area provided by the lattice) are needed in order to corroborate or disprove these results.

Keywords

Plasma Facing Components, Tungsten erosion, FEM simulation.

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PS1-4 ABSTRACT-22b5
A. Plasma-Facing High Heat Flux Components

Design and fabrication of preliminary mock-ups for the Outboard First Wall of the DTT facility

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The Divertor Tokamak Test (DTT) facility is the tokamak under construction at ENEA C.R. Frascati within the EUROfusion programme. Steel-based actively cooled plasma-facing components have been chosen for the First Wall (FW) of DTT, with the aim of developing and manufacturing solutions that could be of interest for a future fusion power plant. In this framework, preliminary mock-ups must be designed and fabricated, in order to build medium-scale prototypes and to identify a viable manufacturing route for the final FW modules.

This work describes the prototyping activities for the DTT outboard FW modules, which consist of water-cooled steel plates coated with a tungsten (W) armour. The coating can be deposited by means of high-pressure plasma spray. Regarding the steel plates, a machining and welding process is selected as a conventional and mature manufacturing solution. The additive manufacturing technology of selective laser melting is also taken into account as a promising alternative, because it provides high geometrical precision, high flexibility in the realization of complex parts and low costs. For both manufacturing solutions, specific small-scale mock-ups have been designed, optimized and fabricated to overcome the main manufacturing issues. In this phase, destructive and non-destructive tests have been carried out on the mock-ups in order to assess the feasibility of the technological processes and the related results are presented in this contribution.

Keywords

DTT, First Wall, Plasma-facing components, Mock-ups.

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PS1-5 ABSTRACT-2b06
A. Plasma-Facing High Heat Flux Components

Multi-physics lumped modeling of micro-channels cooling structure for W7X divertor unit target module

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A novel concept for the cooling system of the W7X divertor unit target module has been proposed, which involves tiles equipped with parallel arrays connected by hundreds of sub-millimeter rectangular micro-channels (MCs), obtained using Additive Manufacturing techniques. The structural material proposed for the Heat Sink Substrate is galvanized copper, while the Plasma Facing Material is tungsten.

To reduce the high computational cost of thermal-hydraulic simulations of the tiles, a lumped modeling approach has been built in previous studies. This involves replacing a group of hydraulic parallel MCs with a porous strip (PS), suitably calibrated to reproduce similar thermal-hydraulic behavior of the MCs. The aim of the current work is to verify the PS-model is also suitable for the thermo-mechanical stress evaluation, comparing the results from arrays with MCs and PS. In particular, since the heat sink and plasma-facing tiles are bonded, the interfacial delamination and shear stresses are analyzed, being crucial for structural integrity.

The analyses, carried out in the elastic regime, show that the PS model returns conservative results when compared to the MCs model, with an overestimation of the delamination and shear stresses at the free edge. Results from both models reveal nearly identical interfacial stress predictions at the free surface edge. Moreover, it is found that both MCs and PS blocks, located near the bond interface, contribute to high-stress fluctuations that could lead to the delamination of the bond interface, suggesting that the distance of the microchannels from the bond interface should be increased.

The PS model can be reliably used to design and verify the most convenient series/parallel connection of the tiles in the divertor unit target module, taking operational constraints into consideration.

Keywords

Divertor, micro-channels, numerical modeling, thermo-mechanics, thermo-hydraulics, lumped modeling.

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PS1-6 ABSTRACT-d38c
A. Plasma-Facing High Heat Flux Components

Capillary porous structure based divertor transient heat event mitigation capabilities modelling

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Capillary porous structure (CPS) based liquid metal divertors are currently being investigated as a possible alternative to the tungsten based solid plasma facing components. The ability of CPS to withstand high heat fluxes (>20 MW/m2) [1] has been already demonstrated in linear devices as well as tokamaks. One area where liquid metal plasma facing components can excel over solid made ones is the survivability of transient heat events (such as ELMs). Flash melting of the tungsten monoblock in the ITER divertor will occur during unmitigated ELMs. The capability of CPS to withstand such shocks has already been proven [1].

Furthermore the strong erosion of the liquid surface can serve as a passive ELM (and other transients such as VDE) mitigation tool. The enhanced erosion during ELMs by immediate sputtering, further increased by evaporation, provided a sufficient surface temperature is reached, results in stronger vapour shielding mitigating the increase of the heat flux. Predictive modelling of future liquid metal experiments on COMPASS upgrade indicates that ELM induced temperature spiking can be suppressed completely [2,3].

This work aims to investigate this effect under ITER (and DEMO) like transient loads via modelling in the HeatLMD code [3]. Temperature evolution as well as cooling power and liquid metal erosion of various scenarios, both in terms of plasma parameters and divertor type will be presented. Moreover the ability of the divertors to withstand (and potentially mitigate ELMs) will be evaluated. Finally the compatibility of the scenarios with fusion plasma central radiation induced by liquid metal erosion, will be addressed.

[1] Tabares, et al., Nucl. Fusion 57 (2017) 016029 (11pp)

[2] Cecrdle, MSc. thesis, 2021, FNSPE CTU

[3] Horacek et al., 2021, Phys. Scr. 96 124013

Keywords

Divertor, Liquid metals, CPS, ELM, VDE.

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PS1-7 ABSTRACT-df22
A. Plasma-Facing High Heat Flux Components

Experimental evaluation of capillary effects relevant to the design of semi-static liquid lithium plasma-facing components

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Liquid plasma facing components are being considered as a viable option for future machines in order to adequately handle the exposure to intense energy fluxes expected in components such as limiters and divertors. In addition to free surface flow and static liquid plasma facing components for fusion devices, approaches based on the capillary flow of lithium across porous structures have been proposed as plasma facing components, which avoid the overheating risks associated with static liquid pools and the risk of macroscopic material ejection that is always present in free surface designs. In this work, macroporous tungsten samples prepared by spark plasma sintering have been exposed to lithium droplets in a vacuum environment. These experiments have provided information not only regarding the wetting properties of lithium on tungsten, but also a way to experimentally measure capillary flow of lithium at zero pressure. An effective diffusion coefficient has been calculated for use in the Lucas-Washburn equation to quantify mass transport flow within the porous structure and also the evaluation of sample performance. Surface analysis results on pre and post lithium exposure by X-ray photoelectron spectroscopy reveal the importance of impurity levels in tungsten that derive into radically different behavior of samples. Details on the design of a pressurized lithium flow vacuum test-stand to study lithium Darcy-like flow across porous structures will be presented as well.

Keywords

Plasma facing components, liquid metals, lithium, tugnstent, spark plasma sintering.

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PS1-8 ABSTRACT-7545
A. Plasma-Facing High Heat Flux Components

Conceptual design of a 3D printed liquid lithium divertor target for DEMO

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Liquid metal divertor (LMD) is being developed as a robust alternative to conventional tungsten divertor. Over the past several years, a series of prototype capillary-based LMD components have been designed and demonstrated promising performance in several plasma pulse operations and high thermal load tests. Some of these prototypes feature pre-loaded 3D printed tungsten porous structures. Based on the foundation of these previous efforts, the 3D-printed liquid metal will undergo a comprehensive structural design to ensure it meets the demanding operating conditions of DEMO. Therefore, the aim of this work is to investigate and showcase novel design that integrate LMD with 3D printing technology, ensuring reliable performance at high heat flux.

To achieve this goal, some relevant requirements were developed in combination with the operating conditions of the DEMO. Lithium was chosen as the preferred liquid metal due to its ability to reduce recycling, increase energy confinement and mitigate anomalous heat transport. A conceptual design was then chosen that consisted of a water-cooled system and a capillary porous system (CPS) made of 3D printed tungsten filled with liquid lithium. Lithium is supplied from a reservoir to the plasma facing surface via capillary action in a wicking structure. This study has been carried out using thermo-mechanical analysis to exhibit the detailed results of the thermal response, stress distribution, and deformation. The results demonstrate the potential of this structure, offering a viewpoint for optimizing its performance and improving power handling limitations. It will provide valuable insights for designers to investigate diverse liquid metal divertor target designs, which will comply with the anticipated higher-parameter fusion reactor operation requirements in the future.

Keywords

Liquid metal divertor, Lithium, Capillary, DEMO.

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PS1-9 ABSTRACT-6a2c
A. Plasma-Facing High Heat Flux Components

Comparative study of ITER conform tungsten grades exposed to high heat flux and neutron irradiation damage

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Tungsten is plasma-facing material (PFM) for divertor and test blanket module to be deployed in ITER. During operation, tungsten sub-components will be subjected to cyclic heat exposure, plasma particles loads and bombardment by fast neutrons. The optimization of tungsten grades hopefully offering enhanced resistance against high heat flux and irradiation damage is ongoing, and recently ALMT has proposed an extra cross-rolling step for the fabrication of thick tungsten plates, showing improvement of tensile properties.

In this work, we present results of recent experiments aimed at characterizing the impact of high flux plasma load, thermal shock and neutron irradiation exposure on the damage induced in three tungsten grades. Two grades are conventional ITER specification plates supplied by ALMT and AT&M, and the third plate is developed by ALMT by applying cross-rolling step. The assessment included the tensile mechanical properties and hardness measurements before and after the irradiation as well as microstructural investigation of surface modification due to the high heat flux testing (i.e. laser beam and plasma exposure). The neutron irradiation was performed at BR2 reactor (Belgium) in a wide range of temperatures (400-1200 °C) up to the irradiation dose of 0.3-0.5 dpa. The thermal shock tests were performed at the linear plasma device PSI-2 (Germany) at the base temperature of RT, 400 and 1000 °C to investigate damage initiation and development below, near and above the DBTT after 100 and 1000 transients. The high flux cyclic plasma exposures were performed at Magnum PSI (the Netherlands) at the base temperature of 600, 800 and 1200 °C with in-cycle temperature escalation of +300 °C to follow initiation of the cracking after 10^4 pulses. The obtained results are discussed and compared with the trends established earlier for the ITER specification tungsten studied over the last decade.

Keywords

Tungsten, products, damage, irradiation.

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PS1-10 ABSTRACT-0795

A. Plasma-Facing High Heat Flux Components

Molecular dynamics study on the slip properties of liquid lithium flow on iron surfaces

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As the use of liquid lithium (Li) as the plasma facing material in a fusion device becomes widespread, more and more research work has been dedicated to the numerical simulations of Li flow based on the Navier-Stokes equations. However, the slip condition of Li on a solid surface hasn't been fully understood. The most common and simplest boundary condition, which is no-slip, is just one of the allowable conditions ranging from pure slip to multilayer locking. In this work, molecular dynamics simulations were performed to investigate the slip properties of liquid Li on iron (Fe) surfaces. The atomic structures near the surfaces were inspected. The influence of surface roughness was discussed. It was found that the slip length is always negative suggesting that the wall always retards the movement of liquid Li. Surface roughness has significant effects on the slippage. Two sectional linear relationships between the slip length and the height of the roughness elements were discovered. As the height of roughness elements reaches a critical point, micro vortexes begin to form and change the slope of the linear relationship.

Keywords

Liquid lithium, Couette flow, Slip length, Molecular dynamics.

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PS1-12 ABSTRACT-08ee

A. Plasma-Facing High Heat Flux Components

Completion of the first lower divertor cassette of JT-60SA

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A lower divertor cassette of JT-60SA, which is the largest superconducting tokamak, has been designed, and the first divertor cassette covering a 10◦ sector in the toroidal direction has been manufactured. Modularized plasma facing components such as vertical targets, baffles, and domes will be assembled into the divertor cassette. All plasma facing components will be actively cooled. Carbon armor tiles bolted to water-cooled copper alloy heat sinks are used to remove heat loads of 0.3–2MW/m2 for 100 s and 15MW/m2 for 5 s. The divertor cassette integrated with coolant pipe connections will be used for remote handling / hands-on maintenance. A remote pipe welding tool, which is accessed from the inside of the pipe, has been developed to assemble the inboard vertical target. The space around the cooling pipes is so limited that they are cut and welded remotely from the inside for maintenance. The laser welding method was used, and the focusing mirror inside the pipe was rotated to perform circumferential welding. As a countermeasure for welding groove misalignment, the aiming accuracy of the laser was improved by simultaneously controlling the rotation and lifting of the tool. Thereby, reliable welding has been achieved even with a groove that is misaligned by 0.5 degrees or more. The outer diameter of the cooling pipe is 59.8 mm, the wall thickness is 2.8 mm, and the material is SUS316L. After connecting the inboard vertical target, the other plasma facing components have been assembled. The first lower cassette has been completed.

Keywords

Divertor cassette, in-vessel component, maintenance, and JT-60SA.

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PS1-14 ABSTRACT-0c22
A. Plasma-Facing High Heat Flux Components

Full-field kinematic and thermal characterisation of a DEMO divertor mono-block under high heat flux loading

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In-vessel components such as divertors are subjected to extreme thermomechanical, magnetic and neutron irradiation loads. Existing divertor mono-blocks are experimentally tested under high heat flux fatigue using various methods. However, the diagnostics used for current tests only provide qualitative information regarding the state of the component in terms of failure. Enabling the use of engineering simulations for fusion component design requires that models are quantitively and systematically validated against experimental data for conditions that can be tested. Therefore, newapproaches are needed to collect quantitative data during high flux tests and include appropriate uncertainty quantification to enable comparison to models.
In this work, we conducted a full-field thermal and mechanical characterisation of the DEMO thermal break mono-block design under steady state high heat flux ranging between 5 and 10 MW/m2. The tested design is composed of a tungsten mono-block, a copper-chromium-zirconium pipe, and a pure copper interlayer. The experiment has been conducted using the Heating by Induction to Verify Extremes (HIVE) facility at UKAEA. A stereo Digital Image Correlation (stereo-DIC) system was used to measure the in-plane and out-of-plane kinematic fields on the side of the mono-block. The temperature field of the same side was measured using an infrared bolometer camera. The termal field was then mapped onto the deformed shape of the sample using a spatial calibration procedure, which aligns the stereo-DIC cameras and the infrared camera to the same coordinate system. Preliminary experimental results including uncertainty quantification of the measured kinematic and thermal fields are presented herein. These results can be used as the input data for a model updating procedure to identify materials parameters and to validate numerical models considering experimental uncertainties.

Keywords

High heat flux, Digital Image Correlation, Plasma facing component, Infrared Thermography.

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PS1-15 ABSTRACT-11f1
A. Plasma-Facing High Heat Flux Components

Selection of EU-DEMO divertor operating condition: design space and power exhaust capabilities

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In the framework of EUROfusion activities for the development of the DEMO reactor design, the divertor configuration represents one of the main challenges to be addressed. The divertor system has to sustain high heat fluxes due to the ion particle while is subjected to intense neutron fluxes, allowing at the same time the shielding of the vacuum vessel as well as the vacuum pumping. The concept design of the divertor is based on the use of EUROFER97 as structural material for the divertor cassette body, while tungsten monoblocks bonded to CuCrZr pipes are used for plasma-facing targets. EUROFER97 was chosen considering its low long-term activation and superior creep and swelling resistance under neutron irradiation. However, depending on the operating temperature under neutron irradiation, a pronounced shift of the Ductile to Brittle Transition Temperature (DBTT) is expected. At the same time, for the plasma-facing targets, the coolant temperature has to be defined such to allow sufficient heat removal capacity. This work presents the activities developed in the conceptual design phase to investigate alternative cooling conditions for the divertor system that are able to ensure the fulfilling of functional requirements and to allow for divertor cassette body re-use during plant lifetime. The aim is to identify the best thermal-hydraulic conditions avoiding material embrittlement (for EUROFER 97) and softening/hardening (for CuCrZr). At the same time, the goal is to reduce the inventories (enthalpy of the cooling circuit) and the radwaste at the end of Divertor lifetime. Parametric thermal-hydraulic analyses of the Vertical Targets cooling circuit have been performed in order to obtain its operating map and to select a suitable design point. For the cassette body, a “high” inlet coolant temperature (285-295°C) was considered as a cooling option for the EUROFER97 to avoid embrittlement up to the end-of-life (2 FPY) damage dose level.

Keywords

DEMO, Divertor, Water-Cooling, Thermo-Hydraulic.

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PS1-16 ABSTRACT-14d5
A. Plasma-Facing High Heat Flux Components

Structural Analysis of the DEMO Upper Limiter preliminary design

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The EU-DEMO Upper Limiter (UL) is part of the first wall protection system primarily designed for facing the energy released by plasma during transients. The UL consists of the Plasma-Facing Wall (PFW) directly exposed to the plasma and a Shielding Block (SB) attached to the VV. The PFW is made of tungsten connected to the SB. The SB and all the attachments are made of Eurofer97.

The UL design must be capable of withstanding a range of loads that are pertinent to in-vessel components. In particular, the most demanding loads that act on the UL are the thermal and electromagnetic loads. Neutronic irradiation generates thermal loads, whereas electromagnetic loads are induced by the current in the conductive structure and the magnetic field.

This paper presents the results of the structural FE modeling (using ANSYS) of the preliminary design of the UL, taking into account both thermal and electromagnetic loads. Primary load structural assessments were conducted in accordance with the RCC-MR design standard. Structural criticalities were identified in the attachment systems, and design improvements have been proposed.

Keywords

EU-DEMO Upper Limiter, Structural analyses, FE Modelling and Structural integrity.

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PS1-17 ABSTRACT-1808

A. Plasma-Facing High Heat Flux Components

A Study on Stress Mitigation of Process Pipes in TBM-set

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Test blanket module (TBM) is to be installed at the equatorial port of ITER to demonstrate the production and transport of tritium, as well as heat removal in high temperature environments. The TBM set consists of a TBM box, a TBM shield, and process pipes passing through the TBM shield that are connected to the TBM box by welding. These pipes include a cooling pipe to cool the TBM box, a pipe to carry the generated tritium, and a guide pipe for instrumentation & control (I&C). The pipes should be designed to ensure structural integrity without affecting the radiation shielding of the TBM shield. For this purpose, mechanical analysis of the TBM set was performed using ANSYS according to the RCC-MRx code and standard, considering various loads and conditions, including thermal load, pressure load, and dead weight. The design conditions were also varied by changing the number of pipe bends, the direction of the bends, the position of the welds at both ends of the pipe, and the shape of the joints. The main stress concentration is caused by the axial expansion of the pipe due to the heat load, and various approaches for stress mitigation in the TBM-set process pipes are addressed in this study.

Keywords

TBM, Process pipe, ANSYS, RCC-MRx.

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PS1-18 ABSTRACT-1a5e
A. Plasma-Facing High Heat Flux Components

A multiscale tool for Plasma Facing Materials aging simulations

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Describing the temporal evolution of plasma facing materials (PFMs), meaning how their physical characteristics (such as erosion length, roughness, conductivity, bulk defectiveness, and plasma components retention) change during plasma exposure, is crucial for designing the energy supply of the near future.

Before operating a direct monitoring of the PFM aging in situ, it is useful to estimate their microstructural modification due to high thermal loads with appropriate predictive models. Our aim is to design a PFM aging simulation tool based on a multiscale approach. This consists of sequential codes which follow the wide range of physical phenomena among their typical scales.

Here we will present the current state of the first two parts of our code and show some original results.

The first one, which describes the plasma thermodynamics and stoichiometry for time-dependent solutions, together with the evaluation of the sheet potential, has already provided results in line with the literature for hydrogen (H), argon (Ar), and helium (He) plasmas. To include surface reactions and to communicate with the other fully 3D tools we have inserted geometric corrections, which means it can be adapted to many devices (for instance linear plasma devices).

Three-dimensional KMC (which simulates the sequences of surface and near-surface structural modifications caused by the impinging of the plasma components on the PFM) allows for the description of several processes that can be entered as routines and calibrated by reaction matrices. This makes the code customizable and adaptable to communicate with other programs and different materials, for example for body centered cubic tungsten (bcc-W).

Our first results show that this approach can describe the quasi-atomic detail of the PFM evolution, revealing that the plasma-wall interaction is a complex problem showing a non-trivial dependence on the initial parameters.

Keywords

Nuclear fusion, Plasma Facing Materials, kMC, plasma, simulations, Hydrogen, Helium, Tungsten bcc.

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PS1-20 ABSTRACT-f5dd
A. Plasma-Facing High Heat Flux Components

Development of the Liquid Metal Shield Plasma-Surface-Interaction (LiMeS-PSI) linear plasma device.

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A promising alternative to tungsten based plasma facing components (PFCs) for the heat exhaust problem of future fusion reactors are liquid metal (LM) based divertor designs. However, compared to the tungsten PFCs designs, LM design are currently less advanced while being technologically more complex. To provide a key stepping stone from small scale LM experiments to medium scale tokamak implementation, the Liquid Metal Shield Laboratory (LiMeS-lab) is currently under development to test and compare LM divertor designs.

Central to LiMeS-Lab will be a new cascaded arc linear plasma generator (LiMeS-PSI), which will be a test-bed for LM based divertor component designs, including circulating liquid metal flows (lithium and tin). To study the plasma-LM interaction, LiMeS-PSI will facilitate plasma loading in DEMO relevant conditions (n_e~10^{20} m^{-3}, T_e~1 to 5 eV, Γ~10^{24} m^{-2} s^{-1}, q>20 MW m^{-2}), utilizing a steady-state superconducting magnetic field at 1.5 T. The use of LMs introduces several challenges in the development of LiMeS-PSI. These include handling reactive lithium, preventing atmospheric contamination, driving the liquid metal circulation, and protecting diagnostics from metal vapour.

Adaptations to deal with these challenges include replacing water cooling with oil cooling and implementing emergency argon gas systems for lithium safety. Additionally, the targets will be mountable and transferable in a protected atmosphere. A specially designed electromagnetic pump concept using the static magnetic field of the linear plasma generator will drive the LM circulation. To protect the diagnostic vacuum windows against LM vapor contamination, mirrors are inserted in the line of sight to avoid degradation of optical transmission in long-pulse operation.

In this contribution, the current status of the LiMeS-PSI development will be shown, the methods for dealing with these challenges and their implementation in LiMeS-PSI will be discussed, and experiments used to verify the design concepts are presented.

Keywords

Linear Plasma Generator, Liquid Metals, Lithium, Tin.

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PS1-21 ABSTRACT-f739
A. Plasma-Facing High Heat Flux Components

Overview of the DTT Assembly Plan

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The Divertor Tokamak Test (DTT) facility is an experimental tokamak devoted to investigating the systems for the treatment of heat exhaust (divertor, configuration control system, diagnostics) to be used in Controlled Thermonuclear Fusion reactors. The mission of the DTT project is to contribute to the early realization of fusion energy by addressing key physics issues for ITER and DEMO and testing several types of magnetic configurations and divertors.

The magnetic system of DTT is composed of 18 toroidal field (TF) coils, 6 central solenoid (CS) stacked modules and 6 poloidal field (PF) coils. All of these coils are superconducting, cooled by supercritical helium at 4.5 K and thermally protected in a metallic cryostat. The reactor assembly procedure of DTT is analysed covering the installation of all the main magnetic and mechanical Tokamak components up to the first plasma. The assembly plan identifies the assembly procedures, Non Destructive Tests (NDTs) and metrology to be used for the machine assembly. The analysis carried out in the paper investigates the assembly of the cryostat base, the pre-assembly of the lower PF coils, the assembly of the Vacuum Vessel (VV) sectors and ports covered with the Thermal Shield (THS), the insertion of the TF coils and the torus closure through the insertion of the final 20° sector. The final upper PF coils installation is followed by: the insertion of the Central Solenoid (CS), fixing He distribution piping and feeders inside the cryostat, mounting the in-vessel components (In-vessel coils, first wall, divertor, etc) and assembling the cryostat thermal shield, the cylindrical body and Top Lid. The preliminary integration of the out-vessel auxiliary systems, including the additional heating systems foreseen in the first phase of the facility, are also analysed.

Keywords

Overview of the DTT Assembly Plan.

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PS1-22 ABSTRACT-1f1c

A. Plasma-Facing High Heat Flux Components

Application of Tungsten Recrystallisation in Plasma Facing Components Design

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Tungsten has been chosen as the main candidate for plasma facing material in fusion reactor design. Its recrystallisation, which is a function of temperature and exposure time, plays an important role in the component lifetime. The recrystallisation process has been widely studied through material tests, thermal shock tests and numerical modelling for different material grades to investigate their microstructure changes and thermal fatigue response. However, the design criterion, i.e. temperature limit against recrystallisation, to be used in steady state and transient thermal analysis of plasma facing components is still poorly defined.

The aim of this work is to 1) present preliminary methodologies and criteria for designing with tungsten, taking recrystallisation into account and 2) propose feasible measures to improve the resilience of tungsten to recrystallisation in the early design stage. For this purpose, the required exposure time before primary and secondary recrystallization occurs has been plotted based on existing test results across a wide temperature range. To explore the acceptable level of tungsten recrystallisation, the impact of recrystallised depth, armour thickness and coolant temperature has been studied by numerical simulations. Initial analysis suggests transient events result in the inevitable onset of recrystallisation and that, once any level of recrystallisation is present, recrystallised depth has a negligible impact on fatigue lifetime during steady state operation. The operational regime will therefore be critical in understanding the impact of recrystallisation. In addition, surface roughening due to recrystallisation may lead to leading edges being exposed to shallow incident angle particle loads. This may cause cascaded local melting and material loss, enhancing erosion and potentially contaminating the plasma. Moreover, the influencing variables on the intrinsic activation energy, which dictates tungsten recrystallisation resistance, have also been investigated. Ultimately, the overall strategy of the design criterion against recrystallisation in plasma facing component design process has been discussed.

Keywords

Tungsten recrystallisation, design criteria, plasma facing components.

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PS1-23 ABSTRACT-0ff3
B. Blanket Technology

Development of repair technology for functional ceramic coating by recoating

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Tritium permeation through structural materials in fusion reactor blankets causes fuel efficiency reduction and radiological hazards. Functional coatings for tritium permeation barrier and corrosion protection have been developed using ceramics such as zirconium oxide (ZrO_2). During the long-term operation in the reactor, the coatings are exposed to thermal cycles, neutron and gamma-ray irradiation, and high-temperature tritium breeders, leading to degradation. Therefore, it is necessary to repair or replace the coated components in the periodic inspection; however, repairing technology for damaged coatings have not been developed yet. In this study, we first investigated on repairing of ceramic coatings through recoating for the practical application of the functional coatings.

ZrO_2 coatings were fabricated on reduced activation ferritic/martensitic steel F82H substrates by metal organic decomposition. The coated sample was thermally damaged by the temperature change from room temperature to 625 °C with an increase/decrease rate of 10 K/min using an infrared gold image furnace. After that, we conducted recoating for the damaged sample with the same coating process. Deuterium permeation tests, scratch tests, and surface/cross-sectional observations by scanning electron microscopy and laser microscopy were conducted to evaluate the coating properties.

The deuterium permeation flux of the damaged sample increased by one order of magnitude compared to that of the undamaged one. On the other hand, the permeation flux of the recoated sample showed two orders of magnitude lower than that of the damaged one and three orders of magnitude lower than that of the sample before damage introduction, proving a recovery of permeation barrier performance by recoating. In the presentation, the results of surface/cross-sectional observations and the scratch tests will be included for further discussion.

Keywords

Ceramic coating, recoating, damage, tritium, blanket, repair technology.

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PS1-24 ABSTRACT-9b26
B. Blanket Technology

Kinetics of helium bubble formation in liquid lead-lithium eutectic: an atomistic-simulation study

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The EU DEMO breeding-blanket design includes liquid Lead-Lithium Eutectic (LLE) alloy as a promising candidate for cooling and/or breeding tritium to attain the self-sufficiency. Helium is produced in a similar molar amount as tritium. The expected low solubility of helium in LLE has raised concerns due to a potential nucleation issue, with first experimental evidence provided around 30 years ago. Phenomena driven by helium state-of-solution in LLE, and generally liquid-metal/noble-gas systems, are complex to study from theoretical, numerical and experimental perspectives due to the extreme supersaturation.

To provide conclusive findings on the potential nucleation, a computational methodology was established which relies on *ab initio* Molecular Dynamics (MD) in its lower-level to validate the selected and derived classical MD potentials that are used in the upper-level of the methodology. SIESTA and LAMMPS codes are used for ab initio and classical MD simulations, respectively. The kinetics of helium bubble formation in liquid LLE are investigated using classical MD. The simulation protocol is modified after the non-equilibrium restrained classical MD techniques with gas-liquid-gas interfaces, with the main objectives being minimum interference with kinetics and providing a direct correspondence to the helium state-of-solution. Furthermore, a novel figure-of-merit based on two-step nucleation is used to monitor the kinetics, which are later analyzed with MFPT and committer-probability methods.

Liquid structure of lithium, lead and LLE results using a newly derived potential for liquid lithium and LLE are reported, which are in good agreement with the *ab initio* MD results. The dependence of helium bubble formation (cluster-size, growth-rate, energy-barrier) with pressure, temperature and supersaturation are presented, suggesting that helium solubility in liquid LLE increases with temperature and pressure. The cavity method is used to verify the estimated solubilities, and the formation mechanism is verified by analyzing the effect of surface tension and by comparing to theoretical hypotheses.

Keywords

Pb17Li, helium, bubble, nucleation, MD.

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PS1-25 ABSTRACT-4916
B. Blanket Technology

RELAP5/Mod3.3 MHD module Development and Validation: WCLL-TBM mock-up model

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Magnetohydrodynamic (MHD) phenomena are crucial for the design of magnetic-confinement fusion reactors and, specifically, for the design of the breeding blanket (BB) concepts that adopt liquid metals (LMs) as working fluids.

MHD effects are due to the interaction between the flowing metal, that is an electro-conductive material, with the magnetic field employed to confine the plasma in the reactor chamber. Electrical currents are induced in the liquid, and in turn, Lorentz forces are generated, thus altering the flow behaviour compared with the ordinary hydrodynamic case. For instance, MHD phenomena modify the velocity distribution and mass transport inside the ducts, enhance pressure losses, affect heat transfer mechanisms, etc. Estimating the impact of all those effects on the component performance is essential for an efficient project of a liquid metal breeding blanket.

Computational tools are vital to carry out fusion-relevant physical analysis but a dedicated MHD code able to simulate all the phenomena involved in a liquid metal blanket is still not available.

Models to predict both distributed and concentrated MHD pressure drop, derived by experimental and numerical works, have been already implemented in the thermal-hydraulic system code RELAP5/Mod3.3. The Verification and Validation procedure of the MHD module involves the comparison of the results obtained by the code with those of direct numerical simulation tools and data obtained by experimental works. As a validation exercise, RELAP5 is used to recreate the experimental results obtained for a mock-up of a Water Cooled Lithium Lead test blanket module at moderate to high magnetic field intensity (Ha = 500 - 2000). The novel MHD subroutines are proven reliable in the prediction of the pressure drop for the parameter range considered.

Keywords

MHD, pressure drop, liquid metal, system magneto-thermal-hydraulic code, breeding blanket.

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PS1-26 ABSTRACT-da35
B. Blanket Technology

Silicon Carbide Neutron Detectors for Tritium Production Assessment in Breeding Blankets

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Neutron detection is one crucial aspect for fusion plasma monitoring. The experience on JET and other tokamaks demonstrated how fusion power, ion temperature and fuel ion ratio can all be obtained through neutron counting and spectroscopy. Fast neutron detection is also going to be essential for the assessment of tritium production in future test blanket modules: this will have to be done by detectors located near or inside the breeding zone, which will feature very harsh conditions for detectors and electronics (>300°C temperature, neutron fluxes up to 10e14cm^{-2}s^{-1}, mechanical stress, etc.). In this context, Solid State Detector (SSD) based on Silicon Carbide (SiC) are very promising candidates for the task. Their small dimensions (mm), fast response (tens of ns) and relative insensitivity to gammas allow them to work in high fluxes conditions, providing a prompt and on-line monitoring of the neutron flux (which is not achievable through passive detectors like activation foils). Being able to measure single neutron interactions, they also allow a degree of spectroscopy. SiC SSDs were demonstrated more radiation hard and resistant to thermal noise than other SSD (e.g. silicon detector) and more stable than diamond SSD. There is also evidence for SiCs being suited for neutron detection at high temperatures (up to 500°C), which would allow for an operation inside the breeding zone without the additional cooling required for other SSDs.

This contribution will summarize the characterizations performed on various SiC prototypes in previous years, highlighting its stability, good energy resolution (2% at 14 MeV) and high-count rate capabilities (100s kHz). Different prototypes of SiC will be compared. The results of the upcoming high temperature neutron irradiation campaign, to be performed in spring 2023, will also be presented, possibly proving the functionality of the SiC as a neutron counter and spectrometer for breeding blanket environments.

Keywords

SiC detector, High Temperature, Fusion Plasma Monitoring, Breeding Blanket.

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PS1-27 ABSTRACT-7245
B. Blanket Technology

Design of the mock-up of breeding blanket aiming for tritium measurement

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The breeding blanket is a core component in future D-T fusion reactors. Accurately assessing the tritium breeding capability in the blanket is crucial for achieving the fuel self-sufficiency. In order to validate the design of candidate breeding blankets for fusion reactors, it is necessary to carry out experiments to measure the tritium production rate (TPR). A small mock-up breeding blanket is proposed to study the tritium production capability of different breeding materials under 14 MeV D–T neutron irradiation conditions with a focus on the TPR measurement for different materials. Li_4SiO_4 and Pb–Li alloy, two leading breeding candidates, are selected in the design of the mock-up model. For TPR measurement, Li_2CO_3 pellet is employed to perform the offline measurement while lithium glass detector serves as the online measurement. For the cubic mock-up model, the radial layers includes tungsten armor, first wall, two cooling plates, two beryllium layers and two breeding zones to represent the geometric configuration of a breeding blanket but in a smaller scale. In two breeding zones and beryllium layers, measurement channels will be placed in the center where Li_2CO_3 pellets and lithium glass detectors will be installed as appropriate. The spatial distribution of neutron flux of mock-up in the breeding zones and the tritium specific activity will be calculated to predict the tritium production capability of the two breeding materials respectively. The detailed neutronics performance in different materials will guide and to analyze two types of measurements that will be verified and validated in future experiment at next step.

Keywords

TPR measurements, Mock-up, Neutronics experiment, Breeding blanket.

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PS1-28 ABSTRACT-7981
B. Blanket Technology

Deuterium permeation and lithium-lead corrosion behaviors of ceramic-iron joint coating

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The main challenges in a fusion reactor blanket are to suppress tritium permeation through and corrosion of structural materials to ensure fuel efficiency and prevent environmental hazards. Our previous study found that ZrO2-Fe joint coatings satisfied the requirements for electrical insulation and shear stress applied by a liquid metal flow. This study focuses on deuterium permeation and Li-Pb corrosion behaviors of ceramic-Fe joint coatings for the development of tritium permeation barriers with corrosion protection.

ZrO2 and Fe2O3 coatings were fabricated on reduced activation ferric/martensitic steel F82H substrates by metal-organic decomposition. The coated samples were joined with Fe foils of 10–20 μm in thickness using a hot press machine at 13.5–25 MPa and 550 ^{0}C for 0.5 h. Gas-driven deuterium permeation measurements for the joint coating were conducted at the driving pressures of 10–80 kPa at 300–600 ºC. Li-Pb exposure tests under rotational flow were carried out at 550 ^{0}C for 500–2000 h. The flow velocity of Li-Pb relative to the sample was 0.064 m s–1. Surface and cross-sectional observations before and after exposure were performed by scanning electron microscopy and X-ray diffraction.

The joint coating decreased the permeation flux by a factor of 2000 in the first test at 400 ºC and 7000 at 550 ºC, indicating a further improvement in coating crystallinity and/or grain growth by joining without degradation. The joint coatings after flowing Li-Pb exposure showed the formation of a corrosion layer at the surface of the Fe layer, and its thickness slowly decreased with exposure time, while the thickness of the Fe layer did not decrease significantly. That indicates the corrosion layer contributed to mitigating the corrosion rate of the Fe layer. From these results, we conclude that the joint coatings are promising to satisfy the requirements for liquid Li-Pb blankets.

Keywords

Permeation, Coating, Joining, Corrosion.

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PS1-29 ABSTRACT-7dd9
B. Blanket Technology

3D MHD analysis on the COOL blanket for BEST

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Numerical simulations have been conducted to investigate magnetohydrodynamic (MHD) flows of liquid metal (LM) in a supercritical CO_2 Cooled Lithium-Lead (COOL) blanket module. This particular module is an advanced candidate for the Burning Plasma Experiment Superconducting Tokamak (BEST). The simulation employs MHD flow equations that account for buoyancy force and energy, and is solved numerically using a DNS-type finite-volume code named 'MHD-UCAS' on an extremely refined mesh comprising 466 × 10^6 computational cells. The strongest applied magnetic field is 5 T (Hartmann number Ha ~ 10^4), the flow rate of the PbLi in the poloidal ducts is 1.25kg/s (Reynolds number Re ~ 10^4), while the maximum volumetric heating is 40 MW/m^3 (Grashof number Gr ~ 10^11). The simulations encompass three distinct cases: (1) forced convection with electrically insulating walls, (2) mixed convection with electrically insulating walls, and (3) mixed convection with electrically conducting walls. For each of these cases, the computed MHD flows were analyzed for MHD pressure drop, flow distribution, temperature distribution, and unsteady flows. The findings of the study reveal important insights into the effectiveness of the flow channel insert (insulating walls) to reduce MHD pressure drop, flow balancing in the parallel channels, characteristic flow patterns, and the relevant temperature distribution.

Keywords

Magnetohydrodynamic, supercritical CO2, blanket module, the Burning Plasma Experiment Superconducting Tokamak.

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PS1-30 ABSTRACT-833a
B. Blanket Technology

Research on improving the system sensitivity for Hot Helium Leak Test

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Hot Helium Leak Test (HHLT) can realize more reliable leak tightness assessment than the conventional cold helium leak testing for components that run at elevated temperatures. To

ensure successful operation of the International Thermonuclear Experimental Reactor (ITER), the in-vessel components are required to ensure high vacuum sealing performance by HHLT.

The biggest challenge for the HHLT is that the background leak rate and sensitivity of the test system can hardly meet the requirements of high accuracy at high temperature. Through years of research, it is found the main factor affecting the rate and the sensitivity of HHLT facility is the hydrogen released from the vacuum-facing materials at high temperature. By adopting a series of measures, the background leakage rate was successfully reduced by one and a half order of magnitude to 1.17×10^{-10} Pa·m^3/s at 253℃, and the system sensitivity was improved by more than one order of magnitude to reach the order of 10^{-12} Pa·m^3/s, even down to 9.85×10^{-13} Pa·m^3/s at the temperature.

The developed hydrogen adsorption device and getter pump can effectively remove the hydrogen at high temperature, reduce the background leakage rate by more than one order of magnitude, and improve the system sensitivity by more than one order of magnitude, returning to the level at normal temperature.

This research solves the problem that it is difficult to improve the minimum detectable leakage rate of Hot Helium Leak Test at high temperatures, greatly reduce the difficulty of HHLT, greatly improve its applicability, and can be applied to fusion reactors to improve the overall sealing performance of vacuum vessel and reduce the leakage risk during reactor operation.

Keywords

Hot Helium Leak Test, ITER, Shield Block, Background leakage rate, System sensitivity, Hydrogen adsorption.

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PS1-31 ABSTRACT-88ad
B. Blanket Technology

System code simulation of DEMO WCLL Central Outboard Blanket equatorial cell operational transients

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Within the framework of EUROfusion FP8 and FP9 research activities, an important work was carried out to optimize the DEMO Breeding Blanket (BB) layout. The blanket concept selected to be investigated in this work is the Water-Cooled Lead-Lithium (WCLL). This design adopts reduced-activation ferritic/martensitic steel as structural material, liquid lead-lithium as breeder, neutron multiplier and tritium carrier, pressurized water as coolant and a tungsten layer to protect the First Wall (FW). The blanket is divided in sixteen toroidal sectors, which are split in outboard and inboard segments. Each segment is constituted of breeding cells, stacked in the poloidal direction. Their layout differs according to the poloidal position and the segment they belong. The reference cell design for this activity is the WCLL2018.v0.6_B, associated with the component located at the equatorial plane of the Central Outboard Blanket (COB).

During last years, the Department of Astronautical, Electrical and Energy Engineering (DIAEE) of Sapienza University of Rome, in collaboration with ENEA, has supported the optimization of the WCLL BB layout. One of the key design issue is evaluating the component thermal-hydraulic performances during operational and accidental conditions. System codes are the reference numerical tools to be used for transient analysis. For this, the aim of this work is developing a RELAP5/Mod3.3 model of the COB equatorial cell suitable to investigate the component behavior during selected operational scenarios. To assess the system code modelling capabilities, due to the lack of experimental data, a benchmark activity was performed by comparing the RELAP5 outcomes with the results obtained with a Computational Fluid Dynamic (CFD) tool. The comparison demonstrates the capability of the system code to properly simulate the cell thermal inertia, as well as to qualitatively reproduce the fluid and structural thermal fields.

Keywords

Water-Cooled Lead-Lithium, RELAP5, CFD benchmarking activity, thermal-hydraulics, transient analysis.

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PS1-32 ABSTRACT-000e
B. Blanket Technology

Dedicated Thermal, Thermal-hydraulic and Thermal-mechanical Analyses in support of DEMO WCLL BB design

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Within the framework of the EUROfusion Work Package Breeding Blanket (WPBB), R&D activities are carried out since 2021 on the conceptual design of the DEMO Water-Cooled Lithium Lead Breeding Blanket (WCLL BB).

The present study focuses on the evolution of the design of the inboard equatorial slice model of the WCLL BB and the associated steady-state thermal-hydraulic, thermal and thermo-mechanical studies.

Full CFD thermal-hydraulic calculations have been carried out on different designs. They aim at establishing both the thermal field in the component and the required cooling flows such that design inlet and outlet temperature values, equal to 295°C and 328°C respectively, are met. They additionally help locating hotspots in Eurofer structure (where temperature locally exceeds the criterion of 550°C) and identifying further improvements. Furthermore, several areas where water could experience local nucleate or sub-cooled boiling point have been highlighted.

Due to the large calculation time of full CFD studies, a local simplified thermal model including heat exchange between fluid and structures was set up and validated (temperature differences between CFD and thermal model do not exceed 2%) in order to perform parametrical studies for design optimization. Simulation results reveal significant improvement: temperature is dropped below the 550°C criterion. However, the potential boiling issue is still observed.

Concerning the mechanical analysis, RCC-MRx Type P damages rules are applied to the previous simplified sub-cell model: irradiation is considered to be always significant, whereas creep can be negligible or significant depending on the area (local temperature). Results show that modified geometries lead to significant improvements, especially concerning criteria which take into account thermal stress, resulting of the reduction of thermal gradients.

Finally, significant improvements of the design have been obtained, and the way to go further is proposed.

Keywords

Water cooled lithium lead breeding blanket, cooling piping layout, CFD thermal-hydraulic, mechanical analysis, RCC-MRx, mechanical design.

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PS1-33 ABSTRACT-c630
B. Blanket Technology

Numerical simulation of buoyant flow in a vertical channel for plasma facing component

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It is important for cooling plasma facing component to investigate characteristics of thermofluid used in the high-temperature plasma of fusion reactors. The working fluid as a pressurized water has a significant role in the design of advanced fusion reactors. In Test Blanket Module, pressurization water of 15MPa, 600K is adopted in coolant [1]. In addition, with the blanket which it carries away liquid metal in a perpendicular rectangular duct designed for a nuclear fusion reactor and cools, the heat transfer is ruled over under the influence of a nature convection (buoyancy) to appear depending on the direction (or it is an upward or is downward) of the flow for the direction where a magnetic field and the gravity in the equivalence plane suffer from [2]. In the case of Aiding Flow, an opposite direction, the turbulence close to the wall is called Opposing Flow each when buoyancy appears in the mainstream direction and the same direction [3]. This suggests that heat transfer increases in the characteristic parameter in the range of the identification of the design specifications of the blanket, and a big temperature change appears for the downward flow. This may be caused by a secondary large-scale vortex formed in a flow. Specifically, in the case of a condition to be beyond a condition of Grb/Reb2> 10-3, a buoyancy effect occurs. When bulk velocity decreases by the operation of the real reactor, this condition may apply to it. In the present study to investigate the condition, we performed a Direct numerical simulation (DNS) of a vertical channel flow with a buoyancy effect. Reynolds number is 1100 based on the channel half width and friction velocity. In this simulation, the values of Pr number was 0.87 and Grashof numbers was 1.0 × 108. The numbers of mesh points used for 1024 × 1024 × 768 in the x−, y−, and z-directions, respectively. As a result, the turbulent quantities such as the mean flow, turbulent stresses, turbulent kinetic energy budget, and the temperature statistics became asymmetric profiles. The tendency was enhanced with the buoyancy effect near wall region.
Keywords: DNS, buoyant flow, coolant wáter flow, Plasma facing component

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Keywords

DNS, buoyant flow, coolant water flow, Plasma facing component.

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PS1-34 ABSTRACT-92f5
B. Blanket Technology

Modelling transport of dust particles in the Helium-Cooled Pebble Bed breeding blanket concept

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

Keywords

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

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PS1-35 ABSTRACT-0b31
B. Blanket Technology

Experimental investigation of MHD flows in a WCLL TBM mock-up

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The current European development strategy for DEMO breeding blankets has recently been revised in favor of a more synergetic approach with the EU ITER TBM program [1]. The new strategy comes with a renewed focus on the water-cooled lead-lithium (WCLL) concept that has been identified as one of the two reference candidates for DEMO and selected to be tested in ITER.

As for all other liquid metal breeding blankets, constraints associated with the magnetohydrodynamic (MHD) effects can jeopardize the feasibility of a WCLL blanket. In particular, the additional MHD pressure drop caused by the motion of the liquid metal in the plasma-confining magnetic field is often viewed as one of the most critical issues. Although low PbLi velocity in the breeder units of WCLL blankets may not raise major concerns, manifolds are expected to generate most of MHD pressure drop and to significantly influence flow partitioning within the breeding modules [2]. Therefore, a careful evaluation remains needed.

Since the complexity of the WCLL geometry still represents a challenging task to examine 3D MHD effects occurring in the whole TBM by means of numerical simulations, experiments happen to be essential to investigate pressure drop and flow distribution at the scale of a module. In the present work, MHD flows in a WCLL TBM mock-up were studied in the MEKKA laboratory at KIT. In order to fit into the gap of the dipole magnet available, a scaled-down mock-up was built according to the latest WCLL TBM design [3]. Experiments were carried out for several Reynolds and Hartmann numbers for which flow properties were measured at relevant locations.

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Keywords

Magnetohydrodynamics (MHD), Water-Cooled Lead-Lithium (WCLL), Test Blanket Module (TBM).

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PS1-36 ABSTRACT-1690
B. Blanket Technology

Progress on the Neutronics Design and Analysis for CN HCCB TBS in PD pase

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Test Blanket Modules (TBM) will be tested in ITER to validate the key technologies of future DEMO blanket. The helium cooled ceramic breeder Test Blanket Module (HCCB TBM) concept has been selected as the primary option of Chinese TBM program. The Preliminary Design (PD) of CN HCCB TBM and its ancillary systems (together called HCCB TBS) has been completed. For HCCB TBS, the neutronics design and analysis are required as inputs for the performance and safety design, such as tritium production and extraction, heat generation, activation of structural materials, shielding and radiation safety. Based on the current design of HCCB TBS and the common components design of Port #18 in PD phase, a three-dimensional neutronics analysis model for HCCB TBS has been developed, which integrates the neutronics models of the TBM Frame, HCCB TBS, Pipe Forest, Bio-shield and Port Cell into the latest ITER C-model (the analysis model of the regular sector of ITER meant for simulation of the radiation transport). By using this neutronics analysis model for HCCB TBS, preliminary neutronics design and analysis for HCCB TBS in PD phase has been performed. The key nuclear parameters of HCCB TBS, such as Tritium Production Rate (TPR), neutron flux, nuclear heating, radiation damage, gas production rate, activation data and Shut Down Dose Rate (SDDR) in the maintenance area, have been obtained and analyzed. These neutronics analysis results are important input parameters for thermo-hydraulics analysis, safety analysis, tritium system design, helium system design, rad-waste management, maintenance strategy, etc. The results can be also used as the basis and support for the HCCB TBS preliminary design review.

Keywords

HCCB TBS, Neutronics design and analysis, Neutronics analysis model, Nuclear parameters.

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PS1-37 ABSTRACT-1a82
B. Blanket Technology

Key Technologies of Manufacturing and Testing for The ITER Blanket Shield Block

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The main function of the ITER blanket system, which is mainly consist with the First Wall (FW) panels and Shield Blocks (SB), is to contribute absorbing radiation and particle heat fluxes from the plasma and neutral beam, and providing neutron shielding to the vacuum vessel and ex-vessel components. And the SB is one of the main in-vessel components manufactured with SS316L(N)-IG to play a major role in neutron shielding, to support the FW panels and to supply the FW panel with cooling water. Each SB is attached to the Vacuum Vessel (VV) through a mechanical attachment system of flexible supports and keys, while electrical straps provide dedicated electrical connection to the VV. Cooling channels are located inside of the SB to remove the neutron heat deposition while keeping the structure temperature to acceptable levels. The water coolant is routed first through the plasma-facing first wall panel which is attached to the SB and then through the SB. A number of deep slits are introduced into the SB to disrupt eddy current loops and reduce the Electro-Magnetic (EM) loads on the support system and VV. Water headers are machined on the side of the module with 7 ~ 10 mm welded cover plates. A Procurement Arrangements (PAs) for the manufacturing and testing of the SBs have been signed with Korea (50%) and China (50%).

As part of the PA in Korean Domestic Agency (KODA), Process Qualification (PQ) to ensure the manufacturing and testing ability of the SB was completed successfully through manufacture of a Full Scale Prototype (FSP) as an essential step before starting series production. This paper reports key technologies, such as machining, welding, NDE and hot helium leak testing, of manufacturing and testing for the ITER SB series production in Korea toward successful completion of the missions.

Keywords

ITER, Blanket Shield Block, Procurement Arrangement, KODA.

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PS1-38 ABSTRACT-21b4
B. Blanket Technology

Instrumentation integration in the European Test Blanket Modules

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Amongst the four different Test Blanket Modules (TBM) that shall be tested in ITER, the European domestic agency (Fusion For Energy) participates in two concepts: the Water-Cooled Lithium Lead (WCLL) and the Helium-Cooled Ceramic Pebbles (HCCP). A progressive set of activities including implementation and integration of instrumentation has been conducted from preliminary design stage. The current work is mainly focused on the thermal and mechanical instrumentation, although other sensors (such as electromagnetic, neutronic or electrical potential) are also considered. The integration of instrumentation in the TBMs is highly challenging because of four reasons: (1) a harsh environment (temperature, radiation, EM fields); (2) the rather congested geometry with little extra space for sensors; (3) the need to cross the high-quality vacuum (by means of hermetic feedthroughs) and (4) the need to limit the impact on the functional materials or structural integrity of the TBM. Consequently, the number and position of the sensors need to be carefully optimized. An important advance of the current work is the use of reconstruction algorithms to provide an estimation of the reconstruction ability of the different sensor sets. These algorithms consist of a series of mathematical tools which combine model results (finite element analysis) with the sensor measurements to infer different variables (reactions, temperatures, displacements, etc.) at positions where no measurement is conducted. The reconstruction algorithms provide an invaluable tool to decide which set of instrumentation maximizes the global information conveyed to the operator. The role of these algorithms may be even more important during the operation phase to infer the real behaviour of the system from the discrete set of sensor measurements. Apart from the use of these algorithms, the final decision on sensor location have been conducted from a holistic point of view considering the different perspectives and integration challenges.

Keywords

Instrumentation, reconstruction algorithms, design integration, thermocouples, measurements, Test Blanket Modules, ITER.

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PS1-39 ABSTRACT-2834

B. Blanket Technology

Multiphysics Modeling and Creep Analysis of the DCLL Blanket of the Fusion Nuclear Science Facility

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Creep is an inelastic time-dependent deformation that occurs when a material is subjected to stress typically much less than the yield strength at sufficiently high temperatures. A thermomechanics model that relies on multiphysics coupling of heat transfer and solid mechanics modules is developed to determine the structural integrity of the recently designed Dual cooled lead lithium (DCLL) blanket for Fusion nuclear science facility under steady state loads. The time and temperature dependent allowable stress intensities of F82H steel are evaluated and used in the high temperature design rules of elastic ITER structural design criteria for in-vessels components (SDC-IC) to evaluate creep damage in the blanket having design life up to 2 years at 70% fusion reactor availability. Creep rate coefficients and stress exponents are derived for a visco-plastic model based on Norton law. The results of the analysis of the blanket using first wall/blanket temperature distributions and assuming higher temperature of 600oC at the first wall are presented.

Keywords

Multiphysics, Creep, F82H steel, ITER SDC-IC, Viscoplastic, Blanket.

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PS1-41 ABSTRACT-2fb5
B. Blanket Technology

Experimental measurement of hydrogen and deuterium solubility in LiPb using absorption and desorption techniques

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The accurate quantification of tritium inventory in the liquid breeder of fusion reactors blankets, tritium permeation into the coolant and the design of tritium extraction systems are strongly dependent on tritium solubility into the lithium-lead eutectic alloy (LiPb, 15.7 at. % Li). Over the past four decades, several measurements of hydrogen isotopologues solubility into the LiPb have been performed by employing mainly absorption and desorption techniques. However, major discrepancies on the Sieverts' constant value are found in literature, and absorption results generally show higher values with respect to desorption experiments. At ENEA Brasimone research centre, a new experimental laboratory-scale device, named Hyper-Quarch II (Hydrogen Permeation Quartz Chamber), was developed based on the experience gained in the past experimental campaigns. This device, characterised by an upgraded test section in quartz and new instrumentation equipment, was used to perform the measurements of the Sieverts' constant using both absorption and desorption techniques. The experimental campaign was carried out in the temperature range 300-450 °C with hydrogen and deuterium partial pressure in the range 10-100 Pa. Within this paper, the main outcomes are reported, showing very similar results from both the measurement techniques. To substantiate the experimental results, a thorough statistical analysis was also performed.

Keywords

Sieverts' constant, tritium, lithium-lead, breeding blanket.

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PS1-42 ABSTRACT-37c0
B. Blanket Technology

Experimental determination of the solubility of hydrogen in a quality certified eutectic PbLi alloy

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The efficient operation of the liquid breeding blankets relies on the correct definition of the transport parameters of hydrogen isotopes in the eutectic PbLi alloy. This is the case of the Dual Coolant Lithium Lead (DCLL) blanket or the Water Cooled Lithium Lead (WCLL), where the solubility and diffusivity values will determine the magnitude and the kinetics of the induced tritium flux.

However, the definition of these transport parameters implies several technical difficulties that lead to uncertainties that can entail a severe disadvantage for the development of these breeding blankets. In fact, there exists a wide band of even three orders of magnitude in the literature for the experimental results of the solubility of hydrogen in the eutectic PbLi alloy, which is unacceptable from the designing point of view. In addition to the unavoidable uncertainties related to the experimental methods, the exact content of lithium and the impurities that the PbLi alloy may contain, seems to be one of the key issues that needs to be faced.

This work shows the preliminary results obtained in a new measurement campaign carried out with samples of PbLi with a certified quality in terms of impurities and lithium content. These samples have been tested in Absorption-Desorption facility of the University of the Basque Country (UPV/EHU) that has been recently repaired and upgraded in the framework of the EUROfusion program in close collaboration with CIEMAT.

Keywords

Lead lithium, solubility, diffusivity, hydrogen, tritium.

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PS1-43 ABSTRACT-3abc
B. Blanket Technology

Hydrogen Extraction from a Helium Purge Gas. Comparison of ZrCo/ZAO Getter Material

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From the known getter materials proposed for the storage, supply, and recovery of hydrogen isotopes the interalloy ZrCo has been selected as reference material. However, because of disproportionation, ZrCo loses its ability to reversibly absorb, or desorb tritium, after prolonged thermal cycling. Therefore, several other interalloys such as ZAO have been suggested as potential candidate materials to replace ZrCo. The new materials need to be tested and compared to ZrCo, before their use as replacement to ZrCo. In this report, both materials ZrCo and ZAO are submitted to the exactly the same experimental conditions i.e. to several successive loading-deloading cycles (LDCs) and their performance in terms of absorption/desorption of hydrogen is compared. The experiments have been carried out at Tritium Laboratory Karlsruhe (TLK), using the HYdrogen-DEuterium (HYDE) loop.

Five consecutive LDCs have been carried out using 48 g of ZrCo, or ZAO. During the loading phase the getter-bed was purged with a constant flow of He and H_2 with a H_2/He ratio of approximately 0.003. During this series of experiments the experimental parameters such as: the getter bed temperature, pressure drop and He and Hydrogen flow rates, have being recorded and the results analysed.

This report describes a comparison of the results obtained for both interalloys. It is remarkable to see that although the consecutive LDCs are very similar for ZrCo, this is not the case for ZAO. A constant feature of ZAO is that the first loading/deloading cycle is very different from the subsequent LDCs, since after the first LDC the material does not release the entire amount of hydrogen absorbed during the 1st cycle. This behaviour is attributed to the formation of several hydrogenated species generated during the loading phase, with some of them having different activation energies for the hydrogen release and thus different (higher) releasing temperatures.

Keywords

Zrco-ZAO, Tritium Extraction, Getter-beds.

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PS1-44 ABSTRACT-5a6c
B. Blanket Technology

Thermal-hydraulic assessment of the inlets and outlets of the Steam Generator mock-up for the EU DEMO WCLL

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The EU DEMO is planned to be the first tokamak fusion reactor to deliver electricity in the EU; therefore, it will feature a balance-of-plant (BoP) to convert the power generated by the fusion reactions into electricity. The design of the Primary Heat Transfer System (PHTS) for the Water-Cooled Lithium-Lead (WCLL) Breeding Blanket can take advantage of the large experience from light water fission reactors, but new challenges arise due to the intermittent operation (pulse-dwell-pulse regime) of the EU DEMO. Therefore, the STEAM facility will be operated by ENEA in order to validate the design of WCLL BoP. In particular, a Once-Through Steam Generator (OTSG) mock-up is being design for its test in STEAM.

In the present work, the water distribution in the inlet and outlet regions of the OTSG mock-up in steady-state conditions is assessed, both on the primary and secondary sides. A Computational Fluid Dynamics (CFD) model is developed with the commercial Star-CCM+ code to investigate the coolant behaviour in the four regions. In order to build a self-consistent model taking into account the two inlets and the two outlets, a co-simulation environment is employed, dramatically reducing the computational cost with respect to a full model of the OTSG mock-up.

Acknowledgement

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Keywords

EU DEMO, WCLL, Steam generator, thermal-hydraulics.

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PS1-45 ABSTRACT-4146
B. Blanket Technology

Mechanical testing of ceramic materials in liquid Pb-16Li at elevated temperaturas

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Flow channel inserts (FCI) are functional components embedded along the Pb-16Li channels inside the Dual Coolant Lithium Lead (DCLL) breeding blanket modules. These components are designed to minimize the magnetohydrodynamic pressure drop and serve also as a thermal and tritium permeation barrier. A simplified design of FCI component consist of two layers made of steel and ceramic. In this arrangement, the ceramic is directly exposed to contact with the Pb-16Li eutectic at elevated temperature. The exposure of these components to demanding operating conditions of the DCLL breeding blanket gives rise to mechanical stress formation caused by elevated temperatures and differential deformations stimulated by interface between ceramic and the steel layer. Therefore, the assessment of potential influence on mechanical performance degradation caused by ceramic corrosion during the exposure in Pb-16Li at elevated temperatures is essential.

Mechanical behaviour of the ceramic samples was investigated by means of three-point bending test in the PAMETES facility. Ceramic samples were subjected to three-point flexural test at constant displacement rate at room temperature, at temperature of 650 °C in argon and at temperature 650 °C in liquid Pb-16Li. Scanning electron microscopy was adopted to investigate the fracture surfaces of tested samples. Temperature related mechanical response of the materials is discussed with respect to fracture morphology of the tested samples.

Keywords

Lead lithium, DCLL breeding blanket, Three-point flexural test, Flow channel inserts.

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PS1-45b ABSTRACT-d837
B. Blanket Technology

Major improvements in the WCLL TBM design towards next review gates

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The European domestic agency (Fusion for Energy) is developing the WCLL-TBS (Water Cooled Lithium Lead Test Blanket System) concept for its implementation and testing in ITER. The system is based on the use of the eutectic Pb–15.7Li enriched at 90 % in 6Li as tritium breeder and tritium carrier. Water is used as coolant with current target nominal inlet/outlet temperatures of 295/328 °C and 15.5 MPa pressure (similar to PWR conditions). The system successfully passed the Conceptual Design Review (CDR) gate in September 2020 and is facing now a significant design evolution towards the next review gates. The design of the TBM (Test Blanket Module), the ITER in-vessel component where tritium generation will take place and whose structural material is EUROFER 97, has been significantly improved in the last years. These design improvements have focused on three major areas: (i) overcoming some weaknesses of the previous design in the rear manifold area associated to the stiffening rod concept, (ii) strengthening the link with manufacturing developments so that a consistent approach from the design and manufacturing points of view is promoted for the TBM assembly and, especially, the most critical welded connections between TBM parts, and (iii) optimising the amount of EUROFER 97 in the TBM box, so that the design converges to the limits imposed by the ITER project requirements. The work presented in this paper describes the latest design evolutions achieved in the WCLL TBM design with respect to that presented at CDR and, in particular, the major re-design of the TBM in the manifold area based on the ‘crossing vertical’ stiffening plate concept proposed and developed by the ESTEYCO Mechanics design team in collaboration with F4E.

Keywords

Test Blanket Modules, mechanical design, ITER.

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PS1-46 ABSTRACT-a9ad
C. Fuel Cycle and Tritium Processing

Experimental investigation of tritium release behavior from neutron irradiated LiAlO2 with Zr for tritium production in high-temperature gas-cooled reactor

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An important issue in a deuterium-tritium fusion reactoris to secure tritium for starting up. The amount of tritium required for the steady operation is estimated to be several kilograms. Since the amount of naturally occurring tritium is extremely little, tritium is proposed to be produced bythe 6Li(n,α)T reaction in a high-temperature gas-cooled reactor. Currently, LiAlO2 powder including Zr is planned to be used for the tritium production. Although the understanding of tritium release property from LiAlO2 is essential from the viewpoint of safety and productionefficiency, there are few reports on it. In this study, the tritium release property of neutron irradiated LiAlO2 were investigated, and the issues on tritium production in actual HTGR were discussed.

The samples of LiAlO2 powder and LiAlO2 powdermixed with Zr powder sealed in a quartz tube in vacuum were neutron irradiated in Kyoto University Research Reactor. The samples were once heated up to 900 °C in sealed state to simulate a high temperature condition. After cooled down to room temperature, the samples were heated up to 900 °C in the Ar flow. The sample without Zr showed that more than 90 % of tritium was released in the form of HTO in the temperature range of 100 ~ 900 °C. The release peak was appeared at low temperatures. On the other hand, for the sample containing Zr, most of tritium was released in the form of HT and the significant amount of HTO was not detected. This result indicates that most of tritium generated in LiAlO2 was released in the gas phase in the sealed quartz tube and absorbed in Zr during pre-heated to 900 oC. It was found that tritium can be absorbed in Zr at 900 oC even if tritium is released as HTO from LiAlO2.

Keywords

Tritium production, tritium behavior, LiAlO2.

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PS1-47 ABSTRACT-3656

C. Fuel Cycle and Tritium Processing

Simulation and concept design of continuous mercury-driven vacuum pumps for EU-DEMO

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As a means of reducing the tritium inventory of future fusion power plants, continuous mercury-driven vacuum pumps are currently under active research and development. The mercury pumping
train consists of linear diffusion pumps (LDPs), booster pumps (BPs) relying on the ejector principle and liquid ring pumps (LRPs) to achieve the necessary compression of about max. 8 orders of magnitude (approx. 10^{-3} to 10^{5} Pa) of the torus exhaust gases.
In the present work we focus on the simulation and concept design of the mercury vapour-driven pumping stages (i.e. LDPs and BPs). The simulation of these pumps has to respect the wide operating pressure range, which spans all rarefaction regimes (free molecular, transitional and continuum flow). For this reason two different simulation methodologies are applied: On the one hand the direct simulation Monte Carlo (DSMC) method is used to simulate the LDPs. On the other
hand a Navier-Stokes based finite volume method (FVM) is used to simulate the booster pumps. The simulation results are used to extract design guidelines for both pump types. Based on these design guidelines and the expected operating conditions during burn and dwell phases in EU-DEMO, preliminary concept designs of both pump types are proposed.

Keywords

Fuel cycle, torus exhaust pumping, linear diffusion pump, booster pump, ejector, mercury.

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PS1-48 ABSTRACT-6d6f
C. Fuel Cycle and Tritium Processing

Hydrogen retention investigation in ITER monoblock using Lattice Boltzmann method

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A new approach based on the Lattice Boltzmann method (LBM) to simulate the hydrogen isotopes transport behavior in monoblocks of ITER has been investigated. This code can deal with physical models such as heat transfer, ion injection, defect trapping, and surface release in multi-materials within two dimensions. Firstly, the thermal filed and the tritium concentration filed simulated by LBM are compared to that of the finite element method in 1D model, and results obtained from these two methods are in good agreement in the aspects of data trends and extreme values, demonstrating the reliability and accuracy of the LBM.

Secondly, the tritium transport behavior in monoblock composed by W-Cu-Cu alloy has been studied. 100 plasma cycles lasting 4.2×10^5 s is simulated to reproduce the operation condition of a fusion reactor, and each operation cycle includes four parts, which are the ramps up and plateau of the plasma current, following the ramp down and the rest part. The following conclusions are drawn from the simulation. The transport of tritium against the concentration gradient at the interface of different is observed which demonstrates the continuity of the chemical potential. The tritium inventory in each monoblock is 1.25×10^{-6} g, and the total inventory is 35 mg after 100 cycles with the assumption that there are 27,777 wet monoblocks in ITER, which means that the monoblock meets the safety requirements. In addition, we calculated the release rate of tritium discharged into the coolant through the monoblock, which is about 6.9×10^{-8} g/s in the 100th cycle.

Keywords

LBM, tritium transport, ITER, monoblock.

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PS1-49 ABSTRACT-c1bc
C. Fuel Cycle and Tritium Processing

Tritium-compatible monitoring of ozone production and depletion rates for the decontamination of tritiated surfaces

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At Tritium Laboratory Karlsruhe (TLK), key tritium technologies required for the operation of the KATRIN experiment, as well as for fusion research, are developed. A major aspect with respect to the maintenance and decommissioning of tritium experiments is to develop dedicated decontamination strategies. These are also essential for analytic tools in order to reduce tritium memory effects. In the case of KATRIN, the formation of surface-bound tritium, especially on the "Rear Wall" of its windowless gaseous tritium source, leads to a systematic uncertainty in the high precision measurement of the beta-spectrum. In order to reduce this, it is important to decontaminate the Rear Wall surface at regular invervals by exposure to ozone. This method is mentioned in the literature multiple times, however, without quantitative investigations. To obtain quantitative data and to optimise the decontamination procedure of the Rear Wall, the UV Ozone (UVO) experiment was set up. With this dedicated experiment operating in a carefully controlled environment, we will be able to obtain quantitative data on the production and life time of ozone, as well as on chemical processes between contaminated surfaces, ozone, and flushing gases (N2, H2, D2, O2, H2O, HDO and others). This contribution gives an overview of the UVO experiment and shows results of the first commissioning and calibration measurements using dedicated spectroscopic tools with respect to commercial non-tritium compatible ozone monitors and theoretical absorption line strengths. First quantitative results towards a pressure dependence of the production and depletion of ozone in synthetic air will be presented.

Keywords

Tritium, decontamination, ozone, UV light, TLK, monitoring.

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PS1-50 ABSTRACT-dca1
C. Fuel Cycle and Tritium Processing

Effects of Impurity Gases on Hydrogen Permeation in Pd-Ag Pipe

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In the fuel processing system of a fusion DEMO reactor, the hydrogen isotope permeation method using Pd alloy membranes is used under various conditions. In the plasma exhaust gas processing system, after extracting most of hydrogen isotopes through Pd alloy membranes in the first step, further hydrogen isotopes are extracted from impurity gases such as hydrocarbons and water vapor using a catalytic reactor and Pd alloy membranes. In the extraction system of tritium bred in the blanket, tritium is extracted from a He purge gas including water vapor released from Li ceramics by Pd alloy membranes. Therefore, it is important to quantify the highly reliable hydrogen permeation rate in the presence of impurity gases such as hydrocarbons and water vapor. In this work, the effect of impurity gases on the hydrogen permeation rate was investigated by hydrogen permeation experiments using Pd-Ag pipes.

Pd-Ag pipes with an outer diameter of 5 mm, thickness of 0.1 mm and length of 80 mm were connected to stainless steel pipes and inserted in a quartz pipe. A H_2/He including impurity gas was introduced to the outside of the Pd-Ag pipe and a He gas was introduced to the inside. A gas chromatograph and QMS were used for evaluating hydrogen permeation rate through the pipe.

At 200℃, 300℃ and 400℃, no effect of methane on hydrogen permeation rate was observed. However, when the sample gases containing ethane and ethylene, or acetylene were introduced at 400℃, the hydrogen permeation rate decreased with time. After that, when a H_2/He gas was introduced, the permeation rate slowly recovered to the initial value. Hydrocarbons were decomposed on the Pd-Ag surface, and hydrogen permeation was considered to be inhibited by deposited carbon. The effect of water vapor will also be reported in detail.

Keywords

Hydrogen, Pd-Ag, Permeation, Impurity.

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PS1-51 ABSTRACT-088a
C. Fuel Cycle and Tritium Processing

Development of the design of a stackable Permeator-Against-Vacuum mock-Up for tritium extraction from liquid PbLi at low speed

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The Permeation Against Vacuum (PAV) is one of the candidate technologies currently under consideration for the EU DEMO for the tritium extraction in those lead-lithium based breeding blankets. The Tritium diffuses through a permeable membrane into a vacuum and storage system. The development from TRL1-2 to TRL2-3 of a PAV design, dimensioned for being tested in the CIEMAT CLIPPER facility, is presented together with FEM thermal stress analyses and a test plan for its functionality and manufacturability.

Following previous works, the PAV is designed as a stack of bolted steel frames separated by vanadium sheets forming volumes which alternate between liquid metal and vacuum. PAV fabrication involves a number of technological challenges due to the operating requirements (temperatures of 550°C and high PbLi velocities of up to 1 m/s). The main challenge identified in this phase is the design of membranes with high permeability to hydrogen isotopes that can withstand the operating requirements.

Vanadium sheets with thickness of 1mm are selected as the most suitable membranes due to their good permeation and mechanical properties at 550°C. Those sheets will be e-beam welded to stainless-steel frames and supported by perforated aluminium honeycombs to avoid the sheet's deformation due to the pressure difference between both sides of the membrane. A four liquid channel design (each channel with a membrane of 100x5x1 cm) has been modelled, obtaining an extraction efficiency above 30% and a pressure loss lower than 1 bar. The dimensional trade-off of the design has also been analysed.

The test campaign will focus on steel-vanadium e-beam trials, leak tests and vacuum gas conductance tests of the honeycomb. The leak tightness of the bolted Helicoflex® seal between the stackable structures will be tested through a torque test were the deformation of the seals between stacked frames will be measured.

Keywords

DEMO breeding blanket, Lithium-Lead, Permeator Against Vacuum (PAV).

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PS1-52 ABSTRACT-129d
C. Fuel Cycle and Tritium Processing

Investigations on Hydrogen Isotopes Separation Factor Employing Palladium-based Solid Metallic Membranes

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Hydrogen isotopes separation is an actuality interest domain, considering that implementation of greenhouse-free technologies represents a major challenge, due to the climate changes that took place in the recent past years. Heavy hydrogen isotopes (deuterium and tritium) are used for energy production in fusion rectors, therefore play an important role in this transition towards a zero-emissions economy.

In present there are several methods for hydrogen isotopes separation (in gas form), the most important being cryogenic distillation, thermal diffusion, fractional adsorption and gas chromatography, however, these have a series of drawbacks, namely high complexity, high energy consumption and associated costs.

Taking into account these disadvantages, another separation method, less studied until now, is the one based on solid metallic membranes, due to their advantages like low energy consumption and reduced complexity. This method uses the difference between some hydrogen isotopes properties, namely solubility, diffusivity and permeability, implicitly.

This work envisages the integration of an isotopic separation module, based on membrane permeation, on the exhaust gas line from the current experimental rigs employed at ICSI, in order to recover and store the hydrogen isotopes. The aim of this work is to study the separation factor variation with temperature in the domain from 293 to 573 K, in order to maximize the separation process efficiency.

Keywords

Isotopic separation, hydrogen isotopes, solid metallic membranes, deuterium, tritium.

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PS1-53 ABSTRACT-1992

C. Fuel Cycle and Tritium Processing

Liquid metal coolant purification assessment

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The analysis of the by-product formation due to the irradiation under fusion conditions of several liquid metals has been conducted for the following liquid metal coolants – Li, LiPb, Ga and Sn. These liquid metals were chosen due to their tritium breeding properties (Li), high thermal conductivity (LiPb), extremely low vapour pressure (Sn) and its good nuclear properties (Ga).

Firstly, a dedicated methodology for the neutronic analysis was developed based on which the assessment of the irradiation produced by-products and of by-products having impact on corrosion has been performed. Secondly existing chemical reactions with tritium were investigated and finally a detailed literature review of the available purification technologies was conducted.

Due to the presence of impurities in each coolant, after representative irradiation time for fusion reactor conditions, a large quantity of by-products was generated by nuclear reactions and decay, among others.

Regarding the activity analysis, it should be noted that there is a difference in the contribution to the activity depending on the spectra that is used. Two different neutron energy spectra have been used, one corresponds to the first wall (FW) while the other one corresponds to the breeding blanket (BB) and the divertor (DV). As expected, the coolant activity is higher for the FW in all of the cases, since the neutron flux is approximately one order of magnitude higher than for the BB.

The present study is a step forward to understand the impact of impurities in liquid metal under fusion conditions from the operation and waste points of view. These scoping calculations were performed with the goal to screen for the most significant by-product, identifying them for the future analysis. This analysis could help discard possible calculation artefacts, resonances, effects of cross section and different libraries, etc.

Keywords

Liquid metal, Li, LiPb, Sn, Ga, impurities, activity, irradiation, neutronic analysis, tritium, coolant purification.

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PS1-54 ABSTRACT-1b7f
C. Fuel Cycle and Tritium Processing

ASPECTS CONCERNING THE MAUNUFACTURE OF IMPROVED CONTACT ELEMENT FOR WATER DETRITIATION SYSTEM

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For tritium separation, produced in ITER fusion reactors and/or from heavy water used as moderator in CANDU fission reactors an effective and efficient water detritiation system it is requested. One of the most promising technology is Liquid Phase Catalytic Exchange (LPCE) coupled with cryogenic distillation of hydrogen, which simultaneously take place in two steps:

H_2 (gas) + HTO (vapor) ßà HT (gas) +H_2O (vapor) (1)

H_2O (vapor) + HTO (liquid) ßà HTO (vapor) + H_2O (liquid) (2)

H_2 (gas) +HTO (liquid) ßà HT (gas) +H_2O (liquid) (3)

For the both steps, *the contact element* (hydrophobic catalyst for step1 and hydrophilic packing for step 2) between phases plays the key role. Th mixture in various ratio between catalyst and hydrophilic packing is so called *"mixed catalytic packing"* and conventional packing, can be arranged in random or ordered (alternated beds) structure.

This paper it is focused on the last achievements and improvements of previously conventional mixed catalytic packing, irrespective a more compact mixed catalytic packing with new inner geometry and structure. Instead of mixed catalytic packing with an alternated structure, a more compacted structure (two in one) having the the catalyst with compatible sizes inserted on inner channels of conventional hydrophilic packing has been manufactured. Recent improvements of the both contact elements(hydrophobic catalyst and hydrophilic packing) as well as the ensemble – mixed catalytic packing are presented. To increase the wettability of hydrophilic packing a special chemical treatment, followed by a thermal treatment in controlled atmosphere have been applied. The manufacture 'aspects concerning the extrapolation from laboratory scale to semi-industrial level with adequate reproducibility and homogeneity are also, discussed. The new improved-contact element called COMPACK -CP has been patented and selected to be applied in LPCE columns of the future Cernavoda Tritium Removal Facility.

Keywords

Tritium separation; heavy water detritiation; contact element ; mixed catalytic packing; hydrophobic catalyst.

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PS1-55 ABSTRACT-dc5e
C. Fuel Cycle and Tritium Processing

The Tritium Exposure and Decontamination (TED) Experiment at the Tritium Laboratory Karlsruhe (TLK)

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Handling tritium always leaves certain residues on surfaces in contact with it. While this is usually of no concern during operation of a vacuum system, analytical devices can already be impacted during standard operation by an elevated or steadily increasing background or for maintenance and dismantling of systems.

This is usually where extensive and time-consuming decontamination campaigns start. By flushing with hydrogen (H_2) or ambient air (containing moisture) an isotopic exchange reaction and thus decontamination is triggered. This produces waste gas which needs to be treated and can only remove tritium lightly bound to surfaces. A bake out can remove more contamination but needs a dedicated heating system, is time consuming and can trigger tritium diffusion into the bulk. Further, heat compatibility of all the parts might significantly limit the beak out temperature (e.g. indium sealing).

All this limits the possibilities of "traditional" decontamination. In literature, the use of Ozone (O_3) for decontamination is described but lacks systematic investigation with respect to Tritium. O_3 is a highly reactive molecule capable of removing residual (tritiated) water and (tritiated) hydrocarbons from surfaces making them accessible to the pumps and hence remove activity. Effectiveness of this method has been shown at the KATRIN experiment leading to the design of a dedicated setup to exploit the possibilities of ozone decontamination and the systematic characterisation and comparison to the "traditional" decontamination methods.

This contribution will show the promising ozone-decontamination results from the KATRIN experiment marking the start point of the TED experiment and the design of the new TED facility with its experimental possibilities. Additionally, we will provide insight to the development of fully vacuum and tritium compatible ozone detectors currently under investigation at TLK which are needed for quantification of the ozone treatment.

Keywords

Tritium, decontamination, ozone, UV light, TLK.

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PS1-56 ABSTRACT-22b4
C. Fuel Cycle and Tritium Processing

The study of AMSB design using adsorption models for coolant purification system of breeding blanket

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An Ambient Molecular Sieve Bed (AMSB) is used in the Coolant Purification System (CPS) of the breeding blanket to remove water impurities, such as Q2O, and maintains the coolant composition. Since the CPS plans to use more than two AMSBs to alternate between adsorption and regeneration phases, an accurate prediction of the AMSB replacement is crucial. Therefore, it is necessary to predict the breakthrough point in a fusion reactor.

In this study, the evaluation between various adsorption models and experimental results using RAVAD (Research Apparatus for Vapor Adsorption and Desorption) is introduced to develop a design of AMSB. Additionally, this study suggests relevant breakthrough points to sustain adsorption performance while considering coolant conditions and operation time. This study expects to contribute to the Q_2O adsorption column design for a fusion reactor and it could be applied to determining the geometric size of the AMSB.

Keywords

Ambient molecular sieve bed (AMSB), Adsorption models, Coolant Purification System (CPS), Breeding blanket.

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PS1-57 ABSTRACT-2685
C. Fuel Cycle and Tritium Processing

2D Tritium retention prediction in the ITER/DEMO actively cooled divertor in the presence of neutron-induced defects

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In future fusion devices, the tritium retention and permeation in plasma facing components (PFCs) is an important safety concerns due to tritium radioactivity. Modelling tritium transport and trapping at defects can be done to assess how fast the tritium inventory builds-up in materials and how much tritium can be lost to the cooling system of actively cooled components.

In this work, we are interested in predicting the tritium inventory and permeation in divertor components of ITER and DEMO, which consists in a tungsten (W) armor actively cooled by water flowing inside copper alloy pipes. 2D modelling of tritium transport in such components has been done with the open-source FESTIM code with trapping parameters corresponding to pristine materials [1]. However, when D/T reactions occurs in the plasma, the PFCs are bombarded by (14.1 MeV) neutrons. It represents up to 3 dpa after 2 full power year for the DEMO [2].

Recently, a neutron-induced trap creation model has been implemented in FESTIM and validated for tungsten [3]. This model is used here to estimate the impact of neutron damage on the evaluation of the tritium retention and permeation. 2D FESTIM simulations in three plasma exposure conditions (leading to hot, medium and low surface temperatures) and in two damaging conditions (fully damaged and transient damaging) are performed. They show that damage acts as a permeation barrier as soon as the damaging rate is high enough to create a significant amount of traps with high (>1.6 eV) detrapping energy. However, this damaging is detrimental for the tritium retention which increases by orders of magnitudes.

[1] E. A. Hodille et al., Nucl. Fusion 61 (2021) 126003

[2] S. Noce et al, Fus. Eng. Design 155 (2020) 111730

[3] J. Dark et al,, 32nd SOFT (2022)

Keywords

Tritium transport, modelling, divertor.

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PS1-58 ABSTRACT-27e9
C. Fuel Cycle and Tritium Processing

Shape changes of beta-ray induced X-ray spectrum by tritium in aluminum and iron with tritium Depth

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Beta-ray induced X-ray spectrometry (BIXS) is a nondestructive tritium measurement method for tritium in solids. The BIXS method is based on the detection of X-rays, induced by beta-particles (BIX). Since the X-ray intensity and spectral shape not just depends on the total amount of tritium but also on the tritium distribution in a solid, the quantitative determination of tritium in a solid by BIXS has been difficult in the past. However, the distribution of tritium in a solid can be obtained by a careful analysis of the shape of BIX spectrum. In this study, the change of the shape of BIX spectra with regard to the tritium distribution was investigated by Monte Carlo simulation.

The Monte Carlo simulation of the BIX spectrum of tritium in a solid was implemented by Galet-BIXS, a user application template for the Geant4 C++ framework. The PENELOPE model was used for a beta particle transport through electromagnetic interactions. The simulation geometry was composed of the sample part and the detector part. Both parts were arranged coaxially. The other space was filled with the process gas. The tritium was generated in the sample part at a given depth to simulate the resulting BIX spectra.

The BIX spectrum simulated, consisted of the characteristic X-ray and bremsstrahlung. Some of the beta particles escaped from the sample solid and induced characteristic X-rays in the process gas. Characteristic X-rays induced within the sample became more intense with increasing tritium depth, up to the range of beta particles. For tritium placed more deeply in the sample, exceeding the range of beta particles, the intensity of characteristic X-rays decreased. The ratio of the characteristic X-ray originating from the sample and from the process gas shows the distribution of tritium in the sample.

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Keywords

Tritium measurement, solid, beta-ray induced X-ray, Monte Carlo simulation.

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PS1-59 ABSTRACT-2d71
C. Fuel Cycle and Tritium Processing

Tritium accountancy and measurement technology in CN HCCB TBS

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TAS is the tritium accountancy and measurement system of HCCB TBS, and its main function is to measure tritium in the HCCB TBS system and to accurately measure the accumulated tritium amount in a certain period, including the real-time monitoring of tritium-containing gas and impurity gas composition at specific locations in the HCCB TBS system and the measurement of tritium delivered to the TAS system from other subsystems of HCCB TBS, and to deliver the measured tritium to the ITER tritium plant. The TAS system is comprised of a tritium-containing hydrogen isotope separation and enrichment loop, a tritium accounting loop, and a tritium measuring loop.

In order to meet the demand for TAS hydrogen isotope separation, it is recommended that the cryogenic distillation method use a combination of stable distillation and intermittent distillation operating procedure. Hydrogen gas that complies with emission requirements of PBS.32 by having a light component tritium concentration of 9.5×10^{7} Bq/m^{3}, and a volume ratio of raw gas to recombinant fraction product gas of 30. Tritium calorimeter is used for accounting the enrich tritium from cryogenic distillation process through 10mg up to 1000mg depending on the fusion power. Meanwhile tritium ionization chambers are utilized for real-time determination of tritium gas content which located at TES and CPS. The home-made micro-GC is used for gas impurities less than 1 ppm and can also determine the abundance of hydrogen isotopes.

Keywords

HCCB TBS, Tritium Accountancy System, Cryogenic distillation, Tritium measurement.

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PS1-60 ABSTRACT-3058

C. Fuel Cycle and Tritium Processing

Advanced Modular System for tritium separation by CECE type process

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The main source of tritium is fission reactors PHWR (CANDU) or molten salt (MSR), nuclear fusion reactors (JET, ITER, DEMO) which, in operation, generate significant amounts of tritiated light water or tritiated heavy water, whose efficient processing, through an advanced technology, would contribute to the reduction of stocks and the recovery of tritium and deuterium.

The research presented in this work aims to promote a modern management of tritium and its associated materials, by developing a technology for separating hydrogen isotopes through the CECE (Combined Electrolysis and Catalytic Exchange) separation process, for the concentration of tritium in liquid state (HTO, DTO).

The conceptual diagram of the Advanced Modular System (AMS), the main component elements, together with the description of 3 modes of experimentation will be presented.

The technological process that takes place within AMS is electrolysis of process water and, respectively, isotopic exchange on mixed catalytic packing.

Keywords

Tritium separation, waste, electrolysis.

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PS1-62 ABSTRACT-bba2
D. Material Engineering for FNT

New processing routes for zr-based ods ferritic steels

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Nuclear fusion reactors represent a technology challenge for both research and industrial development. ODS RAF steels are among the materials that are currently being developed for fusion reactor structural applications. They are characterized by a potentially superior mechanical strength at high temperatures and radiation damage resistance thanks to the presence of a homogeneous distribution of nanosized Y-rich oxide particles, fine grain sizes, formation of secondary phases, and high dislocation densities. Particularly, the nanosized oxide particles may act as sinks for radiation defects increasing the irradiation resistance of the material.

The optimization of the production and processing route is essential to achieve the most homogeneous distribution possible of nanoparticles. Fe-Cr-W-Ti-Y_2O_3 ODS alloys have been widely studied so far; it has been observed that the addition of Ti enhances and refines the nanoparticle dispersion. Exploring novel compositions and new processing routes is relevant to obtain other varieties of ODS steels that may reduce production costs. In this work, Zr is proposed, instead of Ti, to refine the nanoparticle dispersion. The Zr-based ODS RAF steel, Fe-14Cr-2W-0.3Zr-0.24Y (wt. %), was produced using pre-alloyed atomized powder. In this research, three different processing routes are explored: 1.a.) the ODS-powder is mechanically alloyed and consolidated; 2.a.) the ODS-powder is oxidized and directly consolidated; 2.b.) the ODS-powder is oxidized, mechanically alloyed and consolidated. The ODS-powders were consolidated by hot isostatic pressing and later hot rolled obtaining a ~75% thickness reduction. Preliminary results regarding the microstructure of the obtained materials by scanning electron microscopy and X-ray diffraction are presented and discussed with focus on the influence of the different processing routes. This research will provide relevant information on the influence of Zr on the nanoparticle formation and dispersion, as well as on the selection of the processing routes that optimize the microstructure and mechanical properties of the ODS steel.

Keywords

ODS RAF steel, thermal aging, Charpy test, DBTT, EBSD.

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PS1-63 ABSTRACT-50e6
D. Material Engineering for FNT

Sputtering fabrication and characterization of sic coatings for breeder blanket applications

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Liquid eutectic PbLi breeder blankets (BB) are one of the most promising designs for both magnetic and inertial fusion reactors. For these BB to work in a reliable, safe and durable operation mode, it is necessary to deposit a coating on the structural steel that simultaneously minimizes tritium leaks by permeation and prevents steel corrosion.

Here, we report on the influence of the sputtering parameters (Ar gas pressure) on the morphology, structure, and density of SiC coatings. We will show that decreasing the Ar content during the sputtering process leads to changes in the morphology from columnar-like to homogeneous coatings with high density (~3.2 g/cm^3). These dense coatings have good adhesion to the steel substrate, with a critical load before failure around 260 mN.

The thermal stability of the coatings after being subjected to thermal cycles, as well as their permeation reduction factor before and after ionizing radiation will be discussed. Finally, a first approach to PbLi compatibility with this coating candidate will be presented.

Keywords

Breeder blanket, coating, tritium permeation, corrosion barrier, PbLi, SiC.

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PS1-64 ABSTRACT-a37e
D. Material Engineering for FNT

Relating the formation energies for oxygen vacancies defects to the structural properties of tungsten oxides

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To address the growing demand for low carbon energy, the UK government has recently committed to building the world's first nuclear fusion power plant by 2040, the Spherical Tokamak for Energy Production (STEP). One of the challenges faced by the STEP project concerns the oxidation of the reactor's tungsten-based first wall, which may occur during a loss of coolant accident or during remote handling during maintenance.

The oxidation of tungsten is a complex process, starting with the formation of a protective WO2.72 layer. This layer then cracks leading to rapid oxygen transport to the oxide/metal interface and the formation of columnar WO2.92. The abundance of vacancies in WO2.92 leads to rapid growth of the oxide layer and progressive transformation to WO3. WO3 is permeable to oxygen, and its formation rate depends on oxygen ion transport to the WO3/WO2.72 interface. Therefore, in this work we use density functional theory to study the physical properties of all oxide phases involved in the oxidation process. From the DFT simulations we show how key features of the electronic structure of the WOx materials change as the metal-oxygen ratio evolves. Then we calculate the formation energies for oxygen vacancies and activation energies for their diffusion allowing an assessment of their mobility in the different tungsten oxide phases. Our results provide a new level of understanding of the sub-stoichiometric Magnéli phases that are observed during the oxidation of tungsten.

Keywords

Tungsten, materials, materials modelling, simulation, oxidation.

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PS1-65 ABSTRACT-392f
D. Material Engineering for FNT

Scoping the production of 6Li for ITER and DEMO BB procurements

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The maximum programmatic risk for assure the needed (Pb-15.7(2)6Li) is related with the need to procure tens of kilograms of 6Li for EU ITER WCLL TBMs. Tens of Tones of pure 6Li will be required for DEMO WCLL BBs DEMO according to present Plans.

6Li is a dual-use material and 6Li military stocks exit in some ITER Agreement countries. It is always an open geopolitical or geo-strategic question, with strong impact on the public opinion scrutiny if civilian energetic Programs as ITER or future DEMO; should depend on such military supply of 6Li. The confirmed commercial supply of 6Li is of few tens of grams.

In this context, a feasible 6Li industrial-scale (tens of kgs of 6Li) production technology should be urgently demonstrated to guarantee procurement needs.

A detailed review of known available techniques both chemical and physical for 6Li industrial production is presented. Techniques differ in terms of: (1) safety and environmental friendliness; (2) nominal efficiency for industrial scalability and (3) energetic and economical affordability. Industrial routes are proposed formally analogue and close to the available and historically demonstrated experience for Uranium.

A proposal for the design of a 6Li production line is discussed as a key starting point of the proposed 6Li Procurement Plan.

Keywords

Lithium, ITER, DEMO.

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PS1-66 ABSTRACT-0965
D. Material Engineering for FNT

Effects of magnetic field and water radiolysis on corrosion properties of RAFM, F82H

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The in-vessel components of the Japanese DEMO reactors are planned to be water-cooled. In this system, high-temperature and high-pressure water is used as the coolant, which causes corrosion of structural materials in contact with the water. The corrosion database will be used as input data for setting the corrosion allowance in the design and for evaluating radiation protection for workers. Although basic corrosion data for in-vessel components such as blanket and divertor have been obtained, the next step is to examine the necessity of corrosion data in the fusion specific environment and to obtain such data. QST has been developing a magnetic field flow corrosion test system and a hydrogen peroxide injection corrosion test system that simulates water radiolysis. In this presentation, the results will be presented.

The specimen is a reduced activation ferritic steel F82H-BA12. Hydrogen peroxide injection hydrostatic corrosion tests were conducted using 20 x 10 x 1 mm coupon-shaped specimens in high-temperature, high-pressure water at a temperature of 300°C and a pressure of 15 MPa. The amount of hydrogen peroxide injected was controlled to be 1 ppm/h and 10 ppm/h. The dissolved oxygen concentration was kept below 20 ppb. The maximum test time was 250 h.

The weight change rate up to 250 h was calculated to be 0.0111, 0.0082, and 0.0069 $mg/cm^2/t^{0.5}$ for the uninjected, 1 ppm/h, and 10 ppm/h conditions, respectively. Since the weight change in the hydrostatic corrosion test is determined by the balance between the amount of oxide formed and the amount of eluted ions, the injection of hydrogen peroxide at least changes the corrosion properties, although more detailed analysis of the oxide and other details are necessary. The results of magnetic field environmental corrosion tests will also be discussed in this presentation.

Keywords

Reduced activation ferritic steel, Corrosion, Magnetic field, Water radiolysis.

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PS1-67 ABSTRACT-1218
D. Material Engineering for FNT

Investigation of hydrodynamic characteristics of Li jet flowing along concave flow channel using LES for A-FNS

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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. We have been studying the high-speed Li jet experimentally and numerically in collaboration with QST, in which Li Loop of Osaka University (LLOU) and EVEDA Li Test Loop (ELTL) of QST were used. For the safety and the efficiency of such actual FNSs, it is significant to understand the detailed flow structure inside the Li jet for removing heat load due to deuteron beam. At this time, ELTL was already dismantled and thus experiments using LLOU become more important for the development of A-FNS. However, LLOU has the different configuration from A-FNS. Li target in ELTL and A-FNS has concave and vertical flow channel and 25 mm in the jet thickness. Therefore, the evaluation of the flow structure inside the Li jet in A-FNS leads to clarify the difference of the hydrodynamic characteristics of the Li jet between LLOU and A-FNS, and this also enables us to utilize experimental data of LLOU to the development of A-FNS. In this study, firstly, single phase simulation of Li flow inside the two-staged contraction nozzle of ELTL using Large Eddy Simulation (LES) was conducted and the result was compared with that of LLOU for verifying the simulation method. After that, two phase simulation of the Li jet of ELTL using LES was carried out and the results were verified by comparing it to the experimental results in ELTL. Finally, the flow structure inside the Li jet flowing along concave flow channel was evaluated for predicting heat transfer inside the Li jet in beam-on-target.

Keywords

Fusion neutron source, Liquid Li target, LES.

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PS1-69 ABSTRACT-247e
D. Material Engineering for FNT

Fusion magnet quench risk increase with radiation damage

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Superconducting magnets are often proposed to confine plasma in fusion reactors. Superconducting material enables the magnets to carry current densities that melt materials with non-zero resistance. Quench occurs when superconductivity is lost and the current starts to generate heat. Unless prevented with a fast enough control system, the heat generated during a quench can cause catastrophic damage to the coils. This work describes a less-studied heating mechanism that increases the likelihood and consequence of fusion magnet quenches. Defects accumulate in the magnet material under irradiation by the fusion process. The defects store energy in the material and change thermal and normal state electrical properties. Wigner energy is released when defects anneal. After 1 year of operation at 20 K a 10 K disturbance is predicted to result in final temperatures of 30.64 K, 31.35 K, 32.13 K for 2%, 4%, and 6% Wigner energy storage rates. These non-negligible increases in temperature can facilitate an earlier transition to normal state for the superconducting material. The higher-than-expected temperature of the material accelerates the temperature dynamics of the quench by increasing Ohmic heating. The continuous operation of a fusion reactor produces an increasingly unstable thermodynamic system in superconducting magnets. The temperature margin between operation and quench runaway reduces with irradiation. The next step is determining the Wigner energy storage rate for fusion magnet materials. Implications are more stringent specifications on quench control systems and coil operating duration limits between periodic releases of Wigner energy to avoid catastrophic quench failures.

Keywords

Wigner energy, fusion magnets, quench risk, materials.

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PS1-70 ABSTRACT-2a2f
D. Material Engineering for FNT

Creep Rupture Behavior of TIG Weldment of CLAM Steel

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China Low Activation Martensitic (CLAM) steel has been chosen as the structural material in the designs of PbLi blankets for series of fusion reactors (named FDS series) and China fusion engineering test reactor (CFETR) blanket in China. Especially, it has been chosen as the primary candidate structure material for Chinese Test Blanket Module (TBM) for ITER by China International Nuclear Fusion Energy Program Execution Center (CNDA). The R&D activities on welding CLAM steel by different welding methods were being conducted with wide international and domestic cooperation.

In this manuscript, creep rupture behavior of TIG weldment CLAM steel was studied in the temperature range of 500 - 600 °C and at stresses of 155 - 260 MPa. Under similar conditions, the creep resistance of the weldment was lower than that of the base metal. The heat affected zone (HAZ) of the weldment consisted of coarse prior-austenite grain (CGHAZ), fine prior-austenite grain (FGHAZ) and intercritical (ICHAZ) regions in a sequence from the fusion boundary to unaffected base metal. The type IV failure in the FGHAZ was found to be the typical failure mode. The failure was associated with localized creep deformation coupled with creep cavitation. The fracture morphology of the crept welded specimens was found to be a mixture of transgranular and intergranular mode. A lot of cavities formed at grain boundaries during creep, which indicated that the coarse precipitates of Laves phase were considered as the preferential nucleation sites for cavities.

Keywords

CLAM; Weldment; Creep.

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PS1-71 ABSTRACT-be68
D. Material Engineering for FNT

Thermo-Hydraulic Analyses of the Heat Transfer in the Rib-Roughened Air Cooling Channel

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The plasma facing first wall of a fusion power reactor has to absorb high heat fluxes from the plasma. In the helium cooled blanket, heat is transferred through the first wall to the coolant high pressure helium flow. Uniform heat flux densities at the first wall can be expected in a range from 0.5 to 1.0 MW/m^2. Rib-roughened cooling channels in the plasma facing components enhance heat transfer and reduce structural material temperatures.

The aim of the study is to find the most accurate and robust CFD solution that can be applied to cases with flow separation and heat transfer. The applicability of six turbulence models was examined for the prediction of the convective heat transfer and pressure drop in the ribbed gas cooling channel measured in HETREX-PT (HEat TRansfer Enhancement eXperiments - Pressure, Temperature) experiments. Turbulence models such as standard k-ω SST and k-ε models as well as models derived on the basis of the elliptic blending approach (Reynolds Stress model, v2-f model, k-ε model and κ-ε model with the lag term transport equation) were selected.

The flow conditions (flow rate, heating power) were scaled from helium flow in HCPB to air. The rib-arrays consist of transversally oriented 60° V-shaped ribs. The comparative analysis of the turbulence models deals with the investigation of the distribution of vortex structures caused by ribs and their influence on heat transfer.

The standard k-ε model underestimates the heat transfer within the recirculation region between the ribs. The flow between neighbour ribs computed by k-ω SST is fully attached, which also leads to the significant under prediction of the heat transfer. The elliptical blending models give results that are closer to the measurements. The v2-f model gives the best prediction within 6 percent of the measured.

Keywords

First Wall, HCPB, Heat Transfer, CFD.

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PS1-72 ABSTRACT-34bd
D. Material Engineering for FNT

Galvanic process for infiltration of W fibre-reinforced heat sinks

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A challenging aspect in view of the realization of a future magnetic confinement fusion reactor is the design and manufacture of highly loaded divertor target plasma-facing components (PFCs) which have to sustain intense particle, heat and neutron fluxes. In this context, tungsten-copper (W-Cu) composites are currently being investigated as potentially advanced heat sink materials for PFCs. The development and manufacture of W-Cu composite pipes, which are bonded to tungsten monoblocks, poses new challenges in terms of manufacturing and accuracy.

A new approach is proposed to produce the W-Cu composite pipe. The proposed alternative is to directly infiltrate tungsten fibers with galvanic copper instead of by copper infiltration in an oven. The paper presents the status of the development to better asses the impact of various parameters affecting the galvanic process. Different braid thicknesses (3 to 16 layers) and their arrangement (45° and 80°) have been tested. The effects of the electrolytic bath circulation, the variation in the braid material (porosities), and the electrolytic exchange by using pulsating direct current on the diffusion in the braid have been investigated. The influence of the passivation mode has been also carefully analyzed because in the case of a strong passivation copper only grows outwards from the core and not onto the tungsten braid.

Keywords

Plasma facing component, divertor target, galvanic process, tungsten copper.

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PS1-73 ABSTRACT-091b
E. Vacuum Vessel and Ex-vessel Systems

Structural and thermal fluid dynamic analyses of the ITER Pressure Suppression System considering no stable steam condensation regimes

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At University of Pisa, an experimental campaign was performed with the purpose to qualify the ITER Vacuum Vessel Pressure Suppression System (VVPSS) and studying the Direct Contact Condensation at sub-atmospheric conditions.

A Large-Scale Experimental Facility was designed and built, with the financial support of ITER Organization, to investigate the steam condensation in the operation conditions of the ITER VVPSS.

The experimental tests were performed by injecting steam, produced by an electric steam generator (1.7 MW of power), through a multi-holes sparger into a condensation tank called Experimental Test Tank (ETT).

On the sparger support, strain gauges and an accelerometer were installed to analyse the structural behaviour during the experimental tests.

Under certain thermal hydraulic conditions (high subcooling and low steam mass flow rate) a big steam pocket can form and collapse causing high loads on the sparger structure.

Two different spargers were used with different designs (the reduced scale sparger A with 100 holes and the full-scale sparger B with 1000 holes) which developed steam bubbles with different dynamics and dimensions.

The aim of this study is to compare numerical fluid dynamic and structural analyses with the experimental results at different thermal hydraulic conditions in order to determine the pressure impulse caused by bubble collapse.

The bubble collapse was simulated by means of ANSYS Fluent code in order to analyse the bubble dynamics and to determine the pressure impulse. The shape, the dimension and the collapse time of steam pocket were experimentally determined by means of an image analysis of video recorded during the tests.

A dynamic structural FEM analysis was performed applying the pressure impulse on the sparger in order to evaluate the accelerations and the strains.

The experimental acceleration signals were compared with the numerical results. The outcomes demonstrated a good agreement between the numerical and experimental results.

Keywords

Nuclear Fusion, ITER, VVPSS, Direct Contact Condensation.

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PS1-74 ABSTRACT-1e8d
E. Vacuum Vessel and Ex-vessel Systems

Electromagnetic analysis of the Optical Hinge and Optical Relay Unit of the Wide Angle Viewing System diagnostic for ITER

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The ITER Visible and Infrared Wide Angle Viewing System (WAVS) is a diagnostic aiming to optically monitor the tokamak first wall and divertor for machine protection, plasma control and physics analysis. The Optical Hinge (OH) and Optical Relay Unit (ORU) are reflective optical systems, being the first two components of the WAVS diagnostic in the Interspace area. These two components share a common support structure, which is in turn attached to the Interspace Support Structure (ISS).

In order to assure the optical performance, the OH, ORU and the OH-ORU assembly, including its support structure, have to withstand all the relevant loads defined for the Final Design. In particular, electromagnetic (EM) loads are developed in the Interspace area under EM events due to plasma instabilities. Volumetric forces arise during transient EM events from the interaction of the background magnetic field with the eddy currents induced in the conductive components when they experience time-varying magnetic fields. In case of Category III and IV loads, including load combinations with certain Major Disruption and Vertical Displacement Event cases, the OH-ORU integrity has to be ensured since its structure is classified as Safety Relevant.

The EM volumetric loads are calculated through a 3D finite element model and will serve as inputs for the later structural analyses. The paper summarizes the EM analysis of the OH-ORU, performed by CIEMAT, to validate its Final Design.

Keywords

ITER diagnostic, WAVS, Optical Hinge, Optical Relay Unit, electromagnetic analysis, FEM.

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PS1-75 ABSTRACT-d07c
E. Vacuum Vessel and Ex-vessel Systems

Dynamic behaviour of the DTT torus complex during seismic events

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The Divertor Tokamak Test (DTT) facility is currently under construction at the ENEA site of Frascati [1]. The construction site is characterised by high seismic risk which requires analysis and use of the local seismic response; therefore, it is crucial to assess the behaviour of the DTT torus complex in case of a seismic event. Preliminary evaluations on the DTT structure supporting the torus complex, the cryostat base, have been made considering both equivalent static approach and response spectrum analysis to simulate the seismic effects on the machine, as provided by reference standards. The purpose of the present work is to expand the knowledge about the dynamic behaviour of the DTT torus complex during seismic events and to study seismic isolation solutions producing a reduction in stress characteristics. In particular, the employment of two possible types of seismic isolators, elastomeric and pendulum, and two different seismic isolation configurations have been analysed. The first configuration presents isolators located between the base and the reinforced concrete foundation, while the other one features isolation devices directly included at the top of the base columns which support the machine. Both configurations have been investigated and compared. In order to realise the comparative study, two simplified FE models of the DTT cryostat base and its interfaced components have been realised and analysed. Subsequently, the differences in terms of displacements and stress fields have been remarked and a design proposal has been made. The capability of displacement absorption of the plant connections has been verified. Then, seismic spectra calculated at the gravity supports can be applied for the verification of vacuum vessel and magnet system under different load combinations. In the end, a comparison between the selected seismic isolation configuration and the baseline non-isolated is presented to assess the benefits introduced by the seismic isolation system.

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Keywords

Structural analysis, dynamic analysis, seismic analysis, torus complex, cryostat base, vacuum vessel.

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PS1-76 ABSTRACT-660c
E. Vacuum Vessel and Ex-vessel Systems

Characterization of a NEG cartridge under high pressure conditions

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The development of advanced sintered pumping elements based on the ZAO® alloy, characterized by a high affinity towards hydrogen and its isotopes, make Non-Evaporable Getter (NEG) pumps particularly appealing for applications in fusion research, which typically deal with large fluxes of these gaseous species. NEG getters show high pumping speeds and capacity, excellent reliability with ability to withstand a high number of loading and unloading cycles without showing any type of failure, ease of integration within a complex system (such as a fusion reactor) because of their high specific properties and the simplicity of the mechanism that regulates its operation. These properties make NEG technology particularly suitable for the development of compact, efficient and easily scalable solutions, which are also very safe, since the release of absorbed hydrogen is prevented in the case of power outage or subsystem failures, as it can occur only by providing an adequate amount of heat to the getter material.

This paper reports the results of an experimental campaign conducted on a new type of NEG pump, aimed to obtaining an accurate characterization in response to different operating conditions. Hydrogen pumping speed was investigated in a range of pressures of particular interest for the activities of the Divertor Tokamak Test facility (between 0.1 and 10 Pa). The influence of nitrogen absorption and getter material operating temperature on the pumping capacity of the NEG pump were also examined.

Keywords

Vacuum, Pump, Getter, NEG.

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PS1-77 ABSTRACT-8c65
E. Vacuum Vessel and Ex-vessel Systems

Development of Penning ion gauge for fast neutral pressure measurement in VEST

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A fast response Penning ion gauge (PIG) has been developed for measuring the temporal changes in neutral pressure within the magnetic configurations of tokamaks, and it has been successfully operated in Versatile Experiment Spherical Torus (VEST). It has been revealed that the applied voltage, neutral pressure, and magnetic pitch angle are the primary design parameters affecting the gauge operation. In order to characterize the measurable ranges of the PIG within the Spherical Torus (ST) configuration, a preliminary test was conducted to specifically investigate the effect of magnetic pitch angle on Penning gauge discharge behavior. The results of preliminary test indicate that modification of the longitudinal aspect ratio of the gauge cell geometry can expand the gauge operation range, as observed with two different gauge structures. Finally, a microsecond-order time resolution for neutral pressure measurement was achieved using the newly designed PIG under the magnetic pitch angle variation of up to 20 degrees. The developed gauge was employed to measure rapidly evolving neutral influx resulting from plasma wall interaction within the VEST plasma duration. This enabled the investigation of various wall conditioning effect on tokamak plasma discharge in the VEST.

Keywords

Penning ion gauge (PIG), Magnetic pitch angle, Neutral pressure.

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PS1-78 ABSTRACT-0174
E. Vacuum Vessel and Ex-vessel Systems

Analysis of the water flow distribution in the Vacuum Vessel of the Divertor Tokamak Test facility

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The Divertor Tokamak Test (DTT) facility is a superconducting tokamak being built at the ENEA Frascati to foster the development of advanced divertor solutions for the EU DEMO.

The DTT Vacuum Vessel (VV) will be procured in 3 multi-sectors, in turn divided into one or more sectors; due to manufacturing and integration constraints, the latter cannot share the same design: some of them will be identical (“regular” sectors), whereas the remaining will all be different ("non-regular”).

In DTT the VV plays also the role of neutron shield for the superconducting magnets, and therefore it will be actively maintained at the operating temperature of 60 °C by borated water in forced flow to counteract the thermal loads (pulsed heating from the plasma and static radiative cooling from the cryostat). The water will flow in the free space between the two shells composing the VV; given the complexity of the geometry, a careful hydraulic design is mandatory, to avoid local stagnation points which may cause either overheating or freezing. An additional requirement is the possibility to fully drain the entire VV from the bottom to perform baking with nitrogen at ~200 °C. The water will enter the sectors in parallel by 9 inlets at the bottom and leave them by 9 outlets (staggered by 20°) at the top. This identifies 18 separate hydraulic paths, covering 20° toroidal portions and therefore reflecting the differences between the “regular” and “non-regular” sectors; this asks for a proper balance of the mass flow repartition among them.

This work presents a set of Computational Fluid-Dynamics analyses of the DTT VV to address the issues above. All the different hydraulic paths are separately analysed with the Star-CCM+ software, proving the effectiveness of their design. In addition, the coolant mass flow rate distribution among the different paths is assessed.

Keywords

DTT, Vacuum Vessel, thermal-hydraulics, CFD.

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PS1-80 ABSTRACT-2d2b
E. Vacuum Vessel and Ex-vessel Systems

CarMa0NL modelling of disruption forces on COMPASS

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The sideways forces are recognized by the ITPA community as a critical priority for ITER [1]. To address the issue, first we model vertical displacement events and related electromagnetic forces on the COMPASS tokamak vacuum vessel with CarMa0NL code [2]. The chamber is represented with high level of detail to consider the effect of asymmetrically placed ports on the generation of sideways forces [3, 4]. Next, the numerical results are compared with the forces measured using a novel magnetic technique suggested in [5]. Specifically, we use the data obtained with saddle coils only. This allowed us to estimate the disruption forces with higher precision and temporal resolution with respect to the standard approach based on mechanical gauges or accelerometers. Finally, we extrapolate the findings to ITER.

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Keywords

Tokamak, Disruptions, Sideways Forces, Vacuum Vessel, ITER.

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PS1-81 ABSTRACT-330a
E. Vacuum Vessel and Ex-vessel Systems

The FALCON facility for high-power testing of Electron Cyclotron components

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Due to the reliability achieved by Gyrotrons, to the capability of depositing power over a wide range of plasma parameters and to the relevance for future devices such as DEMO, the Electron Cyclotron (EC) is gaining more and more importance as plasma heating system in fusion devices.

ITER relies on a set of 24 EC sources, each producing 1MW of RF power at 170GHz in continuous operation (CW). A set of transmission lines carry the RF beams to the launchers that inject the power into the plasma chamber.

F4E is responsible for the design and procurement of eight High Voltage Power Supplies, six Gyrotrons, four upper launchers, the EC plant control system and the EC components that are part of the ITER first confinement system (FCS), namely the diamond window unit (DWU), ex-vessel waveguides that connect the tokamak up to the DWU, mitre-bends and isolation valve.

As a risk mitigation F4E has built a facility named FALCON, installed and operated at the Swiss Plasma Center of the EPFL in Lausanne, Switzerland, where EC components can be tested under high RF power in virtually CW (~1000s) in order to validate the design and perform qualification of ITER components.

The FALCON facility is now attracting other users, namely industry developing EC components and other Domestic Agencies procuring components for ITER.

The present paper presents the configuration, testing capability and foreseen developments of the FALCON facility and reports on the results obtained so far in the estimation of RF power absorption by prototype waveguides and miter-bends developed for ITER.

Keywords

Electron Cyclotron, Testing Facility.

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PS1-82 ABSTRACT-135e
E. Vacuum Vessel and Ex-vessel Systems

Maintenance study on WCLL-TBS Ancillary systems in Process Room

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The Water-Cooled Lithium Lead Test Blanket System (WCLL-TBS) is one of the European Test Blanket Systems candidate for being installed and operated in ITER. The ongoing activities toward the Preliminary Design Review of the WCLL-TBS Ancillary Systems include the reallocation of their units in the reserved space, also in the view of the proper execution of the maintenance activities.

Starting from the available documentation (i.e. Process Flow Diagrams, Process and Instrumentation Diagrams, System Design Description), detailed 3D CAD models have been developed for two Ancillary Systems to be installed in the Process room: the Tritium Extraction System (TES) and the Tritium Accountancy System (TAS). In the Process room, the largest part of the TES process units are placed inside a Glove-box, as enclosure to avoid the contamination of the room in case of radioactive release due to failure of units. Based on the expected radioactivity content, all the TAS units are placed inside the Glove-box.

Maintainability studies have been performed, focusing on the preventive (i.e. planned) and corrective (i.e. required by failures) maintenance tasks on the units installed inside the Glove-box. They were based on the 3D CAD models and supported by the use of Virtual Reality as intuitive and immersive human-computer interface. They provided insights for the iterative improvement and consolidation of the Systems design and information for the evaluation of their availability.

Keywords

ITER, WCLL, TBM, Glove-Box, Process Room, TES, TAS.

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PS1-83 ABSTRACT-1745

E. Vacuum Vessel and Ex-vessel Systems

Development of a scintillator based fast-ion loss detector for the Wendelstein 7-X stellarator

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A primary scientific goal of the Wendelstein 7-X stellarator (W7-X) is to demonstrate improved fast-ion confinement for this type of fusion reactor. W7-X therefore requires diagnostics capable of inferring the fast-ion confinement properties of the device. Fast-ion loss detectors (FILD) are especially well suited for this purpose. In particular, scintillator based FILDs [1,2] provide the capability of obtaining time-resolved velocity-space measurements of the escaping fast-ions. Furthermore, these detectors offer high signal-to-noise ratios (SNR) allowing for high speed measurements and the ability to infer losses due to magnetohydrodynamic instabilities.

In this contribution we will present the design and development of a scintillator based FILD system for W7-X. The mechanical design of the new so-called sFILD diagnostic will be presented along with a mechanical analyses of the structural and electromagnetic loads expected on the diagnostic. Furthermore, an analysis of the thermal characteristics of the actively cooled sFILD probe head will be presented. Finally, the expected measurement performance of the diagnostic based on the characteristics of the embedded optical relay system and ASCOT5 [3] predicted losses will be presented. Here, the FILDSIM code [4] will be used to determine the instrument response function of the detector and used to calculate synthetic signals via forward modelling. To determine the expected SNR various noise contributions such as photon, camera readout and neutron induced noise will also be modelled.

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Keywords

Fast-ion loss detector, Wendelstein 7-X, stellarator.

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PS1-84 ABSTRACT-17f1

E. Vacuum Vessel and Ex-vessel Systems

Design and Integration of the European Diagnostic Ports of ITER

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In the last few years, F4E and IDOM have been collaborating with the ITER Organization, Domestic Agencies and Fusion Laboratories in five countries for the integration of more than twenty diagnostic instruments of very different nature (magnetic, neutron, optical, bolometric, spectroscopic, microwave and operational diagnostics) in the six European Diagnostic Ports of ITER: four Upper Ports and two Equatorial Ports. This work has entailed an intensive systems engineering and detailed design definition endeavor.

As a result, we have arranged about 500 tons of steel, water, boron carbide and other special materials strategically and with great detail, to enable precise and accurate performance of the diagnostics, while properly protecting these and other systems from the harsh environment inside the vacuum vessel: extreme neutron irradiation, high thermal loads, electromagnetic forces due to plasma disruptions, seismic events, etc.

The design of the diagnostic ports involves in-vessel and ex-vessel components. The in-vessel components, located inside the Port Plug, are the Diagnostic Shielding Modules, identified as the most critical components due to the harsh environment and complexity of the hosted diagnostics, but also the cooling water, gas and electrical services including the feedthroughs installed at the primary confinement barrier which have been specifically qualified for its application in ITER. The ex-vessel components comprise the Interspace, Bioshield Plug and Port Cell structures, also hosting diagnostic components.

In addition to the above mentioned requirements, the Diagnostic Ports include Protection Important Components (PIC), which provide first confinement barrier for tritium and other hazardous substances, classified as a safety function. The verification of this safety function in all the applicable load combinations and conditions has required particular qualification in compliance with specific nuclear safety defined requirements and the applicable nuclear standards.

This paper aims at providing a glimpse of that exciting and challenging project.

Keywords

ITER, Diagnostic Ports, Final Design, In-vessel Components, Ex-vessel Components, Integration, Nuclear Safety.

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PS1-84b ABSTRACT-fbc1
E. Vacuum Vessel and Ex-vessel Systems

Thermal-hydraulic and mechanical analysis of the Beam Line Components for the DTT Neutral Beam Injector

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The Divertor Tokamak Test (DTT) facility is currently under construction at ENEA Frascati. DTT is a compact, superconducting tokamak that will study different solutions to address the issue of power exhaust in EU DEMO perspective. A significant amount of auxiliary heating power will be needed, to reach relevant values of the heat flux on the divertor target: Electron Cyclotron Resonant Heating (ECRH), Ion Cyclotron Resonant Heating (ICRH) and a Neutral Beam Injector (NBI) are foreseen for this purpose.

The NBI is made of an ion source and an accelerator, connected to a vacuum vessel containing the Beam Line Components (BLCs), i.e. the neutralizer, the residual ion dump and the calorimeter. All these three components are water-cooled and currently in their conceptual design phase: their configuration is being continuously updated and their performance needs to be verified by means of detailed numerical analyses before moving to the engineering design.

In this work, suitable numerical models developed for the latest design of the three BLCs are presented. First, hydraulic analyses have been carried out to assess the compliance of the flow characteristic with the constraints prescribed by the cooling system. The pressure drop is compared with that of the previous designs, and further optimizations, if needed, are proposed. Then, the updated estimations of the heat load on the different components are input to the model to perform thermal-hydraulic simulations, highlighting the regions close to (or beyond the) boiling point of the coolant, especially in the calorimeter, and the hot spots in the solids; the possible need of suitable turbulence promoters to increase the heat transfer is also discussed. Finally, the computed temperature map is used as input for a thermo-mechanical analysis, suggesting possible improvements to reduce the stress in the different BLCs.

Keywords

DTT, mechanics, modelling, NBI, thermal-hydraulics.

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PS1-85 ABSTRACT-0693
F. Nuclear System Design

Development of transport-activation internal coupling code based on cosRMC and application to fusion reactors análisis

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The solid components and coolant in fusion reactors are activated under fusion neutron irradiation. Accurate calculation of radionuclide activity and decay photon dose rate generated by fusion neutron irradiation is of great significance for radiation protection design of fusion reactors. In this paper, a transport-activation coupling code was developed based on Monte Carlo particle transport code cosRMC. The internal coupling of transport and activation of cosRMC was developed through the built-in depletion solver Depth. Compared to external coupling which relies on the external transmission of neutron energy spectrum from Monte Carlo code to activation code, the internal coupling mode calculates the activation-related one group reaction cross sections of different nuclides during the neutron transport process, which is more efficient and flexible. By directly using continuous energy cross section to calculate one group cross sections, the pre-processing of energy spectrum related multi-group cross section was no longer required, which improves the accuracy and versatility. The developed transport-activation coupling code is applied to the activation calculation of fusion reactors including the Demonstration Reactor (DEMO), the International Thermonuclear Experimental Reactor (ITER) and the Chinese Fusion Engineering Testing Reactor (CFETR). The verifications are carried out by comparisons with the activation code ALARA. The calculation results of cosRMC are in good agreement with those of ALARA, and the relative error of atomic number density of major nuclides is less than 5%, which verifies the developed transport-activation coupling code can be for activation analysis of fusion reactors.

Keywords

Activation calculation, Monte Carlo, internal coupling, fusion reactor, cosRMC.

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PS1-87 ABSTRACT-51dd
F. Nuclear System Design

Operating Conditions of DEMO Divertor Considering Structural Safety and Thermal Efficiency

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The divertor, which is one of the main components of tokamak type nuclear fusion, is characterized by a cyclic high heat flux and large power load. High heat flux causes structural instability such as fatigue, ratcheting, Critical Heat Flux (CHF), recrystallization. The large amount of plasma power(360MW) is loaded to K-DEMO divertor, which corresponds to 60% of the total plasma power(600MW). Discarding such a large amount of power is a huge loss, it is important to design systems with high thermal efficiency aimed at generating electricity. For this reason, it is essential to select the proper operating condition of the divertor coolant which is the factor that determines safety and thermal efficiency.

Assume that the flow rate of the coolant is increased, and bulk temperature of the coolant is decreased. The CHF value, which is a thermal hydraulic safety index, will be increased to secure safety, but the efficiency will decrease due to the pumping power of the high flow rate and the low exit temperature of the divertor. Safety and thermal efficiency, these two goals are at odds with each other. This study aims to derive operating conditions that show maximum electricity production efficiency in a state where safety is secured.

In order to evaluate the structural safety of divertor according to operating conditions, thermal analysis and structure analysis according to each operating condition must be accompanied. Through thermal analysis, the temperatures of the armor of the facing component were confirmed by the periodic heat flux of plasma. Safety evaluation was conducted through structural analysis based on the Monoblock Elastic Assessment Procedure (MEAP) Rule with the calculated thermal analysis values. The optimal operating conditions were obtained by calculating the electrical efficiency only under the operating conditions in which safety was ensured.

Keywords

Operating Conditions, Critical Heat Flux, Monoblock Elastic Assessment Procedure Rule, Structure Analysis, Thermal Efficiency, Rankine Cycle.

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PS1-88 ABSTRACT-fc7a
F. Nuclear System Design

Conceptual design and supporting analysis of a Double-Wall Heat Exchanger for the ARC fusion reactor primary cooling system

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The fight against climate change claims for the research and development of clean and reliable energy sources. For this, significant efforts are being directed towards the progress of fusion energy. Thanks to the recent developments in high-temperature superconductors and materials technology, new layouts of fusion reactor devices are possible. In this context, the Affordable, Robust and Compact (ARC) fusion reactor design is under development at Commonwealth Fusion Systems and Massachusetts Institute of Technology - Plasma Science and Fusion Center, in collaboration with the ENI S.p.A. The reactor features demountable superconducting toroidal field coils and a replaceable vacuum vessel immersed in a fluorine lithium beryllium (FLiBe) molten salt blanket. The low-pressure molten salt cools the divertors and the blanket, shields the magnets and acts as tritium breeder and tritium carrier. Tritium must be recovered efficiently from the salt to fuel the reactor, minimizing its inventory and its migration outside the FLiBe loop. The current work is aimed at presenting a preliminary design of a Double-Wall Heat eXchanger (DWHX) to be installed in the ARC reactor primary cooling system. The component main characteristic is that the gap between the heat exchanger tube walls is filled with a flowing sweep gas that captures tritium, preventing its diffusion toward the secondary system. For the DWHX, thermal-hydraulic and tritium transport analyses were performed and discussed in the current paper. A supercritical water Rankine power conversion system was considered on the secondary side to carry out the calculations.

Keywords

Thermal-hydraulics, Tritium, Supercritical water Rankine cycle, FLiBe, ARC Fusion Power Plant.

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PS1-89 ABSTRACT-049d
F. Nuclear System Design

Overview and Origin of the STEP Design Point

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Spherical tokamaks have a number of features that make them attractive for a future fusion powerplant. Operating at a higher ratio of plasma-to-magnetic pressure (β) and higher elongation enables a smaller major radius of device and smaller diameter of poloidal field coils, potentially leading to a more cost-effective tokamak solution. Looking to apply these benefits, the Spherical Tokamak for Energy Production (STEP) programme has set out to deliver a UK prototype fusion energy plant, targeting 2040, and a path to commercial viability of fusion.

Identifying a design point for the STEP Prototype Powerplant (SPP) is a complex activity. Fusion powerplants consist of many highly integrated systems, and technology options for each cannot be selected in isolation. Equally, the limits and constraints that drive the SPP design will not necessarily be the same as for other fusion powerplant designs, and these need to be identified. An integrated approach needs to be adopted from the start of the design process to understand which combination of technologies work together and to allow the discovery of the key design drivers.

In this presentation we set out how we arrived at the SPP design point. We outline the methodology of selecting technology options, then using a combination of systems codes and fixed and free boundary equilibria, we quickly scope out preliminary integrated design options. The most promising of these are taken forward for further, more detailed, evaluation (including structures, tritium breeding ratio, heating and current drive, and power balance). Based on the findings of this analysis we present some of the key constraints we identified and how this led to the major architectural parameters being set.

Keywords

STEP, Spherical Tokamak, PROCESS, JETTO, Systems Design, Fusion Powerplant.

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PS1-90 ABSTRAC-0682
F. Nuclear System Design

Design and development of the fiber Bragg grating sensors for MITICA beamline

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Negative hydrogen Neutral Beam Injectors (NBIs) will be the workhorses of ITER reactor where they are foreseen to deliver up to 33MW of power to heat the fusion plasma. The MITICA (Megavolt ITER Injector and Concept Advancement) experiment at Consorzio RFX in Padova, Italy, is the full-scale prototype of the ITER NBIs. It aims to produce a beam of 40MW of negative hydrogen or deuterium ions extracted from a RF (radio frequency) source. The negative ions are then neutralized in a gas neutralizer and the remaining charged ions are deflected in the Residual Ion Dump (RID). After these steps, approximately 19MW of neutral atoms are foreseen to enter the tokamak duct and finally reach the ITER plasma or, in the case of MITICA, to be dumped on an instrumented calorimeter. The remaining 21MW of power are deposited on the beamline components, which will face heat loads as large as 13 MW/m^2. All the beam-facing components are designed to withstand these large heat loads, and a suite of sensors is used to measure temperatures and strains for component protection and beam characterization. Thanks to the large dielectric strength of the fibers, Fiber Bragg Grating (FBG) sensors were selected to monitor all these quantities on the beamline components in MITICA, especially on the components which are polarized with respect to the ground, such as the polarized panels of the RID. Furthermore, a set of FBG accelerometers was also designed to investigate the nucleate boiling conditions which will occur inside the actively cooled structure by assessing the vibrations of the structures. In this contribution we describe the design of the entire set of sensors and the acquisition chains, focusing on the constraints linked to the operation in a UHV (ultra-high vacuum) and nuclear environment and the foreseen operating range.

Keywords

FBG sensors, Neutral Beam Injectors, ITER.

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PS1-91 ABSTRACT-0752
F. Nuclear System Design

Applicability of the ITER-like EC HVPS design to the DEMO ECS

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ECH is presently the lone additional heating system selected on DEMO helping to the plasma ramp-up (resp. ramp-down) phases, as also during the burning phase for stabilisation purposes. The EC RF power is generated through sources, named gyrotrons (RF unit), which require several HV Power Supplies (HVPS) to convert the electrical input energy into microwaves.

The purpose of this work is to determine if the technology choices and the supply structure selected in the design of the EC HVPS of ITER, will remain applicable and relevant to the DEMO ECS requirements. The applicability will be evaluated for each EC HVPS (MPS, BPS, APS) used to the gyrotron operation, identifying the impacts of the DEMO more stringent constraints. This evaluation is based on a typical DEMO plasma pulse, selected as a reference to determine the RF power and pulses duration to be applied during each plasma phases. Then, considering the estimated output power of the future DEMO RF source (the development of such a source is still at its very preliminary phase), the number of RF units to be supplied has been determined as the result of a detailed RAMI analysis, including a margin for the redundancy.

The paper will summarize and discuss the results, including also a review on the impacts of the RF power demand at the interfaces with the plant, like the RF building auxiliaries and the AC electrical network.

Keywords

ECH, RF, Gyrotron, HVPS, ITER, DEMO.

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PS1-92 ABSTRACT-0be1
F. Nuclear System Design

Conceptual Design Study of a Large Bore Superconducting Test Facility Magnet, SUCCEX

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A 16 Tesla, large bore superconducting magnet for testing superconducting CICC(Cable-in-Conduit Conductor) is being designed in parallel with the design of the steady state Korea fusion demonstration reactor (K-DEMO) magnet system. The test facility magnet, named SUCCEX (SUperConducting Condctor EXperiment) was designed in 2014 and some design modifications of the magnet is being conducted since 2020. The peak magnetic field of the K-DEMO toroidal field coil is about 16 T thus the required background field of the conductor test magnet is expected to be over 15 T in a large bore of 1 m diameter. The background field will be added up with the K-DEMO conductor sample's self-field. To reach the target magnetic field, the high current density Nb3Sn strands (Jc > 2600 A/mm2 at 4.2 K, 12 T) is applied in the design of 2 types of CICC. The modified SUCCEX magnet is consist of 2 concentric split pair solenoids, high field inner coil (IC) and low field outer coil (OC). They connected in series with each other and the operating current is about 35 kA. In this study, the overall design concept related to the results of electro-magnetic, structural and thermo-hydraulic analysis are presented.

Keywords

Fusion magnet, Cable-in-conduit conductor, high field magnet, conductor test facility.

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PS1-93 ABSTRACT-0cff
F. Nuclear System Design

New Neutron Scanner for Container Inspection based on Nuclear Fusion

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Current detection systems for containers and large cargo at ports and airports are unlikely to detect explosives or drugs. New legislation being introduced in both the United States and Europe requires detection of such items. Current systems are based on very high energy X-rays. However, heavy chemical elements absorb X-rays. Iron and other heavy elements are OPAQUE to X-rays, showing up as shadows in the image. In contrast, X-rays pass through materials composed of light elements, such as drugs and explosives, so these illegal substances do not appear in the image. Compounds rich in light elements, such as Hydrogen, Nitrogen and Carbon, are OPAQUE to neutrons, so they show up as shadows on a neutron detector. However, neutrons pass easily through HEAVY METALS so they have no difficulty passing through the surface of the container. Hydrogen and other light elements ABSORB neutrons, emitting gamma rays of characteristic energy for each element.most explosives are rich in nitrogen, which can be detected through the peak of the gamma spectrum, in the capture of thermal neutrons by nitrogen. The same is true for other illegal substances. Gamma spectroscopy provides information on the composition of elements in seconds.
Neutron technology, which is more efficient, safer and more reliable than X-rays, could reduce costs per container by around $40 per container.

Most scientific researchers are focused on the energy application of nuclear fusion. However, nuclear fusion technology has many other applications that are much easier to develop. At I3M we are developing a new neutron scanner based on nuclear fusion for container inspection. We present the design of the neutron generator based on deuterium-tritium reaction, the neutron optics for the scanner, the neutron detector panel and the gamma rays spectrometers for element identification.

Keywords

Neutron Scanner.

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PS1-94 ABSTRACT-0d5b
F. Nuclear System Design

Neutron Source Modelling for Commissioning Hydrogen Plasmas in Tokamaks

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It is currently envisioned that plasma operation in ITER Pre-fusion power operation phase (PFPO) will be based on hydrogen and helium plasmas, in order to study plasma start-up, commission the machine and diagnostics, and perform early physics studies. These plasmas are in principle not expected to produce radiation, e.g. neutrons and gammas. However, because of the planned plasma heating scenarios, like synergistic NBI+RF or 3-ion RF schemes, energetic particles in the MeV energy range are expected to be produced. Such particles, can trigger fusion reactions with the beryllium intrinsic metallic wall impurities, producing neutrons and gammas ($^{9}Be(p,n\gamma)^{9}B$ and $^{9}Be(^{3}He,n\gamma)^{11}C$). The neutron yields are expected to be smaller compared to the later power operation phase in a deuterium-tritium plasma – 10^{14} n/s in PFPO compared to 10^{21} n/s in ITER D-T plasma. Nevertheless, neutrons emitted from such plasmas offer the possibility of commissioning diagnostic systems before power operation begins.

The neutron emission characteristics of PFPO plasmas differ significantly from full burning plasma discharges expected in DT plasma. To computationally support the commissioning of diagnostic systems, a novel computational tool has been developed taking into account the discharge specific plasma conditions. The tool is based on combining results of detailed plasma physics simulations and Monte Carlo neutron transport simulations.

This paper will present the results of using the developed computational tool to generate neutron and gamma-ray sources emitted from proton and beryllium reactions. The generated sources will be based on plasma conditions of JET discharges performed in the record 2021 DT experimental campaign and 2022 helium experimental campaign. In both experimental campaigns, fast protons were produced due to synergies between different heating systems. Calculated fission chamber counts and dosimetry foil activation for helium campaign discharges are compared with measurements, as the reactions studied represent the only source of neutrons in the plasma.

Keywords

Plasma physics, TRANSP, MCNP, neutronics.

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PS1-96 ABSTRACT-12b3
F. Nuclear System Design

Study on the Shutdown Dose Rate Calculation Method based on Nuclide Activation

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In the radiation safety analysis, it is necessary to determine the dose rate of decay photons in the surrounding environment released by the activation of component materials during and after the operation of nuclear facilities. At present, the two-step shutdown dose rate calculation method is relatively accurate, but it takes a long time and requires a large amount of computer memory; however, the conventional one-step shutdown dose rate calculation method can only deal with the one-step decay process of some nuclides, but cannot deal with the multi-step reaction and cascade decay, resulting in inaccurate calculation results. In this paper, an improved one-step shutdown dose rate calculation method based on nuclide activation is developed. This method first calculates the decay photon production coefficient of each nuclide; then the photon dose rate is calculated by combining the irradiation scheme and the decay photon production coefficient in the neutron- photon coupling transport calculation. The advantage of this method is that it can deal with multi-step reaction and cascade decay in the shutdown dose rate calculation process, and there is no need to repeat the activation calculation when the material is adjusted. This method is implemented base on TopMC (advanced version for SuperMC) and a typical benchmark example is used to test and verify the method, and the calculation results show that the method has good accuracy.

Keywords

SDR Calculation, Nuclide Activation, Improved D1S, TopMC.

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PS1-97 ABSTRACT-7a68
F. Nuclear System Design

The ITER Radial Neutron Camera ex-port system: nuclear analyses in support of its design development

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The Radial Neutron Camera (RNC) is a diagnostic system located in the ITER Equatorial Port 1 (EP01) composed by two sub-systems (i.e. In-port and Ex-port RNC [1]) probing a poloidal section of the plasma through a set of fan-shaped Lines of Sight (LOS). The RNC is designed to provide a time resolved measurement of the neutron and α particles source profiles and of the total neutron source strength, through the application of reconstruction techniques to the line-integrated neutron fluxes.

The Ex-port sub-system is composed by sixteen LOS distributed in two different toroidal planes and enclosed in a massive shielding unit, extending from the EP01 closure plate through the Port Interspace, up to the Bioshield Plug. Neutrons, generated in the plasma core, stream through dedicated optical paths hollowed out in the central EP01 diagnostic shielding module and reach the detectors units located at the end of collimating structures. The detector unit of each LOS contains one ^{4}He gas scintillator, one plastic scintillator as well as a single Crystal Diamond (sCD) matrix.

The paper presents the outcomes of the neutronic analyses, performed on the latest design of the ex-port sub-system, aimed evaluating the nuclear loads to be withstood by its structural elements, detectors and associated components during Normal Operating Conditions. Moreover, the neutron fluxes and spectra at the detectors positions have been evaluated in order to verify that the signal-to-noise ratio is compliant with the expected measurement performances of the diagnostic system. Finally, issues related to the activation of the detector units as well as the effect of radiation streaming in the Port Interspace and Port Cell areas are discussed and addressed.

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Keywords

Neutron diagnostics, radial neutron camera, ITER, MCNP.

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PS1-101 ABSTRACT-14cb
G. Safety Issues and Waste Management

Development of radiation sources based on CAD models for nuclear analysis of IFMIF-DONES lithium loop

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IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO Oriented NEutron Source) will be a neutron source designed to irradiate materials to be used in future fusion power reactors such as DEMO. The facility is based on a deuteron beam impinging onto a liquid lithium jet to generate a neutron flux suitable for material irradiation. Lithium and Corrosion Products will get activated and produce Be-7 and Activated Corrosion Product (ACP). These products will distribute along the lithium loop, both dissolved in Li and deposited locally in the cold section. Due to the complex Li loop piping design, this complex radiation source should be properly represented to perform radiological safety studies.

In this paper it is presented a new tool to generate radiation sources for MCNP code based on CAD models. Complex sources can be represented by choosing the CAD solids in which each activation product is located, which will be taken as the actual source generation location. Then the CAD model will be associated with the concentration of the Be-7 and ACP, as well as deposition masses. This information can be set directly in the CAD model or by means of plain text processing. Using the CAD model facilitates the source definition and modification.

Tool applicability is demonstrated with the lithium loop source definition and transport located within the Lithium Loop Cell (LLC). This room accommodates most of the piping and containers for the primary heat removal loop. Radiation maps within this room are finally obtained by transport simulation of the gamma rays inside LLC.

Keywords

Source definition, fusion neutronics, IFMIF-DONES, CAD-based simulation.

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PS1-102 ABSTRACT-38f3
G. Safety Issues and Waste Management

Tritium Behavior in Soil and Mineral Rock Components used for Plant Cultivation

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The proposed deuterium-tritium fusion reactor is predicted to generate a large amount of tritiated water. From a viewpoint of safety, it is important to understand tritium behavior in environment assuming an accidental release of the tritiated water and water vapor. As a beta emitter, tritium is not harmful outside the human body, however it is of radiological significance when internally deposited in biological tissues. Studies on tritium behavior in plant tissues has been conducted with varying results. That on its behavior in soil is limited and requires further study.

In this experiment tritium behavior in some soil and mineral rock components for plant cultivation such as peat, dolomite and vermiculite were investigated. Tritium release experiments from tritium contaminated samples were done with heating to 1000ºC at a ramping rate of 5 ºC/min under 100 cc/min Ar flow. Samples were combusted using 20 % O_2/Ar gas and subsequently purged with wet Ar gas to remove the residual tritium from the piping. Two bubblers (for HTO and HT respectively) interconnected with a platinum catalyst, were filled with 12 ml distilled water to trap released tritiated gases. Bubblers were replaced approximately every 5 mins with a corresponding 25 ºC increase in temperature.

1 ml each was sampled for liquid scintillation counting at a count time of 5 mins/sample. Most tritium was released below 200 ºC in the samples. Vermiculite, known for its high moisture retaining capacity showed significant releases around 700 ºC and 800 ºC. Tritium percolation, capture and release experiments are being conducted on lime, dolomite, perlite, vermiculite and peat for further understanding.

Keywords

Soil, HTO, HT, tritium release, mineral rock.

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PS1-103 ABSTRACT-5540
G. Safety Issues and Waste Management

Application of a Multiphysics simulation tool to an in-box LOCA in the breeding unit of the WCLL-BB of LIFUS5/Mod4

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The in-box LOCA is one of the main critical points of the Water Cooled Lithium Lead Breeding Blanket design. A significant effort has been put over the years by the scientific community to study the interaction between Lithium-Lead and Water and the impact that such event could have on the overall plant. This investigation has been carried out both experimentally and numerically. The presence of two working fluids with multiple phases and the need to simulate also the chemical interaction between Lithium-Lead and Water led to the choice of the code SIMMER, which was originally developed to analyse Liquid Metals accidental scenarios. However, the need to simulate the overall Lithium-Lead loop limits the use of SIMMER as standalone code since it is not suitable to be used for complex pipelines, whereas RELAP5, a well-known thermal-hydraulic system code, is able to simulate this kind of geometries. Therefore, the simulation of the complex phenomena involved in an in-box LOCA scenario requires a combination of the characteristics of both codes. This work presents the application of a coupling methodology between the two codes in order to exploit their characteristics. At first it is shown that the two codes, through the developed coupling technique, are able to reach a high level of synchronisation in a steady-state condition and then the technique is also successfully tested in a fast transient in-box LOCA scenario. The in-box LOCA is simulated in a geometry resembling the ITER TBM Breeding Unit nodalized with SIMMER-IV whereas the piping system of LIFUS5/Mod4 is simulated with RELAP5/Mod3.3. The pressure wave propagation due the tube rupture is analysed throughout the whole Lithium-Lead loop; it is also shown and discussed how this coupling tool can be used to support the design and safety analysis of this facility.

Keywords

In box LOCA, Lithium-Lead/Water Interaction, SIMMER, RELAP5, Multi-Physics.

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PS1-104 ABSTRACT-07b4
G. Safety Issues and Waste Management

Fast Shutter optioneering study for the ITER Disruption Mitigation System

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ITER needs a Disruption Mitigation System (DMS) for protection from the consequences of plasma disruptions during high-power operations. The current DMS for ITER is based on the Shattered Pellet Injection (SPI) technology. This works on the basic principle of a cryogenic pipe gun with a several-meter-long barrel. It is important to avoid any contact between the pellet and the internal surfaces of the flight line to minimize the risk of premature pellet breakage. Therefore, the internal dimensions of the flight tube of the SPI are relatively large compared to the pellet and hence providing a bypass for propellant gas to pass the pellet resulting in a significant quantity of propellant arriving in the plasma ahead of the pellet. This will cause an undesirable instability in the plasma and compromise the effectiveness of disruption mitigation. The described events are appearing on a scale of a few milliseconds. One possible solution for this issue could be the use of a fast-acting shutter in the flight line, which will close after the pellet has passed through it to hold back the propellant gas. Due to the environmental requirements, the shutter needs to close an approximately 40 mm aperture within a few ms, the shutter has limited space and limited access and acts under a magnetic field, therefore it should endure a high number of cycles and should perform well in a very demanding situation. This paper describes the optioneering study of the shutter and possible solutions for the demanding criteria. The laboratory testing of the actuator [1] and the final mechanical design [2] are also presented at the conference to give a whole picture of the project.

[1] I. Réfy et al *at this conference*
[2] Csiszár et al *at this conference*

Keywords

DMS, SPI, fast shutter, optioneering.

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PS1-105 ABSTRACT-7ddf
G. Safety Issues and Waste Management

ITER DMS Fast Shutter development and laboratory testing

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The ITER Disruption Mitigation System (DMS) [1] is based on shattered pellet injection [2] (SPI), which accelerates a large hydrogen or neon pellet and shatters it prior to the entrance into the plasma, creating a plume of smaller pellet shards. The pellets are accelerated by high pressure hydrogen gas which can overtake the pellet, triggering an early disruption and deteriorate the disruption mitigation efficiency. A propellant gas recovery chamber is utilized in which most of the propellant gas can be retarded for some time, however, a certain fraction of the gas will still arrive at the shatter exit ahead of the fragments.

Each ITER SPI will utilize a fast shutter after the recovery chamber to block the path of the propellent gas after the pellet has passed. The piston of the valve has to be accelerated to tens of meters per second in order to close the 40 mm orifice in a few milliseconds and decelerated after the closure to avoid high velocity impact at the end position. The device must survive several thousands of cycles since the access for maintenance will be very limited. The setup will need to operate in 400 mT external field, high neutron dose rate, has to be tritium compatible as a part of ITER main vacuum system, and the space restrictions require a compact design.

This contribution describes the development and the laboratory testing of the ITER DMS Fast Shutter from the physics design, through the electromechanical prototyping and the model validation. The optioneering [3] and the detailed mechanical design [4] are presented in separate contributions at this conference.

[1] Luce *et al* 2020 IAEA Fusion Energy Conference, Nice, TECH/1-4Ra.
[2] R. Baylor *et al*2019 *Nucl. Fusion* **59** 066008
[3] Zsákai et al *at this conference*
[4] Csiszár et al *at this conference*

Keywords

ITER DMS, fast shutter, valve, laboratory testing.

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PS1-106 ABSTRACT-17b6
G. Safety Issues and Waste Management

Investigation of a LaBr(Ce)-based Spectrometer at the KSTAR Tokamak with mixed neutron-gamma radiation fields

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The special characteristics of LaBr(Ce) scintillators have found in many literatures. Since LaBr(Ce) scintillators have excellent energy resolution, high count rate capability and good linearity, they enable us to provide useful information on a specific purpose for plasma diagnostics in fusion reactors with mixed neutron-gamma radiation fields.

For this reason, LaBr(Ce) scintillators have proposed for fast ion diagnostics and α particle diagnostics in the KSTAR tokamak as well as other tokamaks including ITER. Measurements with LaBr(Ce) scintillators are based on a gamma-ray spectroscopy measuring neutron-induced gamma rays on LaBr isotopes. In general, LaBr(Ce) scintillators are able to find more peaks and find them faster than the NaI(Tl) detectors.

A LaBr(Ce) scintillator-based spectrometer for monitoring combined neutrons and gammas at the KSTAR tokamak has developed. The spectrometer consists of all-in-one compact system. The development of LaBr(Ce)-based compact spectrometer and its performance on the deuterium-deuterium fusion plasmas in the KSTAR tokamak have demonstrated at the conference.

Keywords

KSTAR, D-D Plasma, mixed neutron-gamma fields, LaBr(Ce) spectrometer, Tokamak

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PS1-107 ABSTRACT-1fd7
G. Safety Issues and Waste Management

Establishing a Dependable Coolant Purification System Design for Fusion Reactors: Achieving Objectives through RAVAD Testing

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In a fusion reactor, the Coolant Purification System (CPS) is responsible for separating tritium and impurities from the Helium Cooling System. CPS comprises several columns, including an oxide bed that oxidizes Q2 to Q2O, a Molecular Sieve Bed (MSB) that adsorbs Q2O, a reduction bed that reduces Q2O back to Q2, and a heated getter that removes other impurities. To the design the system, validation is required for each process in the CPS. To accomplish the study's objectives, the Research Apparatus for Vapor Adsorption and Desorption (RAVAD) was established. Specifically, the study aimed to achieve three objectives: (1) completion of MSB testing and analysis of the results, (2) expansion of the test facility to include oxide bed and reduction bed experiments, and (3) performance of coupled tests of CPS components with the upgraded facility. The study's ultimate goal is to establish a dependable CPS design.

Keywords

Coolant Purification System, Helium Coolant System, oxide bed, reduction bed, molecular sieve bed, vapor adsorption.

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PS1-108 ABSTRACT-2fa7
G. Safety Issues and Waste Management

Prospects for Melt Refining of Radioactive Waste from Fusion Reactors

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One of the principal safety and environmental goals of nuclear fusion is to minimize (or eliminate completely) the generation of long-lived radioactive waste. This is likely possible in future fusion reactors via the judicious us of reduced-activation structural materials. Even so, the resultant volume of activated materials that would require disposal as low-level waste will inevitably be rather large. While low-level waste is simpler to dispose of than high-level waste (shallow land burial, vs. deep geologic repositories), the volume of waste resulting from a fleet of future fusion reactors would certainly require the construction of new low-level waste disposal facilities, which may encounter significant public opposition and/or incur significant cost.

Recycling of such materials has been suggested as a potential alternative. In addition to re-use of components and materials that are only slightly activated or contaminated, reprocessing of the highly activated components has been suggested. Melt refining is one possible method of doing so; this technique has been used to reduce the radionuclide content of large volumes of contaminated steel from fission reactors, often to below "clearance" levels at which the material can be released from regulatory control (though this is not a generally accepted practice in the United States).

Partitioning of radioactive materials during melt refining of steel is well understood. Volatile materials are removed in dust or fumes; those with a higher oxidation potential than iron accumulate in slag; and those with a lower oxidation potential remain in the metal. In this work, we consider the process applied to low-activation ferritic/martensitic steel with activation product inventories typical of fusion reactor components at their end of life (including those resulting from likely impurities). The quantity and nature of the resultant recycled product and waste streams is calculated in order to assess the overall effectiveness of the process.

Keywords

Radioactive Waste, Waste Recycling.

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PS1-109 ABSTRACT-3748
G. Safety Issues and Waste Management

DEMO Toroidal Field Coil Fast Discharge Unit Integration Studies

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The Fast Discharge Units (FDUs) of the superconducting (SC) toroidal field (TF) coils in the European demonstration fusion power plant DEMO warrant the machine integrity over its full lifetime, even in severe failure events such as SC coil quenches or any other plant events requiring the safe TF magnet system discharge. The FDUs are considered as safety important class (SIC) components that need to discharge about 125 GJ of energy stored in the DEMO TF coils in tens of seconds into dump resistors outside of the tokamak building. For this reason, the FDUs must be extremely reliable for the purpose of commutating in short time (~1 s) currents of about 70 kA to discharge the TF magnets safely. Malfunctions of the FDUs must be avoided at all means.

The TF FDUs Circuit Breakers (CBs) shall be installed in the lower level of the tokamak to minimize the length of the connecting busbars. The FDUs integration is challenging because of the high neutron and gamma radiation and stray magnetic fields of the tokamak.

Since in DEMO the neutron fluence over lifetime is much higher than in ITER, the problems of using FDUs with electronics and semiconductor switches was expected to be so severe that their integration have been considered from the beginning of the DEMO project. Sufficient shielding or possible re-positioning of the sensitive FDU components compared to ITER are being investigated, to reduce the neutron fluxes and neutron and gamma ray fluences. Alternative concepts e.g. fully mechanical CBs are studied in the DEMO Work Package Plant Electrical System in parallel.

This paper presents the CAD integration work on the DEMO TF FDUs supported by neutronics assessments. It is assumed the same FDU technology as in ITER. The magnet feeder´s integration is commenced at the same time.

Keywords

Fast Discharge Units, Tokamak Integration, Toroidal Field Coils.

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PS1-110 ABSTRACT-4386
G. Safety Issues and Waste Management

Shielding Optimization to Reduce the Radiation Field in the Accelerator Systems of IFMIF-DONES

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IFMIF-DONES is one of the key facilities in the roadmap towards nuclear fusion. Its main objective is the generation of a high energy neutron source, capable of producing a neutron radiation field like the one that will exist in future nuclear fusion reactors.

To this end, a 125 mA deuteron beam, accelerated up to 40 MeV, will impact on a flowing liquid lithium target, resulting in an intense neutron source. In addition to this neutron source, other radiation sources will arise because of the interactions between deuterons and the beam facing accelerator components. Such radiation fields in the Accelerator Systems will not be so intense as the ones in the Test Cell region, but some negative effects caused by radiation will appear in the accelerator components, such as damage to equipment or the generation of residual radiation sources. The characterization and mitigation of all the radiation sources, along with the production of radiation maps, are crucial for a safe operation and maintenance of the facility.

In this study, the optimization and/or implementation of some radiation shields located in critical areas is considered to reduce the biological dose and the dose to silicon, during operation, and the residual dose during shutdown. Three scenarios are assessed: i) shielding optimization of the high-power beam dump for the accelerator commissioning, ii) shielding optimization for the HEBT scraper and the fast safety isolation valves, and iii) implementation of additional shielding to reduce the radiation transmission through HVAC apertures. Radiation maps of the biological dose and the dose to silicon are computed in the IFMIF-DONES main building, showing the impact of such shielding elements.

Keywords

IFMIF-DONES, RADIATION MAPS, RADIATION SHIELDS, BIOLOGICAL DOSE, DOSE TO SILICON.

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PS1-111 ABSTRACT-da07
G. Safety Issues and Waste Management

New electrode supports in Neutral Beam Injection for HV breakdown incidence reduction risk

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In the accelerators for Neutral Beam Injection (NBI) in fusion applications a very large ion currents (40 A) at High voltage (1 MeV) will be generated and transported to the neutralizer before to reach the tokamak. The electrical technologies involved to keep the accelerating and extracting electrodes at high voltage required different electrical technologies and complex interfaces. Integrated tests of the power supplies that achieved 700 kV in stable operation and further tests by using an auxiliary power supply for the HV withstanding capability of the plant at 1 MV have been already presented. In both tests, however, breakdowns in HV part of the accelerator occurred. The full voltage holding capability of the accelerator has been also achieved but breakdowns close the accelerator grids were very frequent events and the system must be capable of handling without damage to the source grids. In this contribution a different (and complementary) way to mitigate the HV part breakdown problems will be discussed. In agreement with the Paschen low, the breakdown risk decreases sensitively by increasing the vacuum level. The proposal for some modifications in the grids supports design could increase the vacuum conductance in the NBI HV part and then increase pumping velocity enhancing nearby the vacuum level. On the other hand, the breakdown risk increases also in presence of equipotential lines with high distortions that could give high electric fields. The new support grids design aim also to reduce those distortions. Preliminary electrostatic potential simulations of the new grid supports and further discussions with respect to the breakdown risk increase will also be given.

Keywords

Neutral Beam Injection, HV breakdown.

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PS1-111b ABSTRACT-91d9
G. Safety Issues and Waste Management

Comprehensive study of ITER Hot Cell radiation conditions for design optimisation within ALARA approach

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The Hot Cell is supporting the ITER project by contributing to the tokamak components maintenance and to the management of radioactive and contaminated waste in a secure and controlled environment. Although DT neutrons generated during tokamak operation could reach the nearby Hot Cell facility, the main radiation sources within are tokamak components transferred to the Hot Cell which have been previously activated by neutrons when in place in the vacuum vessel during plasma operation. The most relevant among them are (i) In-Vessel components, such as port plugs, divertor cassettes or first wall panels, (ii) Vacuum-Vessel dust spread on the remote-handling tools used to extract In-Vessel components and, (iii) Activated Corrosion Products present in the water-cooling systems. The Preliminary Design Review performed in 2021 on the Hot Cell has led to investigate on possible design optimisation notably on civil structure while respecting radiological limits and enabling Hot Cell functions. In this study, we present a comprehensive computational nuclear analysis of the radiation conditions within and at the boundaries of the Hot Cell Facility and an evaluation of the Occupational Radiation Exposure linked to its operation. More than 50 different radiation sources configurations in the Hot Cell facility have been independently addressed. A 2-step ALARA approach has been followed for each configuration, considering both the 2021 baseline design and a modified model with dedicated optimisation measures.

Keywords

ITER, Hot Cell, ORE.

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PS1-111c ABSTRACT-e7e5
G. Safety Issues and Waste Management

Radwaste analysis and package optimization with the radwaste calculator and application to the ITER 1D benchmark and the IVVS ITER system

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A methodology to evaluate the neutron activation, categorize as radwaste and optimize the packaging for disposal of activated components is presented. The methodology obtains results like specific activity and radwaste relevant parameters in a fine 3D mesh superimposed over the geometry of interest and is able to analyze the complex models common in the nuclear fusion analyses like those of the ITER project. A set of software tools, which perform the post-processing of the large amount of data generated, and a graphical user interface have been developed for this purpose and are used along state-of-the-art radiation transport and activation codes. The 3D nature of the results allows the organization of packages of activated materials in a way that minimizes the amount of mass classified with higher radwaste levels and is subject to more stringent regulations and expensive disposals. Application case studies of the methodology are presented for the idealized ITER 1D benchmark exercise and for the real IVVS ITER system. The IVVS (In-Vessel Viewing System) is especially suited for this type of analysis as its elongated shape pointing towards the plasma results in a significant gradient of neutron flux and activation levels, and therefore radwaste categories. It is demonstrated how the methodology enables the reduction of the total higher radwaste level mass by optimizing the packaging of the IVVS radwaste.

Keywords

Radwaste, neutronics, IVVS, ITER.

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PS1-112 ABSTRACT-2373
H. Models and Experiments for FNT

Construction of GVR Weight Windows Mapas from Very Low Density Transport Simulations

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The assessment of shielding designs for nuclear fusion reactors via Monte Carlo codes requires the usage of global variance reduction (GVR) techniques so that population is preserved, and low statistical uncertainty is achieved throughout the whole reactor in an acceptable amount of simulation time. Weight windows map generation methods make their limits proportional to the radiation flux, which is unknown. These fluxes are usually calculated at an artificial low density to allow for wider propagation, and then the actual attenuation profiles are reconstructed by raising the obtained values to an exponent. However, the density difference between them cannot be too pronounced because the reconstruction of the actual flux from a lower density estimate is no longer accurate, resulting in the need of an iterative increase of density in repeated calculations.

The purpose of this work is to develop an algorithm for reconstructing radiation fluxes from very low density radiation transport calculations, so that one or a few runs can be used to compute a weight windows map in the whole domain. The non-linear behavior of flux as density increases due to the effect of collisions on directionality has been found as the source of discrepancy for the exponential corrections. With the new algorithm, the density steps in the iteration process have been greatly increased, leading to reduced computation time requirements for weight map calculation.

Keywords

Global Variance Reduction, Shielding Analysis, Weight Windows, Weight maps.

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PS1-113 ABSTRACT-32ae

H. Models and Experiments for FNT

A Reduced-Order Model to Estimate First Wall Particle and Heat Fluxes for Systems Codes

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While most particles ejected from the confined region in tokamak plasmas are directed towards the divertor, some are distributed along the First Wall (FW) of Breeding Blankets (BBs) by following magnetic field lines in the so-called far Scrape-Off Layer (far-SOL). Even though these fluxes are not the only phenomenon contributing to the heat deposited on the FW (e.g. photons, charge-exchange), their impact must be assessed since the total heat fluxes expected on FW armor (up to 1 MW/m2) are a challenging design issue. At the same time, these fluxes also contribute to particle implantation in the FW armor (ion bombardment), which can impact systems-level analyses such as tritium permeation into the FW coolant, tritium retention in BB steel, and effusion fluxes during pump-down.

Current tokamak plasma physics prescribes that the continuous production and expulsion of blobs from the confined plasma region is a main phenomenon contributing to these far-SOL fluxes. Simulation tools based on simplified turbulent transport models, such as the TOKES code, have been under development to provide BB engineers with design-relevant information, e.g. prediction of ”hot spots” on the FW. Unfortunately, such tools tend to run in timescales that are prohibitive for incorporation in Systems Codes (SCs), which would enable analyses at a systems level.

This work presents a Reduced-Order Model (ROM) built with results from the TOKES code, to serve as a surrogate model for coupling in SCs. The ROM was developed with Principal Components Regression (PCR) and k-fold cross-validation, applied to the results after transformations based on rational powers of the TOKES inputs (blob temperatures, densities and ejection speeds). Model selection was performed with Kullback-Leibler Divergence (KLD), and validation, with withheld cases. Main results include a relatively low number of modes to represent more than 90% of the data variance.

Keywords

DEMO, far scrape-off layer (SOL), TOKES, Systems Codes, Nuclear Fusion Technology.

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PS1-117 ABSTRACT-0dd9
H. Models and Experiments for FNT

Neutron Emission characterisation of IPR 14 MeV neutron generator

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Accelerator based neutron sources are being developed worldwide due to their wide applications. Institute for Plasma Research, India has developed an accelerator based high yield 14 MeV neutron generator to perform fusion related benchmark experiments. This neutron generator is designed to produce 10^{12} neutrons per second in continuous mode as well as pulse mode. In, accelerator based neutron generator, emission of neutrons is not isotropic due to target geometry and angular dependency of reaction cross-section. To use such neutron generator for various experiments, detailed and accurate description of neutron source emission and distribution is required [1]. To accomplish this, neutron generator is characterised using experiments and simulations.

Energy and angular distributions of neutrons are measured using well calibrated single crystal diamond detector [2]. Experimental data of neutron distributions are used to generate source neutron definition for Monte Carlo simulation. An accurate model of Titanium Tritide (TiT) target and its holder geometry is prepared by combining both experimental and simulation data [3].

Finally to validate the simulation model, simulated neutron spectra is compared with spectra measured by foil activation technique. This paper discusses the methodology adopted to characterise emitted neutrons and its results.

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Keywords

Neutron generator, Characterisation, Anisotropy, Monte Carlo, TiT.

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PS1-119 ABSTRACT-19f1

H. Models and Experiments for FNT

Experimental validation of the fluid solver for SPIDER (FSFS2D)

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The experimental fusion reactor ITER will be heated by injection of fast neutral beams generated by acceleration and neutralization of negative D^- ions. The prototype negative ion source used for this purpose (SPIDER), constructed at the Consorzio RFX (Padua, Italy), consists in driver volumes where radio-frequency (RF) power is inductively coupled to the plasma electrons, and in an expansion chamber containing a magnetic filter (MF). The basic physical and numerical principles of a fluid model of this source are presented. The model implemented into the numerical code FSFS2D gives self-consistent two-dimensional description of the source, including neutral gas flow, plasma chemistry, RF coupling in the source driver and plasma transport through the magnetic filter. Different particle species are described by separate continuity equations and the electron temperature is governed by the electron energy balance equation. The particle fluxes are found from momentum equations neglecting the inertia terms (drift-diffusion approximation). The electrostatic coupling between electrons and ions is described by the Poisson equation. The numerical method is based on finite volume approximation and 9-point discretization is used to account for the anisotropy due to magnetic field. The semi-implicit numerical solver allows for large time steps (> 1000 x explicit time step) producing steady-state solution in a reasonable time (few hours for a typical mesh of 100x100 points).

An important element in the development of the numerical framework is the validation of the code results against the experimental data. For this aim, a series of numerical simulations have been performed and compared with the experimental data from the SPIDER experimental campaigns, including a cesium operation.

The paper presents critical assessment of the validation results and outlines the necessary code/model enhancements required to improve the predictive capability of the FSFS2D code.

Keywords

NBI, negative ion sources, modelling.

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PS1-120 ABSTRACT-1ae4
H. Models and Experiments for FNT

Results from a Prototype Testing Phase of the CHIMERA Heating System

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The CHIMERA (combined heating and magnetic research apparatus) test device, which is to be a major part of the UKAEA fusion technology facilities, includes a large-scale heating system to deliver a heat flux of at least 0.5 MW/m^2 across the entire surface of a module under test. This heat flux simulates the thermal radiation from the anticipated ITER and EU-DEMO plasma experiments. The design requirements for this system are challenging, it must operate in a combined static and pulsed magnetic field environment and transmit 400 kW heat to the module under test.

To realise this heating system, a design using a modular array of bespoke heating modules is proposed and an advanced heating element design constructed from high-purity graphite and coated in titanium carbide has been developed and tested under vacuum conditions. This article details the initial phases of prototype testing in which suitable materials were selected, element design created and experimentation completed, to verify its use in conditions representative of CHIMERA prior to final element production.

Emissivity measurements were carried out on the element using a dual-wavelength pyrometer, which was also used to determine the relationship between element temperature and reference thermocouples. At the anticipated element surface temperatures of 2000°C and above, evaporation of the element material and supporting hardware must be minimised, hence precise weighing of the elements was completed using analytical scales to verify the evaporation requirement of <5.5 $mg/m^2/hr$. In order to determine if the design delivered a uniform incident heat profile, an infra-red (IR) thermal imaging camera was used to inspect the element. Combined with thermal analysis, this allowed the uniformity to be quantified against the target requirement. Further testing is ongoing to determine long-term performance of the heater modules under continuous and cyclic loading.

Keywords

Fusion technology, high heat flux, plasma facing, fusion engineering, test facility.

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PS1-121 ABSTRACT-8b16
H. Models and Experiments for FNT

Beam emission spectroscopy modeling on the ITER diagnostic beam

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The application of neutral beam injection (NBI) for plasma heating and diagnostic purposes is a near-universal practice in magnetically confined fusion devices. This is also true for ITER, with two large-scale heating NBIs, and a smaller, diagnostic hydrogen NBI planned for the machine.

The diagnostic neutral beam will be observed with both charge exchange recombination spectroscopy systems (CXRS)[1], providing information about the impurities, the temperature, and the rotation of the plasma, and a motional Stark effect (MSE) diagnostic system[1], measuring the confining magnetic field, the plasma current, and the q-profile.

However, it has been proposed that fluctuation beam emission spectroscopy is also to be performed, measuring the plasma density profile and its fluctuations over time. A feasibility study of such a system integrated with the pedestal CXRS system has been carried out.

One of the synthetic diagnostic codes the study relies on is CASPER[2], designed to simulate the visible spectroscopic observations of magnetically confined fusion plasma. This code utilizes the Cherab & Raysect[3] framework to assemble scenes for a ray-tracing simulation. For the simulation of beam emission, the recently developed MSE capabilities were used. The inclusion of MSE in the calculations was critical for the differentiation from background plasma emission since the observation is almost perpendicular to the beam, leading to weakly Doppler-shifted beam emission.

The study shows that despite low signal-to-background ratio, caused by the small Doppler shift and strong background, larger fluctuations would be detectable. Additionally, measurement statistics would be further helped by the long duration of ITER discharges.

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

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[3]M. Carr, et al, Rev. Sci. Instrum. 89, 083506 (2018)

Keywords

Beam emission spectroscopy, motional Stark effect, synthetic diagnostics, modeling, CASPER, Cherab, ITER.

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PS1-123 ABSTRACT-21c1
H. Models and Experiments for FNT

Digital image correlation for modal analysis of mock-up blanket components

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Thermal-hydraulic systems in tritium breeding blankets experience fluid-induced vibrations (FIV), which promote high-cycle fatigue and fretting, potentially reducing the safe operating lifetime of these components. FIV-induced loading significantly increases when FIV frequencies match the resonant frequencies of components such as double-walled tubes (DWTs) in tritium breeding blankets. To ensure these resonant frequencies are avoided in the design of these components requires experimental vibration testing of component mock-ups to identify these frequencies and associated mode shapes.

Digital image correlation (DIC) permits non-contact, image-based measurement of surface displacements, allowing data-rich modal analysis of components under a variety of conditions. In this work, we use high-speed cameras to carry out DIC for vibration testing of a Grade 91 U-bend cooling tube, with and without internal turbulent water flow. From this experimental data we identify the natural frequencies and associated mode shapes for each condition along with the structural damping response. We then use a digital model of the experiment to perform a thorough uncertainty quantification on the output natural frequencies and damping coefficients accounting for various sources of systematic and random experimental errors.

The experimental procedure developed here is applicable more widely to transient structural responses to magnetic loading, and for seismic analysis, a safety requirement for all nuclear installations. These experiments will build capability for future transient loading experiments to be carried out under high temperature and magnetic fields using the combined heating and magnetic research apparatus (CHIMERA) at the Fusion Technology Facility (FTF) in Yorkshire.

The application of neutral beam injection (NBI) for plasma heating and diagnostic purposes is a near-universal practice in magnetically confined fusion devices. This is also true for ITER, with two large-scale heating NBIs, and a smaller, diagnostic hydrogen NBI planned for the machine.

Keywords

Digital image correlation, modal analysis, tritium breeding blankets.

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PS1-124 ABSTRACT-2575
H. Models and Experiments for FNT

Electromagnetic modelling of the central column for the STEP programme

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Fusion promises to be a safe, low carbon and sustainable part of the world's future energy supply. STEP (Spherical Tokamak for Energy Production) is a UK Atomic Energy Authority (UKAEA) programme to design and build a prototype fusion energy plant capable of supplying net electricity to the grid by 2040. There is limited space for the central column in the compact spherical Tokamak. This study concerns the electromagnetic analysis of a central column where 16 Toroidal Field (TF) limbs are placed inside the Central Solenoid (CS) using an efficient 2D Finite Element Method (FEM) simulation. Despite the strong electromagnetic interaction between the CS and TF limbs during the current change in the coil sets, the total AC loss in the central column is within the cryogenic design margin. Meanwhile, the hysteresis loss from the High-Temperature Superconducting (HTS) cable conductors is much higher than the eddy current loss and coupling loss from stainless-steel materials, which means adjusting the charging or discharging speed is unlikely to reduce the total heat dissipation. Also, the benefits and drawbacks between an HTS CS and a resistive CS are compared based on the difference in space optimisation and cryogenic design. In addition, insulated or partially insulated HTS cable conductors and the series connection between the conductors inside TF limbs can further reduce AC losses. This study presents the feasibility of a compact central column design and proposes 2D simulation methods useful for the design of other compact devices.

Keywords

Compact fusion, STEP, Electromagnetic analysis, AC loss, FEM simulation.

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PS1-125 ABSTRACT-2584
H. Models and Experiments for FNT

Assessment of the relevancy of ENEA Water Loop facility with respect to ITER WCLL TBS Water Cooling System by considering their thermal-hydraulic performances during selected transient conditions

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The Water-Cooled Lead-Lithium (WCLL) is one of the two candidate concepts for the Breeding Blanket (BB) of DEMO. As part of the process to support its development, a Test Blanket Module (TBM) together with its Water Cooling System (WCS) are going to be installed and tested in the ITER reactor. The WCS acts as primary cooling circuit of the TBM module, and it is designed to reproduce the water thermodynamic conditions expected at the DEMO TBM inlet.

During last years, ENEA and its related partners, including the DIAEE of Sapienza University of Rome, have carried out the conceptualization of the Water Loop (WL) facility, belonging to the W-HYDRA experimental platform planned at C.R. Brasimone. The W-HYDRA platform is composed by three individual facilities called: Water Loop, Steam, and LIFUS5/Mod4. In particular, Water Loop is a 1:1 reproduction, in terms of thermo-hydraulics conditions and components, of the WCLL WCS. The facility is equipped with a test section placed inside a Vacuum Vessel (VV) capable to investigate mock-ups of the whole TBM or its individual parts.

The aim of this paper is assessing the relevancy of the WL facility with respect to ITER WCLL TBM System. For this purpose, their thermal-hydraulic performances were evaluated and compared during selected operational and accidental conditions. To perform the transient calculations, two RELAP5/Mod3.3 models were developed. The simulation outcomes showed a good agreement between the behavior of the two systems, demonstrating the effectiveness of WL facility to support the ITER TBM program.

Keywords

Water Loop, RELAP5, Accidental analysis, design, DEMO Breeding Blanket.

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PS1-126 ABSTRACT-088c
I. Repair and Maintenance

Breeding Blanket Segment Handling: Kinematics and Dynamics

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A critical remote maintenance task for EU-DEMO and CFETR is the removal of breeding blankets from the vacuum vessel, as well as their replacement. This is carried out using a specialized tool, the breeding blanket transporter, which is automatically operated within a sealed cask having high temperature and active radiation. The transporter is lowered through the upper port of a vacuum vessel sector and grips one of the 5 breeding blanket segments within using an interlock integrated in the segment. Using its 7 articulated joints, the transporter disengages the blanket segment from its supports in the vacuum vessel and lifts it into the cask while avoiding collisions. To achieve automation and control, a prerequisite is to establish the kinematic analysis. Therefore, forward kinematic equations of the transporter are derived by analyzing it as a 7-degree of freedom robotic manipulator, including the Jacobian matrix relating joint velocities and end-effector velocities. The forward kinematic equations are employed to characterize the reachable workspace, identify critical target points and joint limits, and calculate static loads at the joints. Inverse kinematics solutions are found for the case when the trolley tilting joint is set to a known position, using both graphical and algebraic methods. These are applied to generate valid waypoints for the blanket segment removal trajectories, including the engagement of the gripper and application of pre-load about a constant axis. The removal of the blanket segments from the vacuum vessel using the transporter is verified and animated using a minimal number of waypoints which avoid collisions when assuming joint-level linear motion. The waypoints are optimized with respect to the static loads at the joints. With a solid kinematic foundation, the dynamic equations of motion are derived and used to estimate the dynamic loads, achievable joint velocities, and overall durations of the removal trajectories.

Keywords

Breeding Blanket, Remote Maintenance, Kinematics, Dynamics, Robotic Handling.

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PS1-127 ABSTRACT-ebeb
I. Repair and Maintenance

Conceptual design of DEMO Breeding Blanket In-Vessel Toroidal Transporter

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The development of a viable maintenance strategy for the replacement of the Breeding Blanket (BB) segments of DEMO is a key aspect in the way for fusion energy. Previous work concerned the development of a BB vertical transporter to lift, tilt and handle the BB segment inside the remote handling (RH) ports. A potential solution for the replacement of the BB segments could consist in lifting all BB segments from four RH ports, by mean of BB vertical transporter coupled with a BB toroidal in-vessel transporter avoiding the opening of the remaining ports during maintenance. In this way, the activities concerning the removal and re-installation of the auxiliary components (BB feeding pipes, permanent structure for pipes handling, closure plate, port plug, etc) inside the non-remote handling ports will be avoided, having a strong impact on the reduction of the remote maintenance times and hence on the overall availability of the machine. To assure the feasibility of the proposed strategy a dedicated tool, capable to move toroidally the huge BB segments, shall be developed. The main function of the tool will consist in lifting and toroidal translation of the BB segments that have to be aligned to the RH port, where they will be grabbed and lifted by the BB vertical transporter currently under development [1]. The work here presented focuses on the conceptual design of a toroidal transporter to lift and toroidally translate the BB segments inside the Vacuum Vessel. The conceptual design of the BB Toroidal Transporter (BBTT) was developed to handle both the inboard and the outboard; a foldable lower foot, equipped with toroidal trucks and suspension system, has been designed to withstand the assumed loads and seismic loads. Preliminary FEM analysis has been carried out to check the structural integrity of the proposed design.

Keywords

DEMO, Remote Maintenance, Breeding Blanket, CAD, FEM.

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PS1-128 ABSTRACT.-068f
I. Repair and Maintenance

Induction brazing of DEMO's large bore cooling pipes

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DEMO's first wall blanket components will be exposed to high neutron fluence and high levels of thermal energy, the latter being removed by a coolant fluid transmitted out of the reactor via service piping. Due to the thermal and nuclear loading on the first wall, the components will need to be removed and replaced. Installation of new blankets requires the joining of their service pipes. Due to the nature of fusion reactors, almost all maintenance will be done by remote handling.

Brazing is one of the alternatives that are considered for joining the service pipes of the DEMO breeding blankets. This technique has great potential and has a number of major advantages compared to welding: small heat affected zone (HAZ); parent metal does not need to be melted, which reduces the chance of a change in the material properties; lower waste production, no sputtering.

The Centre for Energy Research has been working on the development of a viable brazing concept of thick, large bore pipings inside the ports of DEMO. Induction has been chosen as the heat source for the brazing operation, which is a well-studied technology and has many uses in the industry already. However, brazing in this size is a completely new horizon for induction heating so CER's work includes a significant amount of development. First, the coil is naturally a bespoke component, which has been optimized by electromagnetic analysis. Secondly, a test rig has been designed and manufactured in-house by CER, to test the performance of the induction heater in a real scenario.

This paper outlines the development strategy of the induction brazing of DEMO's large bore pipes.

Keywords

Remote maintenance, induction heating, brazing, electromagnetic analysis.

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PS1-129 ABSTRACT-0ba9
I. Repair and Maintenance

Applicability Study of Mechanical Multi-Pipe Connections for DEMO Breeding Blanket Maintenance concept

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Maintenance of DEMO breeding blanket includes the removal and replacement of plasma facing components, for example the breeding blankets (BB). For access, multiple coolant pipes need to be removed. As an option to reduce downtime and increase maintenance speed, a so-called mechanical multi-pipe connection (MPC) concept is developed to allow removal of multiple pipes at the same time using remotely operated mechanical connections. This BB-maintenance concept aims for further reduction of downtime by reducing the number of ports to be opened. That means, that all pipe connections have to be disconnected remotely directly at the top of the BBs - without removing the pipes and the shielding blocks above the BBs. The BBs then will be moved toroidal to an open port for replacement. MPC is one promising candidate to allow fast remotely controlled opening and closing of multiple pipe connections by using flanges and bolts.
Major challenges for the application of MPC are highly constrained space on top of the BBs, accessibility for appropriate tools and homogenous distribution of sealing forces on the gaskets. Numerous studies on arrangement and force application methods were performed to identify suitable concepts – of which a compact flange design using direct bolted connection is favourable. Finite Element Analyses on the shape and geometry optimisation of the MPC led to analytically verified solutions. Furthermore, material selection and behaviour under radiation, vacuum, and thermal cycling at high temperature conditions requires advancement of available material treatment solutions in terms of e.g. coated surfaces of threads and contact areas on flanges to avoid diffusion bonding. For integration of these aspects into proof-of-principle prototypes, test rig trials for validation are currently ongoing and first test results are presented.

Keywords

DEMO, Remote Maintenance, Mechanical Pipe Connection (MPC).

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PS1-130 ABSTRACT-0c65
I. Repair and Maintenance

Fabrication and Load Test of ITER Assembly Tools for Lifting Heavy Components

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Maintenance of DEMO breeding blanket includes the removal and replacement of plasma facing components, for example the breeding blankets (BB). For access, multiple coolant pipes need to be removed. As an option to reduce downtime and increase maintenance speed, a so-called mechanical multi-pipe connection (MPC) concept is developed to allow removal of multiple pipes at the same time using remotely operated mechanical connections. This BB-maintenance concept aims for further reduction of downtime by reducing the number of ports to be opened. That means, that all pipe connections have to be disconnected remotely directly at the top of the BBs - without removing the pipes and the shielding blocks above the BBs. The BBs then will be moved toroidal to an open port for replacement. MPC is one promising candidate to allow fast remotely controlled opening and closing of multiple pipe connections by using flanges and bolts.
Major challenges for the application of MPC are highly constrained space on top of the BBs, accessibility for appropriate tools and homogenous distribution of sealing forces on the gaskets. Numerous studies on arrangement and force application methods were performed to identify suitable concepts – of which a compact flange design using direct bolted connection is favourable. Finite Element Analyses on the shape and geometry optimisation of the MPC led to analytically verified solutions. Furthermore, material selection and behaviour under radiation, vacuum, and thermal cycling at high temperature conditions requires advancement of available material treatment solutions in terms of e.g. coated surfaces of threads and contact areas on flanges to avoid diffusion bonding. For integration of these aspects into proof-of-principle prototypes, test rig trials for validation are currently ongoing and first test results are presented.

Keywords

ITER, Assembly Tools, CS Lifting Frame, Sector Lifting Frame, Load Test.

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PS1-131 ABSTRACT-1191
I. Repair and Maintenance

Architecture for human-in-the-loop control of robotic equipment for remote maintenance: case study on the NEFERTARI Project

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In this work, we report an innovative architecture for human-in-the-loop control of robotic systems used for the remote maintenance of fusion power plants. The architecture comprises three layers: a low-level layer, an intermediate layer, user high-level layer. In the low-level layer, the robot's low-level control and safety are computed as well as the low-level trajectory planning. The intermediate layer is composed of multiple modules: a high-level trajectory planning module, collision detection module, Application Programming Interface (API) for vision and force sensing module, and robot dynamic modelling module for computing the dynamic model using rigid and/or flexible robot behaviours. The user high-level layer involves a virtual reality (VR) module for the visualization of the robot's movements, enabled by a digital twin of the real robot. The modules and layers communicate via ROS2 framework. A typical usage of such a framework involves a human operator who is teleoperating the real robot, the encoder data are read from the motors and given as input for the dynamic model module which is used to compute the forward dynamics in real-time and provide the visualization of the robot in the virtual environment including its elastic behaviour. This paper deals with the detailed description of each architecture module and its usage on a robotic manipulator that is representative of performing remote maintenance operations in fusion power plants. A case study on the NEFERTARI project dealing with the RFX-Mod2 machine is also presented.

Keywords

Remote maintenance, virtual reality, teleoperation, robotics.

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PS1-132 ABSTRACT-1309
I. Repair and Maintenance

Overview of in-bore pipe cutting and welding tools for the maintenance of CFETR and EU-DEMO

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In the remote maintenance of fusion nuclear devices, one of the most challenging tasks is the cutting and the re-welding of large pipes that feed the breeding blanket. The narrow space around some of the pipes hinders the suitability of conventional orbital processes. For such pipes, both the cutting system and the welding one have to be compact enough to move along their interior, lock against the pipe inner wall and keep the position during the operation. The inner diameter and thickness of the pipes together with the other severe requirements related to the remote maintenance drive the choice of the processing technologies and the design of the tooling configuration. In this work, activities carried out by a joint team from EU-DEMO and CFETR on the in-bore cutting and welding operations are presented and discussed.

For the in-bore cutting system a solution for the simultaneous parting and bevelling has been developed opting for a mechanical cutting with a pair of symmetrically distributed knives. A preliminary dimensioning of the motors has been made assuming the radial feed and the cutting speed of the knives.

For the in-bore welding system, the TIG welding with filler wire has been considered as the most reliable and suitable technology within prescribed requirements and constraints. A multi-pass welding operation has been assumed.

A conceptual design of both tools has been developed. The space constraint remains the most critical issue. The motion system of the welding torch is also another issue to be still addressed. Further verification and validation work is planned in close collaboration between European laboratories and the Comprehensive Research fAcility for Fusion Technology (CRAFT) at ASIPP in Hefei.

Keywords

DEMO, CFETR, Remote Maintenance, Breeding Blanket, Feeding pipes.

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PS1-133 ABSTRACT-1b98
I. Repair and Maintenance

Design and development of a Digital Maintenance Manual for RH operations in fusion reactors

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Remote Handling (RH) is a key aspect in fusion reactors since all the in-vessel components are subject to very harsh operative conditions and may need to be replaced more times during the lifetime of the machine, using teleoperated robots to guarantee operators radioprotection. The development of a well-structured Maintenance Manual, where to explain all the RH procedures, is necessary to provide guidance to Human Operators (HOs) controlling robots in the Control Room. This allows an efficient, safety and time-saving execution of the maintenance procedures. A Digital Maintenance Manual has been designed basing on the principles of usability, manageability and accessibility. Usability is necessary to make the HO properly understand and conduct the assigned tasks. Manageability allows to promptly update the manual to several possible scenarios (procedures improvements, procedures/tasks addition, machine design modifications). An easy accessibility guarantees the widest possible spread of the document in the project team, making it a reference where to keep track of the impact of the RH systems on the in-vessel design. In light of this, the manual consists in a web-based multimedia platform available and editable to whoever has the required permissions. The HO shall easily navigate it and reach the on-going task, while also having the awareness of the whole procedure in which the task is placed. In addition to written instructions, he/she will be guided by video animations and explanatory images (obtained through 3D virtual models and simulations), organized to describe the actions as clearly as possible. The Digital Maintenance Manual presented in this paper has been developed for the upcoming Divertor Tokamak Test (DTT) facility, although it aims at representing a framework replicable by each kind of fusion reactor.

Keywords

Remote Handling, Maintenance Manual, DTT, Robots teleoperation, Control System.

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PS1-134b ABSTRACT.-b9f3
I. Repair and Maintenance

Replacement strategy of the EU-DEMO and CFETR breeding blanket pipes

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Both European DEMO and CFETR need a workable replacement scheme for their respective tritium breeding blanket (BB). The radioactive environment dictates that all associated operations must be carried out through remotely controlled tools. Accessing and extracting the large BB segments requires their feeding pipes to be first removed from the upper vessel port and later re-installed. In the current design, each of the 80 BB segments is connected with four feeding pipes, two for cooling and two for tritium extraction. Thus, in the process of BB maintenance operation, 320 pipes must be cut, removed and rewelded. A strategy for such a task has been developed aiming at reducing the associated plant downtime, thereby increasing the overall plant availability. Features underpinning this integrated BB pipe service concept are: (i) parallel pipe service operations, (ii) pipes in each upper port are grouped in a pipe forest and handled as a single component, (iii) the configuration of the individual pipes is standardized such that cutting and joining locations are aligned and with good accessibility from the top, (iv) the number of pipe sizes is limited to two, reducing the number of required tool sets, and (v) the same pipe configuration is adopted in each of the 16 upper ports. The paper will present design solutions and the progress on the manufacturing of prototypes developed for the challenging cutting and welding tasks from both within and outside, as well as the leak detection methodology and pipe stub handling. The prototypes will be used to perform design validation and verification on dedicated test benches currently being implemented in close collaboration between European laboratories and the Comprehensive Research fAcility for Fusion Technology (CRAFT) at ASIPP in China. Finally, the paper will show that the proposed scheme is consistent with the replacement approach of the BB segments.

Keywords

DEMO, CFETR, Remote Maintenance, Breeding Blanket, Feeding pipes.

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PS1-135 ABSTRACT-3fa0
J. Burning Plasma Control and Operation

On the accuracy of the fast time resolution inversion method for the detection of different radiation patterns in fusion reactors

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Accurate measurements of the emitted radiation are crucial for controlling a fusion reactor. In addition to affecting the global power balances, high levels of radiation emission can indicate that the plasma is losing stability and can disrupt. Several radiation events can occur in a tokamak differing in localization, shape, and dimension. Each of these can be originated by various causes and degrade the plasma confinement in different ways. On current tokamaks, radiation is measured by bolometers, but these provide only line integrated values. For this reason, quite sophisticated tomography inversion algorithms are required to obtain local information, but this approach is slow and cannot be used in real time. A fast inversion method, which provides local information with low spatial resolution, has been developed. This allows to reconstruct the emissivity in relevant regions of the plasma and so to recognise different anomalous radiation events. In this work, a validation for the fast inversion method and the analysis of its performances are presented using synthetic data for a ASDEX-like shape. The reliability of the method is tested by simulating different patterns of radiation. Then the accuracy is evaluated by analysing the impact of the different features, like shape and position, on the reconstruction. Finally, the methodology is also compared with the most commonly used techniques.

Keywords

Disruption avoidance, Plasma emissivity, Impurity tracking.

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PS1-136 ABSTRACT-d419
J. Burning Plasma Control and Operation

Study on burning start-up scenario in DEMO with consideration of fuel isotope effect and helium ash transport

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This study has investigated the effect of particle transport of fuel ions (deuterium (D) and tritium (T)) and helium ash (He) as well as their heat transport on the nuclear fusion output in a DEMO reactor. The result was assessed and possible start up scenarios for DEMO was discussed.

While development of a reliable startup scenario for DEMO that aligns with physical design is important, it still remains in research with many different assumptions over the plasma physics. Particle control is an open issue because of immature physics understanding and insufficient technological readiness. Establishment of a fueling and exhaust scheme is inevitable for both stable burning control and efficient T fuel circulation in a fusion plant.

In this study, the hydrogen isotope effect in D/H/He mixture plasmas on transport and resultant fusion power output has been discussed with the integrated tokamak modeling code TASK/TR which incorporates the CDBM model . The consequence of He buildup has been also assessed by presuming the enhancement of He diffusive transport.

It has been shown that fueling and heating scheme significantly affects stability and fusion power output. Achievement of the Q value of 10 is possible when the heating and particle fueling scheme is controlled properly, despite deviation of DT ratio from 1:1 due to the isotope effect. The result has also suggested that efficient fueling of D and T to the core region is difficult in particular in the start-up phase without anomalous inward pinch. Together with the buildup of He, fuel ions show hollow density profiles, which hinders higher Q value. While the enhancement of He diffusion prevents DT fuel dilution, simple increase of DT fueling decreases the plasma temperature and consequently degrades fusion power output. Further optimization of start-up scenarios is addressed.

Keywords

DEMO start up, fuel transport, isotope effect, helium ash, TASK.

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PS1-137 ABSTRACT-0aa7
J. Burning Plasma Control and Operation

Proposal of a control scheme for testing a centrifuge-based pellet injection system in DIPAK-PET

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The envisaged Pellet Engineering Testbed (PET) in the framework of Direct Internal Recycling Development Platform Karlsruhe (DIPAK) will provide unique opportunities to advance pellet injection technology and its implementation in the fuel cycle as well as into plasma control of EU-DEMO.

This work analyses the parameters for Pellet Launching System (PLS) control according to requirements from plasma control. Main parameter is the particle flux and the related accuracy with respect to amount of flux and arrival time of pellets. Others are addressed as well like dynamic response to setpoint changes and adaptivity of isotope composition of the hydrogen ice. The performance of the PLS is limited by the mass loss in the guiding tubes, both by erosion (applies to any pellet) and loss of an entire pellet (only a very few). The total amount of mass loss is depending on the guiding tube geometry. DIPAK-PET has to provide data points in order to enable proper guiding tube design activities based on modelling. This is essential for the design of these components for EU-DEMO.

A short description of the proposed technical solution for the Pellet Launching System is provided, as some of the parameters are connected to a dedicated technology. A setup for the Programmable Logic Control (PLC) is suggested comprising a MasterPLC which is controlling the system via the PLC of subsystems and a dedicated Human Machine Interface (operators' desk). A preliminary list of IO- parameters of the control system is compiled.

A non-exhaustive proposal of experiments to be performed in DIPAK-PET is listed in order to verify the estimated performance values for the relevant PLS parameters.

Keywords

Pellet, DEMO, DIPAK-PET.

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PS1-138 ABSTRACT-0ecf
J. Burning Plasma Control and Operation

Development of a one-step ion optics for beam emission spectroscopy and atomic beam probe measurements

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At Alkali Beam Emission Spectroscopy (BES) [1] and at Atomic Beam Probe (ABP) [2] (or also called imaging heavy ion beam probe [3]) systems the accelerated beam current is critical to obtain reasonable information about fusion plasmas. A two-step ion optics was developed and used mostly in the past ~40 years [4], where the extraction voltage is about tenth of the beam energy (~6kV at 60kV beam energy).

The extracted beam current for heavier alkali species is limited by the extraction voltage (space charge) of the ion optics (). As currently higher emission capacity ion sources are available, the beam current could be at least doubled (limited only by the ion source) if higher extraction voltages were used. To overcome this problem, a one-step ion optic is developed for BES and ABP diagnostics considering the recent accelerator structure, modifying only the electrodes of the ion optic.

In this paper the computational and experimental result of a BES/ABP diagnostics, operating with Li, Na, K, Rb and Cs ion sources, are 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

[4] Review of Scientific Instruments 56, 1063 (1985)

Keywords

Keywords: thermionic alkali ion sources, ion optics, plasma diagnostics, accelerator design.

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PS1-139 ABSTRACT-1031
J. Burning Plasma Control and Operation

Assembly Procedure and Tooling Development for In-Situ Winding of ITER In-Vessel Vertical Stability Coils

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The ITER In-Vessel Vertical Stabilization (VS) Coils, composed by the Upper and Lower VS Coils, are designed to provide fast vertical stabilization of the plasma through a fast current noise profile to compensate small plasma oscillations and to recover vertical displacement events.

The VS coils are made by 4 continuous turns of Mineral Insulated Conductor (MIC) joined by Inconel ® 625 welded Brackets, installed on rails on the inner wall of ITER Vacuum Vessel (VV). Thus, the coil turns winding/forming and the entire coils assembly have to be performed inside ITER VV, forcing the assembly procedure and tooling design to cope with the very limited space and strict constraints of ITER VV environment. The proposed procedure foresees that MIC conductor is un-spooled in the Neutral Beam Port Area and inserted into the ITER VV through a NB Equatorial Port, then a set of tooling is installed on a reinforced staging platform to wind, form and seal the conductor turns and finally to weld Brackets under pre-compression. For this purpose, the tooling design including the reinforcement staging has been developed to meet the required tooling function, configuration optimization, assembly procedure and environmental constraints. The dedicated tools to be used for the assembly procedure of the VS coils are categorized into ex-vessel tools for unspooling and cleaning, in-vessel tools for winding and bump forming, bracket assembly tools for pre-compression and welding, termination and sealing tools, ancillary lifting tools and reinforcement staging platforms.

This paper provides the overall assembly procedure and tooling development for the in-situ winding of VS coils.

Keywords

ITER, Vertical Stabilization Coils, Assembly Procedure, Tooling Design.

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PS1-139b ABSTRACT-ecdb
J. Burning Plasma Control and Operation

Design and Performance analysis of a High Field Side antenna for Plasma Position Reflectometry control on DTT

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The Divertor Test Tokamak (DTT) offers the possibility of testing and validating, in reactor-like regimes, relevant non-magnetic control diagnostics in support of the DEMO design. One such diagnostic, O-mode reflectometry, was proposed as an alternative source of real-time (RT) plasma position and shape measurements for control purposes. At DEMO, Plasma Position Reflectometry (PPR) involves the use of several poloidally distributed lines-of-sight (LOS). However, since on present devices profile reflectometers are mainly built at the equatorial plane, experience in PPR control has only been gained by probing the plasma at this plane, simultaneously both from the Low and High Field Sides (HFS). To gather more comprehensible experimental knowledge on the operation of such systems, before a full deployment on DEMO, a four LOS PPR in DTT is presently under consideration. Priority has been given to planning the HFS reflectometer due to its impact on the design of the first wall and vessel. Two optimized, small-footprint, bistatic and monostatic hog-horn antenna designs are proposed to cope with the severe constraints imposed on the HFS antennas' size and access to the chamber and in-vessel waveguide routing. Herein we assess the proposed antennas' measurement performance in the DTT single null scenario using 2D and 3D finite-difference time-domain codes of the REFMUL family. The simulation results are further complemented by laboratory tests of the proposed antennas' 3D-printed mockups. The CAD models of the antennas embedded in the plasma-facing wall structures, used in the simulations, were also used to produce a preliminary neutronics and thermal analyses to be discussed in the frame of the proposed HFS antennas' viability assessment. This exercise is the first application of an integrated design workflow that incorporates reflectometry 2D and 3D advanced full-wave simulations and thermal and neutronic analysis in the iterative development cycle of a PPR system.

Keywords

Plasma position reflectometry, DTT, antenna design, FDTD full wave simulation, neutronics and thermal analyses.

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PS1-140 ABSTRACT.-1eff
K. Inertial Confinement Fusion Studies and Technologies

Theoretical methods for stopping power of plasmas in magnetic fields

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Interaction between hot partially or fully ionized plasmas and ion beams is currently a popular subject of study in the present times in many fields of physics. These interactions are especially useful to understand the Inertial Confinement Fusion (ICF) problem.

When interacting with a plasma, particles interchange energy with the target depending on its stopping power (SP) [1-4]. This is why many theoretical models are under development to estimate plasma SP at variable conditions. Within the dielectric formalism, the polarized SP can be calculated using the system dielectric function. However, there exist also other theoretical methods.

It is crucial to consider the influence of magnetic fields, as they can modify the direction or the energy of the projectiles [5-7]. Although there are not many, different models focusing on SP in a magnetic field have been tested and compared to the experimental results, the aim is to employ them together with our own theoretical model which is to be improved to obtain a reliable method capable of calculating SP of both free and bounded electrons in plasma under the influence of a magnetic field.

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Keywords

Energy deposition, Stopping power, Magnetic plasmas.

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PS1-141 ABSTRACT-421c
K. Inertial Confinement Fusion Studies and Technologies

Development of a Novel IEC Neutron Research Facility

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Inertial Electrostatic Confinement (IEC) fusion-based particle generators utilise high voltages in the 10’s of kV and a current in the 10’s of mA applied to a mBar pressure vacuum vessel through a centralised grid-like cathode. Ionising gas inside the chamber and providing kinetic energy through an electric field between electrodes, prompting a fusion reaction. The following light gas mixtures can be introduced to the system to spark plasma and enable the production of a variety of high-energy particles; 2.45 MeV neutrons and 3.03 MeV protons from Deuterium-Deuterium (DD), and 14.1 MeV neutrons from Deuterium-Tritium (DT), and 14.7 MeV protons from Deuterium-Helium-3 (DHe).

Access to neutrons for research and commercial purposes is currently limited in the UK, where much of the global supply made up of ageing fission reactors will also go offline in the coming decade. In light of this, fusion-based neutron sources such as IEC particle generators offer a small footprint and inherently safe alternative. Recent advances in fusion materials research has enabled such IEC particle generators to have fusion rates in excess of 1×10^{11} n/s DT. This allows for such systems to be useful in many applications such as fusion breeder blanket testing, medical radioisotope production, as well as neutron interrogation and activation analysis. In 2023, a new research facility has been opened at the Dorset Innovation Park at the decommissioned Magnox nuclear site at Winfrith consisting of several IEC devices and a low-activation water-based shielded bunker capable of withstanding 1×10^{9} n/s DD.

The capability of such a facility allows for research into maximising IEC performance, spearheaded by Astral Systems’ commercially proven core architecture, through cutting edge fusion materials research. Deploying fusion technologies in the near-term helps bring the associated societal benefits of the future fusion era, such as the harnessing of high energy particles, into the present.

Keywords

Fusion, Inertial Electrostatic Confinement, facility, research, materials, plasma, deuterium, nuclear medicine.

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PS1-142 ABSTRACT.-489c

K. Inertial Confinement Fusion Studies and Technologies

Spatially resolved hot implosion core conditions via Kr K-shell X-ray spectroscopy

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X-ray spectroscopy is a powerful diagnostic of plasma electron temperature (T_e) and density (n_e). In the past, argon K-shell has been extensively used in Inertial Confinement Fusion experiments as spectroscopic tracer. However, argon becomes too ionized at high electron temperatures ($T_e >$ 2keV). To overcome this limitation, krypton K-shell is an attractive alternative to diagnose hot implosion cores. To this end, we have fielded two crystal spectrometers in Kr-doped implosion experiments at OMEGA: a slit imaging spectrometer (XRS) providing 1D spatial resolution and a pinhole imager (Kr MMI) providing 2D spatial resolution. The new Kr MMI is a modification of the previous argon version which has been designed for this research[1]. The Kr MMI has been successfully fielded in two OMEGA campaigns. Results show both krypton He-like alpha and beta line spectrum, and associated Li-like satellite transitions. We have used the atomic model and code PrismSpect[2] to solve the atomic rate equations self consistently with the radiation transport equation to obtain the level population distribution. Detailed Stark broadened spectral line shapes were computed with MERL[3] including the microfield effects of plasma electron and ions. Combining both results and transporting the radiation through the plasma to obtain the emerging intensity distribution we obtained a synthetic spectral database that accounts for the T_e and n_e sensitivity of level populations as well as the n_e dependence of Stark broadening. This database is compared to the experimental spectra to infer T_e and n_e plasma conditions in the implosion core. The comparison of our database with spatially resolved spectra provides T_e and n_e spatial distributions in the plasma.

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Keywords

ICF, spectroscopy, krypton, X-ray, diagnostic, imager, plasma conditions, design.

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PS1-143 ABSTRACT-5543
K. Inertial Confinement Fusion Studies and Technologies

Relativistic effects on the stopping power of high atomic number plasmas

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This work presents the stopping power of plasmas composed of heavy ions (Z>54). Using the dielectric formalism, we compute the energy loss contribution of free and bound electrons separately. We employ the Arista dielectric function for quantum plasmas [1], which includes a temperature dependence, to describe the free plasma electrons. Bound electrons are considered within the shellwise local plasma approximation (SLPA) with the Levine-Louie dielectric function [2,3]. This method allows calculating sub-shell energy loss contributions independently by introducing a dependence on the target's orbital binding energies and electronic densities. These approaches have been successfully employed for H on various targets, such as C, Si, and Fe [4]. In this work, we deal with heavier targets. Therefore, we include relativistic effects in our calculations, particularly in the bound electron contribution. We solve the corresponding Dirac Hamiltonian to describe the electronic structure of the target. These corrections to the bound electron stopping power contribution have shown to be necessary, especially for ions with low charge states. We compare our values with non-relativistic models and experiments when available. Our results show these corrections are essential to describe the stopping correctly.

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Keywords

Stopping power; Energy loss; Plasmas; Relativistic electrons; Inertial Confinement Fusion.

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PS1-144 ABSTRACT.-00a7
K. Inertial Confinement Fusion Studies and Technologies

Effects of impurities in DT plasmas on burning gains and self-heating and ignition curves in ion fast ignition scheme.

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Fundamental research in ion beam-target plasma interaction is essential to advance in the understanding of high energy density plasmas. This interaction is expected to occur at a future confinement fusion power plant in the ion fast ignition (IFI) scheme. The target contains the fuel which is typically made up of deuterium-tritium (DT), although it is possible to find traces of impurities due to the detached components from the ablator material. Therefore, it is a priority to analyse the influence of these impurities in the system. The aim of this work is to study the effect of impurities in DT plasmas for the burning gain, the self-heating and the ignition curves. The analysis of these curves will be done in terms of the beam parameters (atomic number, particle density and energy). For this purpose, we use a 1D spatial-temporal numerical model to simulate the slowing down of an ion beam in a plasma target, as well as the plasma heating process. Three kinds of beams have been considered (proton , carbon and vanadium) in a wide range of kinetic energies and number of particles. With respect to the plasma, we study a compressed sphere of DT fuel, with homogeneous temperature and density distributions. We analyse different conditions of interest in the IFI scheme, such as the plasma density, its initial temperature and a variety of impurities at different concentrations.

Keywords

Fast Ignition, Simulation, Thermonuclear Gain.

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PS2-1 ABSTRACT-1204
A. Plasma-Facing High Heat Flux Components

Thermal-hydraulic study of the Primary Heat Transport System of the EU-DEMO Divertor Plasma Facing Components

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In accordance with the Horizon Europe research framework programme, the EUROfusion consortium is devoting itself to the development of the DEMOnstration Fusion Reactor (EU-DEMO), which will be the first fusion device on a large scale to produce net electricity. Although many of the design criteria and safety requirements of the EU-DEMO reactor are similar to common nuclear power stations, the tokamak reactor is supposed to undergo a pulsed duty cycle under normal conditions and for this reason, due to the thermal and mechanical cycling, the qualified lifetime of the main equipment could be challenged. Furthermore, the plasma control strategy in EU-DEMO considers the potential occurrence of planned and off-normal plasma overpower transients that might harm the structural integrity of the plasma facing components. Therefore, it is essential to predict the thermal-hydraulic behaviour of the EU-DEMO Primary Heat Transport System (PHTS) through purposely conceived tools. In particular, in this work the attention has been focussed on the PHTS of the Divertor Plasma Facing Components (DIV PFCs). In this regard, the University of Palermo in collaboration with the DEMO Central Team has developed a thermal-hydraulic model to study the DIV PFCs PHTS under steady-state and transient conditions. The study has been carried out following a computational-theoretical approach based on the use of the TRACE thermal-hydraulic system code. In order to catch all main geometrical, hydraulic and heat transfer features characterizing both in-vessel and ex-vessel components, a detailed finite volume model has been developed. The analysis of the thermal-hydraulic behaviour of the DIV PFCs PHTS has been conducted both under hypothetical steady-state conditions and during the typical DEMO duty cycle so as to verify whether the current PHTS design is able to sustain the pulsed loads it undergoes under nominal conditions. Models, hypotheses and results of the study are reported and critically discussed.

Keywords

DEMO, Divertor, Plasma Facing Components, PHTS, Thermal hydraulics.

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PS2-2 ABSTRACT-c83e

A. Plasma-Facing High Heat Flux Components

Beryllium limiters from JET tokamak with the ITER-like wall: Surface modification and fuel retention

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The JET tokamak with the ITER-like wall (JET-ILW) [1] is operated with beryllium (Be) limiters and tungsten (W) in the divertor. The determination of erosion-deposition effects and fuel retention is part of the JET-ILW top missions. Gas balance measurements and a range of ex-situ analysis methods are applied to quantify deuterium (D) content in plasma-facing components (PFC).

This work is focused on the morphology of Be limiters after the three JET-ILW campaigns (ILW-1, ILW-2, ILW-3). In particular, fuel retention, its lateral distribution and total contents are of interest in view of the JET D-T operation. Samples from dump plates (DP), inner wall guard limiters (IWGL) and outer poloidal limiters (OPL) were examined. In total 40 samples from those tiles and one unexposed reference sample were studied. They were analyzed with heavy ion elastic recoil detection analysis (HI-ERDA) and scanning electron microscopy (SEM). The main results are:

(a) Comparison of fuel content between ILW-1 and ILW-2 on the corresponding position of the IWGL shows the reduction of D content below the HI-ERDA detection limit for ILW-2, which is related to the H_2 fueling of the last 300 shots of ILW-2.

(b) The lateral distribution of co-deposits on the OPL is examined. The wing areas of the samples show a significantly higher amount of retained species than the plasma-wetted areas.

(c) The retrieved dump plates have molten parts, which consist of 99 atomic % clean Be with very small amounts of retained D.

The results are compared with data from [2] and [3] and placed in context with plasma times and fueling of the campaigns.

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Keywords

Erosion-deposition, beryllium, fuel retention, JET-ILW.

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PS2-3 ABSTRACT-607a
A. Plasma-Facing High Heat Flux Components

Experimental Evaluation of Thermal-Fluids Performance of Helium-Cooled Flat Plate Divertor

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Various helium (He)-cooled solid-tungsten (W) divertor concepts have been proposed for long-pulse magnetic fusion energy reactors. Among these concepts, the He-cooled flat plate divertor (HCFP), originally proposed more than 20 years ago, has the largest plasma-facing surface area, with nine 2 cm wide cooling units arranged under a rectangular 100 cm x ~19 cm castellated W armor plate. In each cooling unit, He issues from a 0.5 mm wide slot to form a planar jet that impinges on, and cools, the tungsten-alloy pressure boundary. This design was optimized by ARIES, and simulations have shown that the most recent version can withstand heat fluxes as great as 8 MW/m^2.

Earlier experimental studies of a single shortened HCFP cooling unit with a slot length of 7.6 cm, used air at ambient temperatures and pressures less than 0.6 MPa. Here, we present initial experimental studies of a high-strength copper alloy and steel test section modeling a shortened HCFP cooling unit at prototypical helium pressure of 10 MPa. Since the He mass flow rate was limited to 10 g/s, the slot length was reduced to 3 cm to match the dimensionless prototypical mass flow rate, or Reynolds number *Re*, of 3.3x10^4. Experiments were performed at He inlet temperatures T_i <= 200 °C and steady-state incident heat fluxes $q^²$ <= 1.2 MW/m^2. We will present experimental data for average dimensionless heat transfer coefficients, or Nusselt number *Nu*, and pressure loss coefficients, both as a function of *Re*, for *Re* = 1.2x10^4-3.3x10^4. These data are used to validate a numerical model of the test section using commercial computational fluid dynamics (CFD) software, which is used in turn to evaluate the impact of the shortened slot. Based on these analyses, numerical predictions of the thermal-fluids performance of the HCFP at prototypical conditions will be presented.

Keywords

Divertor, Helium cooled divertor, Thermal-fluids.

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PS2-4 ABSTRACT-d517
A. Plasma-Facing High Heat Flux Components

Design Progress of DTT Divertor Fixation System

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One of the main challenges for the construction of DEMO, the first fusion demonstration reactor, is the power exhaust issue. To deal with it, EUROfusion has foreseen the buildup of DTT (Divertor Tokamak Test) facility which will be held in Frascati in Italy. It aims to test various divertor models which can be integrated and used under different plasma confinement configurations. The current design of DTT embed 54 divertor cassettes which must be anchored to the vacuum vessel using suitable fixation systems. This paper deals with the assessment of the Fixation System which must ensure compliance with multiple design requirements. The Remote Handling compatibility is one of the most demanding, since it includes a preloading phase of the divertor necessary to mitigate the shaking of the cassette due to dynamic forces. Moreover, the system must withstand the conspicuous loads acting on the Divertor due to the full or partial loss of plasma confinement (Disruptive Events). Different conceptual solutions have been developed and evaluated, resulting in the identification of the best conceptual solution fulfilling the project requirements. Structural verifications through Finite Element Method (FEM) have been performed on the Divertor-Fixation System, considering a set of load combinations, with particular focus on the most demanding case related to Slow-transient Vertical Disruption Event-downward.

Keywords

DTT, Divertor, Concept Design, FEA.

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PS2-5 ABSTRACT-6ab4
A. Plasma-Facing High Heat Flux Components

Effect of neutron irradiation on the W/CuCrZr joint strength

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One of the main challenges for the construction of DEMO, the first fusion demonstration reactor, is the power exhaust issue. To deal with it, EUROfusion has foreseen the buildup of DTT (Divertor Tokamak Test) facility which will be held in Frascati in Italy. It aims to test various divertor models which can be integrated and used under different plasma confinement configurations. The current design of DTT embed 54 divertor cassettes which must be anchored to the vacuum vessel using suitable fixation systems. This paper deals with the assessment of the Fixation System which must ensure compliance with multiple design requirements. The Remote Handling compatibility is one of the most demanding, since it includes a preloading phase of the divertor necessary to mitigate the shaking of the cassette due to dynamic forces. Moreover, the system must withstand the conspicuous loads acting on the Divertor due to the full or partial loss of plasma confinement (Disruptive Events). Different conceptual solutions have been developed and evaluated, resulting in the identification of the best conceptual solution fulfilling the project requirements. Structural verifications through Finite Element Method (FEM) have been performed on the Divertor-Fixation System, considering a set of load combinations, with particular focus on the most demanding case related to Slow-transient Vertical Disruption Event-downward.

Keywords

W/Cu joint, W/CuCrZr joint, neutron irradiation, ITER-specification monoblock, joint strength, microhardness.

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PS2-7 ABSTRACT-f5c0
A. Plasma-Facing High Heat Flux Components

Engineering tool to optimize for leading edges in a full-tungsten divertor for W7-X

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The Wendelstein 7-X divertor is designed to be intersected by magnetic field lines at very shallow angles. Because of inevitable steps between divertor plates, small exposed areas called ′′leading edges′′ can be intersected almost perpendicularly by magnetic field lines and thus receive highly increased heat fluxes. High levels of fuel retention due to co-deposition make carbon-based materials such as CFC, currently used as plasma facing material for the W7-X divertor, incompatible with a tritium-based fusion reactor. As part of an ongoing investigation for a tungsten or tungsten-heavy alloy divertor for W7-X, an engineering tool for the optimization of such leading edges is being developed. In W7-X, particle fluxes can impinge from opposite directions on a same target surface in different magnetic configurations. Thus, the usual strategy of “shadowing” neighboring divertor target plates relative to each other, in order to avoid leading edges entirely, is not applicable. The tool uses a dataset compiled with a commercial thermal analysis software to match heat flux and incidence angle resulting from EMC3-Lite simulations to a maximum temperature experienced at a potential leading edge. All major magnetic configurations in a range of plasma pressures are evaluated simultaneously, while also accounting for manufacturing and assembly tolerances. The areas where the particle flux impinges from different directions and where the potential leading edge temperature exceeds the material limits are identified as "problematic" areas that cannot be avoided by either chamfering or tilting of the target element. The overall geometry of the divertor is subsequently changed iteratively to avoid these "problematic" areas. Thanks to the speed of the newly developed EMC3-Lite code, a full iteration can be completed locally in few minutes, opening the door to a fast exploration of the vast design space.

Keywords

Leading edge, divertor, tungsten, tungsten heavy alloy, full-W, optimization, engineering tool, EMC3-Lite, W7-X, Wendelstein 7-X, assembly tolerance, manufacturing tolerance, IPP, Max-Planck Instiut für Plasmaphysik, TUM, Technische Universität München.

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PS2-8 ABSTRACT-fba8
A. Plasma-Facing High Heat Flux Components

Brittle Fracture Assessment for Tungsten and Tungsten alloy components

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Tungsten materials are promising candidates for armour applications in future nuclear fusion reactors due to their good thermo-physical properties. A particular challenge in the design of tungsten components is the inherent brittle behaviour in the temperature range below the DBTT. In addition to common deterministic design rules, suitable probabilistic design rules are required for specific failure modes, like brittle fracture, to include details about failure probability. Aiming these rules, it is decisive to get database for fracture toughness as a function of temperature and probability. To build this database, results from fracture mechanics and four-point bending tests are required. Four-point bending tests support the determination of the Weibull parameters needed for the prediction of failure probability. Fracture mechanics test determines the fracture toughness and together with dedicated Finite Element (FE) Analyses, the probabilistic aspect can be considered for each temperature of interest. In these FE-analyses, a cohesive zone model (CZM) is utilized to simulate the fracture process observed in fracture mechanical tests including the Weibull parameters. Thereby the dependence on the orientation of the grain texture due to their manufacturing process shall also be taken into account. Results from (modified) Weibull analysis between room temperature and 400 °C and fracture toughness test results for several orientations as well as an approach for the link to the CZM will be presented and discussed. In addition, the overall scheme for Brittle Fracture Assessment (BFA) of tungsten and tungsten alloy components will be introduced.

Keywords

Brittle fracture, Weibull analysis, tungsten, Cohesive zone modelling.

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PS2-11 ABSTRACT-243d
A. Plasma-Facing High Heat Flux Components

Manufacturing and high heat flux testing of zirconium carbide tungsten monoblock for CFETR

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The divertor is the core component used for ash and heat removal in Tokamak devices. Monoblock are currently the most widely used plasma facing components. With the continuous improvement of EAST parameters, the heat flux received by the divertor tungsten string gradually increases. In high power operation mode or CFETR, the temperature of pure tungsten may exceed its recrystallization temperature. Therefore, it is necessary to use a more heat-resistant alloy tungsten material to replace pure tungsten. In this paper, zirconium carbide tungsten is used instead of pure tungsten. The recrystallization temperature of zirconium tungsten carbide is about 300 degrees Celsius higher than that of pure tungsten. The monoblock was made of zirconium tungsten carbide and subjected to thermal fatigue tests of 10 MW/m2 for 1000 times. The test results showed that the zirconium carbide tungsten monoblock passed the test, with a maximum surface temperature of 850 degrees, and no defects were detected before and after the test. The heat exchange ability of zirconium carbide tungsten string is good.

Keywords

Divertor; Monoblock; Zirconium carbide tungsten.

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PS2-12 ABSTRACT-24b3
A. Plasma-Facing High Heat Flux Components

Modelling of NBI shine-through in ITER PFPO phase to limit heat fluxes on first wall

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The occurrence of localized high heat fluxes on the ITER first wall can limit the lifetime of plasma-facing components. This can be the case of excessive shine-through losses due to high-energy Neutral Beam Injection (NBI) in low density plasmas. Of particular concern is the so-called Pre-Fusion-Power-Operation (PFPO) phase of ITER with low-density H or He plasmas and H NBI, at nominal energy of 870 keV and power of 33 MW. According to the current design, the energetic shine-through neutrals will partly penetrate into a gap between two adjacent first wall panels, reaching unprotected blanket shield blocks. The use of NBI will be therefore limited to densities that guarantee acceptable power fluxes on the first wall and blanket shield blocks, as first evaluated in [Singh New J. Phys. 2017].

In this work we tackle the NBI shine-through problem in the ITER PFPO phase, extending previous modelling assumptions and exploiting the IMAS modelling framework. An ad-hoc workflow has been generated, capable of a wide-range of parameter scans in plasma averaged density, density profile peaking (not extensively considered previously) and NBI energy/power. Two NBI ionization codes (BBNBI [Asunta Comput. Phys. Commun. 2015] and NEMO [Schneider Nucl. Fusion 2011]) are employed and their results compared. The BTR code [Dlougach Appl. Sci. 2022] is used to predict the beam intensity distribution at the first wall, in order to estimate the heat flux on plasma-facing components due to the calculated shine-through losses.

The present work reviews first wall heat flux calculations and previous estimations of the resulting ITER NBI operability in plasma density, with novel numerical methods in an extended parameter range. The use of IMAS allows code comparisons in order to discuss the impact of modelling choices on final results.

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

Keywords

ITER, NBI, heat flux, first wall, shine-through, IMAS.

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PS2-13 ABSTRACT-26e7
A. Plasma-Facing High Heat Flux Components

Coupled Model for Liquid Lithium Plasma Facing Components

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Application of liquid lithium Plasma Facing Components (PFC) in fusion devices provides several advantages, e.g., a renewable protective cover, enhancement of power exhaust and improvement of energy confinement via particle pumping. However, there are also many technological challenges associated with the controlled delivery and extraction of liquid metals in the fusion device environment, characterized by high heat fluxes and strong magnetic fields. Several concepts of liquid metal PFC have been proposed, including a new PFC design [Khodak and Maingi, NME, 2021] where a porous wall is used to stabilize the liquid metal surface, while magnetohydrodynamic drive is used to push the liquid metal flow inside the PFC. This arrangement allows efficient heat exhaust, and enhanced control of the liquid metal surface temperature, leading to spatial control of evaporation of liquid lithium on the plasma interface. This feature is particularly attractive when vapor shielding is introduced for enhanced volumetric plasma heat dissipation [Emdee and Goldston, NME 2023]. Numerical analysis informs the design choice and operating window of the liquid metal PFC concept. Coupled analysis of boundary plasma together with the surrounding boundary structures is required. To achieve this goal, PPPL is developing a comprehensive multi-physics model for modeling of PFCs in fusion devices. The model includes the fluid-kinetic code SOLPS-ITER and the flow and heat transfer code CFX from ANSYS. SOLPS-ITER code was augmented with the liquid metal boundary condition algorithm, allowing direct two-way coupling of the plasma analysis with the two-dimensional analytical slab flow model which includes heat convection in the liquid metal PFC. Results of this coupled analysis are used as a boundary condition for detailed 3D computational fluid dynamics magnetohydrodynamic analysis. Results of the 3D analysis can be used as a boundary condition for the new SOLPS iteration, to achieve coupling and consistency with 3D PFC analysis.

Keywords

Plasma Facing Components Liquid Lithium Numerical Analysis.

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PS2-14 ABSTRACT-29c7

A. Plasma-Facing High Heat Flux Components

Thermal and structural analyses of an additive manufactured panel mock-up for Wendelstein7-X

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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. A series of the stainless steel wall protection panels is positioned behind the divertor pumping gaps for average stationary heat loads of 100 kW/m². These panels were produced by electron beam welding of stainless steel parts to build the housing of the cooling channels. This technology was more demanding than foreseen due the complicated 3D weld seams required by the shaping of the panels. An innovative technology which is more adapted to the panel geometry is additive manufacturing. It intends to bring significant advantages by printing the housing as a single piece without welds.

The paper introduces the thermal and structural analyses of an additive manufactured panel mock-up performed with ANSYS V2022R2. The purpose of the calculation is to compare the thermal, hydraulic and structural performance of the improved design based on additive manufacturing and the present one. The welding process dictated the design of the cooling channel that induced local recirculation areas, additional local pressure drops, non well cooled areas for weld seams integration. In addition the weld strength assessment required special codes and methods, which cost more time and efforts. Without these constraints, additive manufacturing offers the opportunity of more freedom for the design of the cooling channel to reach better heat transfer performance of the panel. The calculation will provide hints for further panel improvement.

Keywords

Stellarator, Wendelstein 7-X, plasma facing components, thermomechanical análisis.

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PS2-15 ABSTRACT-3164
A. Plasma-Facing High Heat Flux Components

Screening of the metal foil from energetic neutrals

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The Direct Internal Recycling features a metal foil pump as the prime candidate for the separation of a majority of unburnt fuel from the torus exhaust. This technology works using plasma-driven permeation (based on the superpermeation process), which we experimentally research in the setup HERMESplus.

In this work, we analyse the origion of energetic neutrals coming from the core of the microwave plasma as responsible for damage to the foil surface monolayer and hence, for the plasma driven permiation (PDP) degradeition. The presence of noble gases in the hydrogen plasma causes especially large amounts of energetic neutrals, leading to higher permeation fluxes in short-term but also to more damage to the monolayer.

Several mechanisms of higher energetic neutrals origination is considered, including the acceleration of H^+ ions in the electric sheath layer. It is explained, why the presence of Ne helps to increase the permeation, whereas the introduction of Ar increased the number of fast neutral atoms and degrades the PDP.

We also consider the screening of the MF from fast neutrals by applying the electric biasing. It is shown that at some loop voltage biasing helps to protect membrane from the energetic atoms. Effect of negative and positive biasing on the PDP is discussed.

Keywords

Metal foil pumps, energetic neutrals, fusion reactor, DEMO.

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PS2-16 ABSTRACT-3327
A. Plasma-Facing High Heat Flux Components

Development of non-destructive X-ray based techniques for evaluation of Tungsten-metal joints quality

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This study aims to test the joints between tungsten and different heatsink or structural materials like Cu-alloys or other materials used in nuclear fusion reactors, which is a major concern due to thermal and mechanical stresses. The study uses X-ray microtomography and microbeam fluorescence techniques to assess the quality of the joints and provide feedback on the development of joining technologies.

Small samples containing the interfaces of various multi-layered materials (W-(Cu–ZrO2)-CuCrZr, W-(Cu–SiC)-CuCrZr, and W-(V–ZrO2)-CuCrZr) were analyzed using a custom-built microCT system available at http://tomography2.inflpr.ro/. The analysis revealed the presence of pores, inclusions, and microcracks, mainly at the W-Cu interface.

After microCT inspection, the multi-layered samples were exposed to electron beams of 6 MeV at an energy density of about 4 kJ/cm2 at a LINAC facility. The behavior of the interface layers under irradiation, as revealed by post-mortem metallographic imaging, was found to be well-correlated with their 3D morphology visualized by microCT, specifically in the case of Cu-based TB materials.

For tomography analysis of parts of a massive W block bonded directly or brazed with Cu using high penetration power, an in-house built microCT system with a maximum beam energy of 320 keV was utilized. X-ray micro-beam fluorescence elemental composition mapping was also performed to record interface profiles, and any discontinuities in the W-Cu interfaces were confirmed through microXRF line profiles.

The studies indicate that X-ray-based non-destructive techniques have advanced in maturity for evaluating components that consist of multi-layered joints made of dissimilar materials.

Keywords

X-ray microtomography, XRF, joints of dissimilar materials.

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PS2-17 ABSTRACT-380c
A. Plasma-Facing High Heat Flux Components

Integrity assessment of armour materials for innovative high heat flux components for fusion reactor by the ultrasonic method

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Within the frame of the EUROfusion Consortium the "*Characterization of armour, heat sinks materials and joints*" sub-project inside the Work Package 'Material' (WP-MAT) has been dedicated to test different tungsten (W) monoblock mock-ups and new advanced monoblocks materials for the divertor target in the EU-DEMO demonstration fusion power plant.

ENEA's Special Technologies Laboratory has performed several non-destructive tests due to assess the integrity of the several advanced W monoblocks as well as their welding to the cooling tubes, with and without the use of copper interlayer.

The non-destructive method used is the ultrasonic pulse echo technique (UT). The UT examinations were conducted on monoblocks, and related jointed mock-ups, made with different materials. Potassium-doped W laminates, W fiber reinforces W and W-matrix with ditungsten carbide W_2C inclusions are three different W-alloys armour materials that within the WP-MAT sub-projects revealed better fracture toughness and recrystallization resistance compared to pure W after the high thermal stress during operation. W-10Cr-0.5Y and W-10Cr-0.5Y-0.5Zr are self-passivating W-based alloys that provide an important safety advantage compared to pure W avoiding the formation of volatile and radioactive tungsten trioxide in case of a loss-of-coolant accident with simultaneous air ingress.

The ultrasonic pulse-echo technique allows to identify not only the size and position of the defects in the plane orthogonal to the ultrasonic beam, but also their depth in the material. During the analysis the samples are immersed in water and examined with an automatic system equipped with 4 degrees of freedom that allows to perform both flat and cylindrical scanning. The frequency of the probe has been chosen, time by time, to obtain the maximum resolution compatible with the thickness to be analyzed.

Keywords

Ultrasonic technique, Tungsten Alloy, Divertor, Plasma facing units, Composite materials.

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PS2-18 ABSTRACT-41fb
A. Plasma-Facing High Heat Flux Components

Optimization in cooling design of poloidal horseshoe limiter concept for JA DEMO

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Design of the poloidal horseshoe limiter concept for JA DEMO is ongoing. The role of the limiter is to mitigate the charged particle load deposited on the breeding blanket. The apex of the limiter is protruded from the blanket first wall to shade it from charged particles. The protruded limiter is continuous poloidally on the first wall except for the divertor area, and discretized toroidally at 90 degrees interval. The tungsten monoblock with Reduced Activation Ferritic/Martensite(RAFM) steel pipe is faced on the plasma for high heat removal capability and neutron irradiation tolerance. For steady state operation, each heat removal capability of the limiter and the blanket has been simulated to be 4.6 MW/m^2 and 1.0 MW/m^2, respectively. In the previous study, the distance between the blanket and the plasma surface was set at the beginning, which led to high heat flux deposition on the protruded limiter and small margin to the heat removal capability. Besides, the feasibility of cooling using coolant of pressurized water (15.5 MPa, 290 °C) and the maintainability for the limiter component were required to be clarified.

In this study, the limiter apex has been reconsidered by optimizing its distance to the plasma surface firstly, and the position of the blanket has been arranged subsequently. Taking the heat removal capabilities as constraints, the heat load of the limiter and the blanket at steady state operation and rump-up scenario is calculated with the three-dimensional magnetic field line tracing code. To cool the heat load of the limiter, the heat balance has been analyzed and a modular limiter concept is modeled. The cooling circuit for the limiter modules is designed to meet the heat removal and space requirement. The strategy to replace the limiter independently and remotely is also studied and reflected to the limiter design for JA DEMO.

Keywords

JA DEMO, first wall shaping, limiter, plasma facing component, maintainability.

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PS2-20 ABSTRACT-4601
A. Plasma-Facing High Heat Flux Components

Overview of Japan-US Collaboration FRONTEIR Project —Fusion Research Oriented to Neutron Irradiation Effects and Tritium Behavior at Material Interfaces—

Yuji Hatano[1], Daniel Clark[2], Takehiko Yokomine[3], Takuya Nagasaka[4], Yutai Katoh[5], Tatsuya Hinoki[3], Naoyuki Hashimoto[6], Lauren M Garrison[5], Takaaki Koyanagi[5], Josina W Geringer[5], Weicheng Zhong[5], Yasuhisa Oya[7], Teppei Otsuka[8], Masashi Shimada[9], Robert Kolasinski[10], Masatoshi Kondo[11], Naoko Oono[12], Junichi Miyazawa[13],Bruce A Pint[14], Jiheon Jun[5], Marie Romedenne[5]

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The FRONTIER collaboration started in April 2019 to provide the scientific foundations for reaction dynamics in interfaces of plasma facing components (PFCs) for DEMO reactors. The project has 4 task groups to reach this goal.

The objectives of Task 1 are to understand neutron-induced microstructure modification and the consequent changes in mechanical and heat transfer properties of interfaces between plasma-facing material and structural material. This task performs neutron-irradiation in the High Flux Isotope Reactor (HFIR) and post-irradiation examinations at Oak Ridge National Laboratory (ORNL). Task 2 performs permeation experiments with layered materials in the liner plasma machine Tritium Plasma Experiment (TPE) at Idaho National Laboratory (INL) to examine H/D/T transport through the interface before and after neutron irradiation. The task also examines the oxidation of neutron-irradiated W materials by steam, air etc. and tritium release to construct fundamental database for evaluation of radioisotope emission under accidental conditions. Task 3 studies the corrosion characteristics of liquid Sn for a divertor coolant with and without neutron irradiation. In-pile corrosion tests in the HFIR using a specially designed irradiation capsule will be

performed. Task 4 aims to consolidate the results of each task interdisciplinary, link them organically as a phenomenon appeared in the PFCs, and sublimate to an engineering model. Details of neutron irradiation and plans of post-irradiation examination will be reported in the presentation.

Keywords

Plasma Facing Component, Joined Material, Composite Material, Liquid Metal Divertor.

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PS2-21 ABSTRACT-491a
A. Plasma-Facing High Heat Flux Components

Research on a renewable extruded plasma-facing divertor material for magnetic fusion energy reactors

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Laboratory research has been conducted toward developing a renewable plasma-facing wall design for application to magnetic fusion energy reactor divertors. The concept uses low-Z (carbon or boron) pebble rods extruded in a brush pattern out of extrusion channels inside divertor tiles. The pebble rods are initially formed out of a slurry of mixed pebbles and volatile hydrocarbon binder and solidify during extrusion by being baked out under vacuum. Upon experiencing reactor-relevant heat fluxes, the pebble rods experience thermal shock and crumble into constituent pebbles, which fall in gravity and are recovered to be recycled, remove heat from the system, and recover tritium fuel. Pebble rods consisting of 1 mm pebbles of carbon, boron, silicon carbide, silicon nitride, or boron nitride have been produced and tested under reactor-relevant heat loads, 5 – 50 MW/m^2. Front-surface disintegration and intact pebble recovery have been demonstrated. Comparison with finite-element simulations indicates that the disintegration can be understood in terms of pebble thermal expansion resulting in inter-pebble matrix cracking. Sufficiently low front-surface outgassing and sufficiently low extrusion channel friction for use in a magnetic fusion energy (MFE) reactor environment have been demonstrated. Tunability of the pebble rod design for different heat loads has been achieved by changing pebble thermal conductivity. The motivation of this work is to develop a divertor concept that will allow (a) low-Z for good core performance, (b) resistance to reactor heat loads, (c) resilience to high erosion, (d) good neutron resistance, and (e) tritium fuel recovery. The outlook for application in MFE device divertors and necessary future developments will be discussed.

Keywords

Divertor, first wall, tritium retention.

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PS2-22 ABSTRACT-4979

A. Plasma-Facing High Heat Flux Components

Hypothetical porous medium concept as a virtual swirl tape: A novel modelling technique towards efficient CFD simulation of swirl tape cooling pipe

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The divertor is a critical in-vessel component of a fusion reactor, responsible for power handling and impurity removal via guided plasma exhaust. In power plant scale reactors, as in the EU-DEMO, the high thermal loads to which the divertor targets are exposed require a suitable cooling circuit and must therefore rely on heat transfer enhancement techniques to improve the cooling performance. In the case of the EU-DEMO, Swirl Tape (ST) inserts are employed inside the target cooling channels. Unfortunately, an accurate 3D CFD-based thermohydraulic assessment of the target cooling system equipped with STs requires a high computational cost and a laborious pre-processing modelling effort due to the complex geometrical feature of the twisted ST. Thus, for parametric design studies requiring a heavy computational load, a cost-efficient simulation method is desired.

To this end, we developed a simplified CFD simulation technique based on an equivalent porous medium concept, which enables the reduction of computational costs without compromising the numerical quality of the results. The key idea is to replace the physical presence of the ST in the coolant domain with a hypothetical porous medium which is supposed to deliver equivalent thermohydraulic effects as an actual ST. In this work, several different porous medium models were studied as virtual surrogates of ST. The models were calibrated by means of optimisation techniques against the CFD simulations results obtained for detailed 3D ST models and tested to assess their predictive capability in terms of pressure drop and heat transfer coefficient. In this contribution, an application case is presented as an example of cooling performance estimation for the EU-DEMO divertor target (2022 CAD model). The proof-of-principle and feasibility are discussed. This technique was developed using the commercial CFD code ANSYS CFX, coupled with a multi-objective optimization algorithm available in the ANSYS Direct Optimization tool.

Keywords

DEMO, Divertor, PFC, Thermofluid-dynamics, CFD analysis, Optimization.

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PS2-23 ABSTRACT-8b63
B. Blanket Technology

Equilibrium distribution of lithium isotope between liquid metal and chloride molten salt containing CsCl

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This study aims to clarify the lithium isotope effects in equilibrium distribution between liquid metal and chloride molten salt containing cesium chloride (CsCl). To establish the tritium fuel cycle of fusion reactors, it is necessary to develop the lithium-6 (^{6}Li) enrichment technology. The lithium amalgam method by a two-liquid-phase chemical exchange is a well-known conventional method. However, there are concerns about its environmental impact because it uses toxic mercury. Recently, electrodialysis using lithium-ion conductors has been studied in Japan. As the amount of lithium recovered is larger, the isotope separation factor becomes closer to 1.00. Solvent extraction using crown ethers is being considered, especially in China, and lithium recovery is low in single-stage enrichment. Therefore, our research group focuses on a new two-liquid-phase system for ^{6}Li enrichment without mercury, liquid metal / molten salt system. This system uses lithium (Li) or lithium alloys such as lead lithium eutectic alloy (Li–Pb) as liquid metal, and for example, a mixture of lithium chloride (LiCl) and other alkali metals' chloride as molten salt. As an experimental procedure, pieces of metal and powdered salt were sealed in capsules made of SUS316L. The capsules were rotated in an electric furnace at 4 rpm for more than 12 hours at a constant temperature. After heating, each sample was taken out and prepared in dilute nitric acid solution. Inductively Coupled Plasma Mass Spectrometry (ICP-MS) measurement was performed and the lithium isotope ratio of ^{6}Li to ^{7}Li was obtained for each sample. In both systems Li / LiCl–KCl–CsCl eutectic and Li–Pb / LiCl–KCl–CsCl, the isotope separation factor was obtained. The additional runs using LiCl–CsCl eutectic and more detailed ICP-MS measurements are ongoing to evaluate more accurately lithium isotope effects in these systems.

Keywords

Lithium isotope separation, isotope effects, lithium, lead lithium, liquid metal, molten salt.

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PS2-24 ABSTRACT-8dda
B. Blanket Technology

Quest for magneto-convective unstable flow regime in horizontal MHD duct flows

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The molten metal coolants in nuclear fusion reactor would experience magnetohydrodynamic (MHD) effects due to the presence of plasma confining magnetic field. Recent experiments have revealed the presence of a low frequency high amplitude temperature fluctuations at the outlet of heated duct under transverse magnetic field. These fluctuations don't occur at all flowing conditions, but appear only for certain combination of fluid velocity, heat flux and magnetic field. There is a possibility that such temperature fluctuations will be generated in the blankets responsible for neutronic heat removal. These fluctuation would act as a cyclic load on the walls of the blanket, weld joints as well as in the nearby components like heat exchanger, valve, sensors etc. Thereby, limiting the life time of such components. Hence, there is a need to study their behavior in fusion relevant conditions. As an initial step, a parametric numerical study has been performed to find the flow regime, where such unstable mangnetoconvective flow occurs. A horizontal rectangular duct is considered for the simulation, which is heated at the bottom and the magnetic field is applied horizontally transverse to the flow. Such flowing condition would appear in blankets located at the poloidal upper parts of a typical fusion reactor. The transient simulations have been carried out using COMSOL to capture the temperature fluctuations and various non-dimensional parameters like Richardson number (Ri), Hartmann number (Ha), Reynolds number (Re) are varied up to fusion relevant conditions. E.g. Ha is varied up to 10^3, Ri is varied up to 200, Re is varied up to 10000 etc. It was observed that the velocity profile is significantly altered in the unstable flow regime compared to the stable flow regime, which manifests into observed temperature fluctuations.

Keywords

Liquid metals, Nuclear Coolants, Instabilities.

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PS2-25 ABSTRACT-4472
B. Blanket Technology

Tritium Breeding Ratio Optimization in Simple Multi-Layer Blanket with Genetic Algorithm

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Investigation on tritium production via fast neutrons generated by magnetically confined plasma in a tokamak device is conducted for optimization of tritium breeding ratio (TBR). The blanket is configured as a 1-dimensional multiple layer of materials for the wall, moderation, reflection, and tritium productions, and we adopted genetic algorithm to select the optimal material on each order and location. The configuration selected in the process is evaluated in aspect of tritium production to incoming neutron ratio. To construct the algorithm, we parametrized the ratio of tritium production by neutron energies from 0 to 14 MeV for the materials with Geant4 Monte Carlo simulation toolkit, and the simulated tritium in the algorithm are evaluated for the selection of parent for the next generation. Result configuration from the algorithm is put back to the Geant4 simulation for verification, and TBR is evaluated with blanket designs in other facilities.

Keywords

Genetic Algorithm, Tritium Breeding Ratio, Optimization.

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PS2-26 ABSTRACT-9c69
B. Blanket Technology

Neutronic Studies for the Dual Coolant Lithium Lead Breeding Blanket HELIAS design

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The Stellarator Power Plant Studies Prospective R&D Work Package was settled to bring the stellarator engineering to maturity, so that stellarators and particularly the HELIAS (Helical-Axis Advanced Stellarator) configuration could be a possible alternative to tokamaks.

This brings the necessity to develop a fully functional Breeding Blanket (BB) that fulfils the requirements for such an HELIAS configuration. However, its complex geometry and the limited space between the plasma and the coils make difficult to implement the current *tokamak-oriented* BB concepts.

Exploiting previous experience in BB designs for DEMO tokamak, CIEMAT is leading the development of a Dual Coolant Lithium-Lead (DCLL) BB with high potentialities to answer the specific challenges posed by the complex HELIAS configuration.

Thanks to novel *ad-hoc* created tools, the DCLL BB design is being optimized by fast parametrized neutronic models of a 72º HELIAS period, implemented to address the viability of the new configurations under the neutronic point of view. Hence, Tritium Breeding Ratio (TBR) and damage responses have been primarily addressed to validate the concept through radiation transport simulations by MCNP code.

Furthermore, other advanced solutions have been proposed to simplify the remote maintenance and integration of the BB segments, as the use of fully detached First Wall (FW) (i.e. fingers or liquid metal Capillary Porous Systems (CPS)). This solution would allow switching the maintenance problem mainly to small FW panels that could be manageable with a more conventional maintenance scheme through ports. For that, neutronic analyses have been performed in support of different FW designs, to demonstrate the possibility to reduce the damage to the BB (and increase the availability of the machine) while keeping a viable TBR.

The possibility of using modified plasma-facing surfaces for smoothing the strong variations of the Neutron Wall Loading, has been also analyzed, addressing its impact on the TBR.

Keywords

DCLL, Breeding Blanket, HELIAS, TBR, neutronic.

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PS2-27 ABSTRACT-5f0a
B. Blanket Technology

Fabrication and characterization of enamel coating as a promising hydrogen and oxygen diffusion barrier

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Functional coatings have been developed to reduce tritium permeation and corrosion by tritium breeders in fusion reactor blanket systems. Ceramic coatings such as metal oxides showed high hydrogen isotope permeation reduction performance; however, oxidization of steel substrates by oxygen diffusion through the coating at high temperatures causes degradation of the coating. In this study, we fabricated enamel coatings consisting primarily of silicon dioxide that has a low oxygen diffusion coefficient and analyzed microstructure and deuterium permeation properties as a promising functional coating.

The enamel coatings were fabricated on 430 stainless-steel substrates by dip-coating and heat-treatment at 600–850 ºC for 10–60 min in an argon-hydrogen mixture gas. The coatings were characterized through surface and cross-sectional observation using a scanning electron microscope and X-ray diffraction. Gas-driven deuterium permeation measurements were conducted for some the coatings at the driving pressure of 10–80 kPa and 400–700 ºC.

The coating thickness was 18–20 µm and depended on the heat-treatment temperature. The coating heat-treated at 650 ºC showed numerous cavities at the surface, whereas the sample heat-treated at 700 ºC and more showed a dense surface structure. This result suggests that the grains containing in the enamel liquid do not melt sufficiently below 650 ºC. Deuterium permeation flux of the coating heat-treated at 850 ºC for 10 min decreased with the test temperature and was finally three orders of magnitude lower than the uncoated steel. In the presentation, the permeation behaviors for the coatings heat-treated at lower temperatures will be also discussed.

Keywords

Ceramic coating, permeation, tritium, silicon dioxide.

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PS2-28 ABSTRACT-f33d
B. Blanket Technology

Buoyancy-assisted thermal convection and tritium diffusion in an upward magnetohydrodynamic duct Flow

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In this work, 2D numerical simulations are conducted to study magnetohydrodynamic (MHD) thermal convection and tritium diffusion in the Dual-Coolant Lead-Lithium (DCLL) blanket. We consider a vertical rectangular duct with electrically and thermal insulated walls without mass fluxes. The lead-lithium flows upward under the influence of Lorentz force by the strong transverse magnetic field and buoyancy forces by the large volumetric heat source and tritium concentration source. The finite volume method is employed to solve thermal and solutal buoyancy-assisted MHD momentum equation, energy equation and tritium concentration transport equation. The characteristics of flow and heat transfer, and distribution of tritium concentration are investigated by numerous of simulation with parameters ranging: Hartmann number of 10^3 to 10^4, Reynold number of 10^3 to 10^4, both thermal and solutal Grashof numbers above 10^12. A parametric study about Ha, Re and Gr of the flow instability is performed. The results show that the buoyancy by exponentially distributed heat source and tritium concentration source makes the velocity field become nonuniform in the main flow. Lorentz force of magnetic field can reduce the nonuniformity and stabilize the flow field. The scaling relationship between critical parameters of instability has been obtained. This work will provide basis and support to the blanket design.

Keywords

MHD, thermal convection, tritium diffusion, buoyancy flow, instability.

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PS2-29 ABSTRACT-fe8f
B. Blanket Technology

Numerical study of MHD mixed convection flow in the the EU DEMO WCLL breeding blanket

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The Water-Cooled Lithium Lead (WCLL) breeding blanket is a proposed candidate option for European DEMO nuclear fusion reactor. PbLi occupies the main areas of the blanket volume where the interaction of the Lorentz forces(Ha ≈9800), intense buoyancy forces (Gr ≈ 1.77 ∗ 10^10), and pressure-driven flow (Re ≈ 236.77) dominate the MHD/heat transfer phenomena of the bulk region. Numerical simulation is performed using 3-D DNS to imitate the dynamics of the PbLi accounting for the effect of the intense magnetic field used for the plasma confinement and the high nonhomogeneous heat source transformed by nuclear heating. Due to the effect of the buoyancy forces, several middle-sized circulations which are elongated along the magnetic field direction occupy the plenum region between the cooling pipes and impact the heat transfer there. Electrical interaction between the liquid metal and the conducting wall leads to high-velocity flow jets, the peak velocities of which grow with the magnitude of the magnetic field accordingly. Local Nusselt and Grashof numbers that applied to assume the transferring phenomena have been obtained in the different regions of the system.

Keywords

WCLL, MHD, mixed convection.

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PS2-30 ABSTRACT-41ef
B. Blanket Technology

Magneto-hydraulic and MHD mixed convection flow in a rectangular channel filled with streamwise obstacles

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The Water-Cooled Lead Lithium (WCLL) breeding blanket (BB) is one of the two leading candidates for implementation as driver blanket in the EU DEMO reactor, which is expected to start operation in the late 2050s and serve as a BB test facility. One of the main risks associated with the current WCLL concept is the large number of welds necessary to realize its horizontal (toroidal-radial) breeding zone cooling system and stiffening structure, which negatively affects its reliability. Hence, an alternative variant concept has been proposed which relies on vertical (radial-poloidal) "double-bundle" cooling tubes and stiffening plates: the WCLL double-bundle (WCLL-db). In this paper, we report the results of numerical analyses performed to characterize the magnetohydrodynamic (MHD) regime in the WCLL-db where the liquid metal breeder flows vertically but it is obstructed by a large number of electrically conductive obstacles aligned with the streamwise direction.

Direct numerical simulations are performed with the aid of computational MHD tools to characterize the flow features and pressure losses in the WCLL-db breeding zone. The presence of streamwise obstacles affects the velocity distribution since the cooling tubes provide additional paths for the electric current closure. The most striking feature is the breaking down of the classic MHD slug core flow into many separate smaller regions, separated by internal layers. These occur tangential to the pipe walls, featuring velocity overshoots, and tend to propagate along the imposed magnetic field. A moderate (5-10%) pressure penalty is found compared with a channel devoid of obstacles. The role of buoyancy forces is also investigated by comparing simulations performed for a purely magneto-hydraulic flow and a MHD mixed convection regime in the same conditions. In particular, the stability of the velocity jet formed close to the chamber wall at the operative magnetic field intensity is assessed.

Keywords

MHD, pressure drop, liquid metal, flow around obstacles, breeding blanket, WCLL-db.

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PS2-31 ABSTRACT-45c0
B. Blanket Technology

Analysis and modelling of gas-liquid contactors for tritium extraction from eutectic PbLi

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Reliable, efficient, and fast tritium extraction from eutectic PbLi is necessary to achieve the required tritium breeding ratio, minimisation of tritium inventories, and mitigation of unwanted movement of tritium in fusion power plants (FPP). In this work we analyse and model gas-liquid contactor (GLC) tritium stripping columns, one of the leading options for tritium extraction in the WCLL EU-DEMO project. An in-house model was developed with the aim of providing insights into the effect of several design and operation parameters, namely, flow rates (and their ratio), temperature, the addition of H2 in the stripping gas, and the isotope effect if the same tests conducted with tritium. The model developed combines Sieverts' Law and the mass balances across the GLC; crucially, mass transfer coefficients are needed to solve all the equations in the mass balance. In collaboration with the TRIEX-II facility (ENEA, Italy), the experimental data from recently completed GLC campaigns is analysed to estimate the mass transfer coefficients of protium transport in the flowing liquid metal. The mass transfer coefficients obtained are within the expected range of conventional contactors and previous liquid metal GLCs. Furthermore, there is a positive correlation between the mass transfer coefficients and the liquid metal flow rate, as expected from the theory. These results were subsequently used to develop a model of a GLC where the stripping gas (normally pure He) includes H2, and deuterium is dissolved in the PbLi (instead of protium). We calculate the mass transfer enhancement factor different H2 concentrations provide, and we highlight the trade-offs of this strategy, namely dissolving protium into the breeder blanket and mixing protium with deuterium (or tritium in the FPP) that must be separated in the fuel cycle. Sensitivity and uncertainty analyses show opportunities to refine the models and the upcoming experimental campaigns.

Keywords

Tritium Extraction, Gas Liquid Contactor.

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PS2-32 ABSTRACT-490c
B. Blanket Technology

Helical-shaped Double Wall Tubes solution for the Breeding Zone cooling in the WCLL Breeding Blanket

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One of the two most promising Breeding Blanket (BB) concepts to be chosen as driver design of the EU-DEMO fusion reactor is the Water-Cooled Lead Lithium (WCLL) BB. A crucial point of this key component is the cooling of the BB structural elements and of the Lithium-Lead alloy, used as neutronic multiplier and Tritium breeder. Indeed, the high neutronic flux needed for breeding implies high volumetric power affecting the Breeding Zone (BZ) materials. The current BB layout employs water-cooled Double Wall Tubes (DWTs) with a C-shaped configuration for the BZ cooling. Despite the WCLL BB have reached a mature design in the last years, some open issues remain to be solved to increase the reliability and performance of this technology. The present paper describes a promising upgrade of the BZ cooling layout adopting helical-shaped DWTs. This solution has the potential to increase the BB reliability, cooling performance, Tritium Breeding Ratio and cooling water flow-path simplicity. The new proposed layout is described in the paper, along with the results of CFD analyses carried out to investigate the new cooling performances in the BZ. Advantages and drawbacks with respect to the current BB layout have been also highlighted in the paper.

Keywords

WCLL, Breeding Blanket, Doble Wall Tubes, CFD analysis, helical-shaped DWTs.

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PS2-33 ABSTRACT-e23e
B. Blanket Technology

Implications of Reduced Geometric Modeling on MHD Predictions in a Liquid Metal Breeding Blanket Concept

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In liquid metal(LM) breeding blankets, the interactions between the high electrically conductive liquid metals and the large plasma-confining magnetic fields give rise to strong magnetohydrodynamic(MHD) effects. With an additional substantial convection effect in the blanket/FW, the prediction of flow features and heat transport phenomena in LM breeding blanket designs under such complex conditions is indispensable but numerically challenging. Furthermore, the complexity of model geometry introduced by the LM flow configuration substantially increases the requirements of computational numerical tools. In this study, the prediction of lead-lithium flows in a prototypical equator unit cell of a water-cooled lithium lead(WCLL) breeding blanket(BB) under designed operational conditions is performed by COMSOL Multiphysics code. In addition, several reduced geometrical models of this WCLL unit cell including a corner duct with a full or a truncated inlet and outlet channel length and a mid-region single-channel geometry were introduced and analyzed against the results from the full geometry study.

In the evaluation of the prediction in reduced geometrical models, simplification with a truncated inlet and outlet channel length should be avoided as both temperature distribution and thermal convective flow are of significant interest inside the channel. Compared with the full geometry simulation, the mid-region single-channel model was able to reproduce temperature distribution as well as the main flow features in the plenum region near the first wall. However, in the plenum area apart from the first wall, buoyance-induced flow circulation was diverted leading to more than 10% of flow imbalance across different channels. Such a flow rate imbalance was not observed in this single channel model, of which the maximum velocity is found to be three times larger than the calculated value found in the full model. The applicability of numerical prediction in using reduced geometrical models is further evaluated through comparisons between COMSOL and Fluent simulations.

Keywords

WCLL liquid breeder blanket, MHD Mixed Convection, Numerical Modeling.

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PS2-34 ABSTRACT-507c
B. Blanket Technology

Gas-Liquid Contactor experimental campaign in TRIEX-II facility

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The Gas-Liquid Contactor (GLC) is the reference tritium extraction technology for the Test Blanket Module of ITER and historically it has been the most studied among the technologies, with experiments performed at CEA with MELODIE facility in the late '90s and early 2000s and later at ENEA Brasimone with TRIEX in 2006-2007. In the GLC, the extraction of the hydrogen isotopes from the LiPb is realized by using a purge gas (helium, or helium plus hydrogen) which is flushed in counter-current with a flow G with respect to the LiPb flow (L) in the column. Due to the low solubility of the hydrogen isotopes Q in LiPb and to its concentration gradient at the LiPb/gas interface, a mass transport with the passage of hydrogen isotopes from LiPb to the gas phase occurs. The main advantage of GLC is the larger experience with respect to the other technologies coming from years of tests and experiments and from its utilization in many conventional industrial fields. This paper will present the latest experimental results obtained in TRIEX-II facility with a GLC mock-up 1:1 scale with the TBM one. The tests were performed at 450°C with a LiPb flowrate in the range of 0.3-1.2 kg/s and with hydrogen and deuterium at different partial pressure in the range of 50-300 Pa.

Keywords

Tritium extraction, WCLL BB, Gas-Liquid Contactor, Lithium-lead eutectic.

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INTERNATIONAL SYMPOSIUM ON FUSION NUCLEAR TECHNOLOGIES

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PS2-35 ABSTRACT-5489

B. Blanket Technology

Investigation and Qualification on First Wall Fabrication Techno logies of Fusion Reactor Blanket in China

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The blanket is the key component of fusion reactor for energy conversion and tritium breeding, which will suffer 14.1 MeV fusion neutron irradiation, elevated temperature and high heat flux heat load. It proposes high requirements on the structural material and the structure of the blanket for heat transferring. On one hand, reduced activation martensitic/ferritic (RAFM) steel is chosen as the primary structural material of fusion reactor blanket for its advantages of low activation, neutron irradiation resistance and favorable properties at elevated temperature. On the other hand, in order to transfer the heat out of the blanket, the blanket components facing the fusion plasma are designed with intensive cooling channels meandering in limited thickness.

The blanket fabrication is one of the key issues of the blanket related technologies. Hot isostatic pressing (HIP) - diffusion bonding is considered as the preferred method for the manufacturing of the first wall. Based on the previous research, surface roughness is one of the key factors to affect the HIP joints quality. In this work, the effect of surface roughness on HIP joints of China low activation martensitic steel was investigated. Small specimens for HIP were prepared with different surface roughness, and the microstructures of the joints were analyzed with SEM and TEM, and the tensile properties and toughness of the joints were tested. Moreover, large scaled first wall of fusion blanket was fabricated to verify the HIP procedure with optimal surface roughness, and the joining analysis were performed by microstructure observation and mechanical property tests.

Keywords

Fusion Reactor Blanket, Reduced Activation Martensitic/Ferritic Steel, Hot Isostatic Pressing - Diffusion Bonding, Surface Roughness.

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PS2-36 ABSTRACT-3b8f
B. Blanket Technology

Experimental investigation of the corrosion behavior of Eurofer97 steel in contact with Lithium ceramic breeder pebbles under specific Helium Cooled Pebble Bed breeding zone atmosphere

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For specimens made of EUROFER in unconstrained contact to ceramic breeder pebbles, exposed to purge gas conditions for different durations, a chemical surface attack was observed which led to a significant reduction of fatigue lifetime (see Aktaa, J. et al.; FED 2020, https://doi.org/10.1016/j.fusengdes.2020.111732). In the cited reference, the samples were placed in a crucible together with the pebbles and heated in an oven up to 550°C under a continuous helium stream having 0.1vol.% hydrogen. To better reproduce the flow of the purge gas in the breeding zone of a Helium-Cooled Pebble Bed Breeder Blanket, the test was repeated by placing the same kind of samples in the helium loop HELOKA-HEMAT, where the gas is permanently circulated in a closed circuit. The composition of the coolant was constantly monitored with a mass spectrometer and moisture sensors. To keep the water content in the coolant below the required threshold, a zeolite getter bed was included in the circuit. Similar to the previous experiment, the samples were exposed to a mixture of helium and 0.1vol% hydrogen for a duration of 8, 16, 32 and 64 days. The holding capsules wall are made of perforated Eurofer97 sheets allowing the gas mixture to flush the pebble bed. A special procedure for the handling of the test section was implemented to avoid contact of the hygroscopic pebbles with air humidity during the preparation phase and during the extraction of the samples from the test rig.

Keywords

Lithium ceramic breeder pebbles, Helium cooling, Eurofer.

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PS2-37 ABSTRACT-5ba8
B. Blanket Technology

Evaluation of the impact of plasma operation scenarios on the fusion reactor blanket design using an integrated numerical plasma and neutronics analysis suite

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A very limited operation scenario has been used for fusion neutronics for the design of tritium breeding blankets [1]. However, many advanced operation scenarios are under development and the impact of choosing a plasma operation scenario for fusion neutronics, which requires integration of neutronics analysis with plasma analysis, needs to be evaluated. Refs. [2,3] considered an automated process from plasma analysis to neutronics analysis to create a viable fusion plasma neutron source. However, plasma analysis and neutronics analysis are not linked. In this work, we develop an integrated suite combining plasma modeling codes and neutronics codes. McCARD [4], a Monte-Carlo neutron-photon transport simulation code, is integrated into TRIASSIC [5], a tokamak reactor integrated automation suite for simulation and calculation, as a module for neutronics analysis of various plasma operation scenarios. McCARD integration allows TRIASSIC to perform neutronics analysis along with analytical and predictive plasma simulations. To validate this integrated suite, KSTAR experimental data were used to calculate the neutron wall loading via TRIASSIC with the McCARD module. Lastly, when applied to DEMO, the effect of various plasma operation scenarios on the design of the breeding blanket can be evaluated.

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Keywords

Plasma transport, TRIASSIC, neutronics, McCARD, plasma neutron source,.

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PS2-38 ABSTRACT-6076
B. Blanket Technology

Functional tests for water cooled ceramic breeder blanket system using full-scale mockups

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Intensive research and development activities have been conducted for preliminary design of a water- cooled ceramic breeder (WCCB) blanket system. As a part of these activities, some full scale mockups have been fabricated to verify feasibility, including manufacturing and inspecting procedures. This paper introduces an overview of the recent achievements in the mock-up testing on the WCCB blanket system. An blanket sub-module mockup, including hemispherical first wall, cylindrical body with built-in cooling channels, manifolds for coolant and tritium purge gas, have been manufactured using a reduced activation ferritic/martensitic steel, F82H with conventional industrial resources. The mockup have demonstrated leak tightness against coolant water pressure, and is to be tested at high heat flux test stand using 600 kW electron beam gun with 15.5 MPa / 573 K of cooling water. As for water cooling system component, a mock-up of pneumatic isolation valve for DN50 pipe. The valve demonstrated sound operation after vertical and horizontal vibration simulating seismic loads. Moreover, the valve has demonstrated close / open operation in less than three seconds in magnetic field ranging from 290 to 500 mT, and the operation time was not significantly affected by the magnetic field. On the contrary, solenoid valve which provides compressed air to the pneumatic valve halted with magnetic field above 200 mT and failed to close the pneumatic isolation valve. Mockups for palladium membrane diffuser and metal hydrate bed were also manufactured to find show stoppers in application as equipment for tritium extraction sub system in the WCCB-blanket system.

Keywords

Water cooled ceramic breeder blanket. Water cooling system. Tritium extraction system.
Reduced activation ferritin steel.

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PS2-39 ABSTRACT-6217
B. Blanket Technology

Characterization of aged Li4SiO4 pebbles performance

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Many studies and research have been carried out in the recent years to identify which breeder material and (in which) form, solid or liquid, are most suitable to be used in the breeder blanket (BB) modules. Nonetheless, the choice of a stable material, with high thermal conductivity and melting point remains unsolved and becomes even more complex when durability is considered. The breeder material must guarantee unchanged thermo-mechanical properties under environmental and operating conditions.

The aim of this study is to investigate the ageing effects on the mechanical behaviour of lithium orthosilicate (Li_4SiO_4) in form of pebbles. The pebbles had been produced at the University of Pisa at room temperature by using the drip casting forming technique combined with hydrolytic sol-gel method and subsequently stored for 4 years in an environment simulating normal storage (i.e., standard condition).

These pebbles of about 1.5 mm diameter have been subjected to uniaxial compression tests, without radial constraints, and SEM analysis. The results obtained were compared to those of non-aged pebbles in order to highlight the degradation caused by the aging. SEM examination showed increased porosity and a pebble surface that was not entirely homogeneous and uniform. This affected the crack shapes on the contact surface and also the crushing load.

Keywords

Aging, Li4SiO4 pebbles, testing, compression.

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PS2-40 ABSTRACT-623c
B. Blanket Technology

A multi-region MHD OpenFOAM solver for fusion reactor análisis

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Breeding Blanket (BB) and advanced concept of Plasma Facing Components (PFCs) are two systems of a nuclear fusion reactor which involve the use of a Liquid Metal (LM) as the working fluid. Due to the electrical conductivity of LMs, their motion is influenced by the magnetic field, used to confine the plasma, generating a complex phenomenology which is studied by the LM branch of magnetohydrodynamics (MHD). These phenomena include the generation of an electric current within the liquid and of the electromagnetic drag, turbulence suppression, modified heat and mass transport and electromagnetic coupling phenomena that profoundly impact the flow features. In this framework, intense studies and research activities are essential to provide high-quality numerical data and to develop accurate predictive numerical tools. This work presents the OpenFOAM *mhdMultiRegionFoam (mMRF)* solver and its verification and validation for both forced convection (magneto-hydraulic) and natural convection (magneto-convective) test cases. The solver can simulate transient, incompressible, inductionless, single-phase, MHD flow for multiple domains. It supports conjugated heat transfer and fluid/solid electric potential calculations. OpenFOAM is a very versatile open-source C++ suite for Computational Fluid Dynamics (CFD), ideal for generating complex models such as MHD models. The *mMRF* solver is based on the "multi-region" segregated approach to deal with fluid and solid domains, a strategy currently engaged in various codes for the thermal coupling between fluid and structures that solves the fluid and solid domains separately and pairs them with an appropriate boundary condition, making it very efficient for complex geometries, and that has been extended to electromagnetic coupling between domains.

Keywords

Breeding Blanket, Conjugated Electric Potential, Liquid Metal, MHD, OpenFOAM.

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PS2-41 ABSTRACT-64ca
B. Blanket Technology

Tritium release from titanium beryllide after high-dose neutron irradiation

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Titanium beryllide $TiBe_{12}$ is planned to be used as an advanced neutron multiplier in the DEMO breeding blanket due to its advantages over pure beryllium, such as reduced swelling, lower tritium retention, and greater stability in water vapor. Cylindrical samples of $TiBe_{12}$ of Ø3mm×3mm manufactured using the arc-melting method and irradiated in the HFR at temperatures of 652, 744, 866, 972 K to 369, 508, 600, 647 appm tritium, 3638, 4788, 5557, 5947 appm helium, corresponding to 19, 28, 34, 37 dpa, respectively. The post-irradiation examinations (PIE) involved temperature-programmed desorption (TPD) tests that focused on the tritium and helium release behavior after heating at 7 K/min to 1373 K, as well as optical metallography and micro-hardness measurements. Additionally, TPD tests were performed on irregularly shaped $TiBe_{12}$ samples that were loaded with a mixture of ^{2}H and 500 appm ^{3}H gas at 1123 K for 6 hours and heated at rates of 1 K/min and 7 K/min to 1373 K. The tests utilized various finds of $TiBe_{12}$, including coarse and fine grained samples after hot pressing, and samples in a melted state.

The tritium release spectra of irradiated titanium beryllide display a single peak, regardless of the irradiation temperature. After irradiation at 652 K, the peak is observed at 1140 K, and as the irradiation temperature increases, the peak temperature also increases, reaching its maximum possible testing temperature of 1373 K. On the other hand, the tritium release spectra of gas-loaded titanium beryllide show two peaks, with peak temperatures that vary based on the fabrication method and grain size. The melted titanium beryllide spectrum exhibits peaks at 560 K and 1230 K. The TPD results are further analyzed in light of the microstructural evolution of titanium beryllide after neutron irradiation.

Keywords

Titanium beryllide, neutron irradiation, tritium, thermal desorption.

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PS2-42 ABSTRACT-684a
B. Blanket Technology

Optimization of the first wall of WCCB for the CFETR

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The optimization of the first wall of the CFETR water-cooled blanket and the studies on the breeder pebble bed performance are carried out, according to the design parameters of the CFETR project. Firstly, the turbulent flow and heat transfer process in the complex structure are numerically solved by the Reynolds-averaged RANS method, and the first wall is optimized with the consideration of different amounts of surface heat flows. In order to reduce the thermal stress in the first wall, a copper transition layer is arranged between the tungsten armor and the first wall, and 16 stress relief grooves are evenly set in the tungsten armor and the copper transition layer. FeCrAl alloy is coated on the inner surfaces of the first wall cooling channel, which effectively reduces the penetration of tritium. Based on the three-dimensional smooth square tube, the square tube with straight rib and the square tube with V-shaped rib, the thermal wall structure of the channel has been optimized to improve the heat transfer efficiency. Secondly, using the turbulent heat transfer theory, the flow field analysis and thermal hydraulic analysis of the channel are carried out. The heat-line and field synergistic analysis methods are used in the first time to conduct an in-depth analysis of the enhanced heat transfer mechanism of the blanket first wall. From the perspectives of entropy production and entransy dissipation, the results of enhanced heat transfer by adding fins are analyzed. The methods and principles of enhanced heat transfer are further supplemented to optimize further the heat transfer of the first wall channel of the water-cooled blanket.

Keywords

CFETR WCCB First Wall entropy production entransy dissipation.

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PS2-44 ABSTRACT-6af3
B. Blanket Technology

Characteristics of Pb-16Li droplets generated by a perforated plate

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Tritium self-sufficiency is a key aspect for future fusion reactors such as EU DEMO which operation is based on the D-T reaction. Tritium is produced via nuclear reaction of neutrons with lithium, which however generates almost equal amount of helium as a by-product. In case of breeding blankets using liquid metal Pb-16Li, closed circulation of the liquid breeder requires efficient separation of the two gases from the main flow of the liquid metal. Desorption of the gases, tritium and helium, from Pb-16Li into vacuum in a sieve tray column is one of the preferred candidate technologies. Main advantage of this technology is its robustness as the interfacial area consist of the liquid droplets surface. To further enhance reliability of the technology, it is suggested to use a simple perforated plate

For studying this process, a dedicated experimental facility VOSA was built and commissioned in Research Centre Rez (CVR). This contribution reports on optical determination of characteristics of Pb-16Li droplets generates by a perforated plate during an initial series of experiments with Pb-16Li carried out at the VOSA facility.

Keywords

DEMO, breeding blanket, Pb-16Li, tritium.

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PS2-45 ABSTRACT-6c08
B. Blanket Technology

Thermofluid-Dynamic and Thermal-Structural Assessment of the EU-DEMO WCLL "Double Bundle" Breeding Blanket Concept Left Outboard segment

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As part of the activities performed by the DEMO Central Team (DCT) to study alternative configurations of the Water-Cooled Lead Lithium (WCLL) Breeding Blanket (BB), to overcome the open issues that emerged at the end of the Pre-Conceptual Design phase, a new concept, the WCLL BB "double bundle" (db), was developed. This concept adopts an array of db tubes poloidally distributed inside the breeding zone, mimicking the arrangement inside heat exchangers. These are coaxial pipes the gap between which is filled with PbBi (or alternatively with gas and fins), which avoids direct contact between PbLi and water in case of in-box LOCA events. Similarly, the First Wall channel cooling water is separated from the PbLi. The use of PbBi as an inert fluid makes the Segment Box (SB) sizing less stringent: in the case of an in-box LOCA due to a break of the water cooling circuit inside the BB SB, the pressurized volume is only the PbBi one, instead of the whole SB volume as in the reference WCLL design configuration. Consequently, the total amount of steel inside the BB can be reduced, with benefits in terms of increased tritium breeding ratio.

Within this framework, University of Palermo, in support of the DCT and under the EUROfusion action, preliminarily analysed the steady-state thermal, thermo-hydraulic and normal and off-normal thermo-mechanical performances of this concept, following a theoretical-numerical approach based on finite volume and finite element methods, and adopting the ANSYS CFX and ANSYS Mechanical codes. The analyses carried out allowed a preliminary optimisation of the design, improving the thermal and thermo-mechanical performances of the WCLL-db BB, in compliance with the design requirements, while reducing the total steel amount.

Models, assumptions and main results are herewith reported and critically discussed.

Keywords

DEMO, Breeding Blanket, WCLL, Thermofluid-dynamics, CFD analysis, FEM analysis.

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PS2-46 ABSTRACT-6c4e
B. Blanket Technology

Adaptation of the first wall cooling circuits to achieve an efficient distribution of the coolant flow in the SMS DCLL breeding blanket

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CIEMAT

The current design proposal of the Dual Coolant Lithium Lead (DCLL) breeding blanket for the EU DEMO reactor is meant to operate up to 700 ºC to obtain high efficiencies in the thermodynamic cycle. This design follows a single module segment (SMS) architecture, in which self-cooled PbLi flows through an electrically insulating ceramic box to minimize the magneto hydrodynamic (MHD) effects. The ceramic structure is embedded in a steel case, which includes the first wall (FW).
The FW is subjected to a very irregular distribution of heat fluxes due to charged particles and radiation from the plasma. It is cooled by a succession of helium channels arranged along the poloidal length. A conservative oversizing of the coolant mass flow rate can severely jeopardize the blanket thermal efficiency and the consumption of the auxiliary systems, since He extracts around 30% of the blanket thermal power.

According to the cooling demands, the code TOMFLOW (Tool for Optimization of helium Mass FLOW rate distribution) has been developed to estimate the optimal mass flow rate in each channel which ensures the steel structure is kept under 550ºC, assuming they are individually fed. In this work, the hydraulic network composed by the cooling channels and the feeding/collecting manifolds is analyzed by means of new capabilities of TOMFLOW, supported by Finite Element Analysis (FEA) and Computational Fluid Dynamics (CFD) with ANSYS. These models are used to develop a method to feed, from a common manifold, the optimum mass flow rate in each channel. The adequate alteration of the flow rate distribution through modifications of the hydraulic resistance of each branch of the circuit is achieved by varying the manifold/channels cross-sections and wall roughness or introducing heat transfer enhancement structures. The used methodology looks for implementing the least intrusive approach.

Keywords

DCLL, First Wall, Breading Blanket, hydraulic network, helium.

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PS2-47 ABSTRACT-078e
C. Fuel Cycle and Tritium Processing

Tritium Management in ITER Test Blanket Systems Port Cell for Maintenance Operations

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The ITER TBM Program aims to provide the first experimental data on the performance of the breeding blankets in the integrated fusion nuclear environment. Four TBSs will simultaneously operate in ITER equatorial port #16 and #18 to verify the tritium cycle and heat extraction. A significant quantity of tritium would be produced and then release into the PC from TBS pipework because of the strong permeation at high temperatures.

This work built a multi-physics/dimension model to characterize the HT/HTO permeation, absorption/desorption, and diffusion in PC# 16. The effect of paint thickness, ventilation rate, and release source term during STM/LTM were discussed. The 3D flow field presents several dead zones in the corner or near the wall surface, leading to a low detritiation efficiency. A higher ventilation rate could accelerate the detritiation process, while minimizing a radioactive source is the most direct method. So, a high TPRF coating is indispensable to prevent the tube from leakage. The HTO desorption from paint and concrete is the dominate source term in detritiation process, which act like a delayed source in PC and significantly prolong decontamination time. A low tritium solubility paint is strongly suggested because low HTO retention in paint could eliminate this delay effect.

Keywords

Tritium management, Multi-physics model, TBM port cell, Tritium safety.

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PS2-49 ABSTRACT-d88d

C. Fuel Cycle and Tritium Processing

Increasing the accuracy of the Viscosity Measurement Apparatus (ViMA) for measuring the viscosity of tritium between 77K and 300K

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The viscosity of the radioactive tritiated hydrogen isotopologues (HT, DT; T_2) is of importance for many fluid simulations needed in fusion science and particle physics. Values to be found in literature are ab initio values, calculated through a classical approach and are only valid for temperatures above 300K. For lower temperatures, only extrapolations from hydrogen and deuterium exist with an estimated uncertainty of worse than 5%. The challenge of measuring the tritium viscosity at 80K is, that all wetted materials have to be compatible with tritium and cryogenic temperatures. Moreover, everything has to be measured with hydrogen and deuterium, to understand the systems behavior and the accuracy of the measurement, before mounting the setup inside a glovebox. We developed a setup, based on a spinning rotor gauge (SRG), with which we will measure the viscosity of tritium, between 77K and 300K. In previous publications, we already showed, that these measurements work at room temperature and at liquid nitrogen temperature, which had not been known, since the SRG is only specified down to 0°C and apart from electronics problems at low temperature, effects like freezing out of impurities and condensation could affect the measurement procedure and accuracy. First measurements show an uncertainty of less than 2% and therefore demonstrate that it will be possible to improve the knowledge of tritium viscosity. We already identified that the 2% are mainly caused by systematics of how a measurement is conducted and temperature effects, caused by the spinning rotor gauge itself. With this contribution we describe, how the uncertainty on the measurement can be reduced by regarding temperature gradients inside the SRG. With simulations, conducted with ANSYS, we were able to understand the heating of the rotor and correct the measured values for the residual gas temperature, improving the accuracy to nearly 1%.

Keywords

Tritium viscosity, Tritium Laboratory Karlsruhe, spinning rotor gauge, cryogenic tritium compatible setup, ANSYS simulation.

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PS2-50 ABSTRACT-758e
C. Fuel Cycle and Tritium Processing

Process Design of Hydrogen Isotope Separation using Temperature Swing Absorption in the EU-DEMO Fuel Cycle

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The EU-DEMO plant foresees a closed fuel cycle based on direct internal recycling. Instead of separating the circulated fuel completely, the isotopic composition is adjusted by the Isotope Rebalancing and Protium Removal (IRPR) unit, which counteracts the protium build-up and performs only the necessary isotopic separation. The system is based on a membrane-coupled temperature swing absorption (MC-TSA) process which is being developed at KIT. The process exploits isotope-dependent gaseous diffusion (MC stage) and metal-hydrogen-interactions (TSA stage) to achieve efficient separation of hydrogen isotopologues. Since the separation of hydrogen isotopes involves high tritium inventories and long processing times, a rigorous optimization of the system is required.

The process design is supported by experimental investigations of suitable absorption materials which are used in the TSA stage. The observed material characteristics such as isotope-dependent absorption kinetics and equilibrium pressure are implemented in a transient process simulation tool and are extrapolated for tritium. Furthermore, rigorous heat and mass transfer computations are performed to mimic TSA process conditions occurring during operation. The tool can be applied to inform experimental campaigns using the MC-TSA test facility for demonstration and scale-up of the process as well as for a wide range of parameter studies to identify and optimize the main design drivers. The separation experiments, in turn, allow the validation of the developed process model.

This poster presents the current status of the material characterization and the resulting approaches for the TSA process simulation. An outlook for following separation experiments is given.

Keywords

Hydrogen Isotope Separation, Metal-Hydrogen-Interactions, Hydrogen Absorption, EU-DEMO.

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PS2-52 ABSTRACT-36e3
C. Fuel Cycle and Tritium Processing

Plasma Exhaust Processing System with Tritium Compatible Reciprocating Pump

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Fusion reactors require continuous evacuation of plasma exhaust while returning DT fuel to plasma. Tritium compatible dry vacuum and transfer pumps was developed for backing and roughing for torus evacuation, fuel cycle gas transfer and processing. Together with Metal Foil Pump or Proton Conductor Pump, simplified direct internal fuel recycling driven by a roughing pump can provide closed tritium fuel loop. Tritium compatible dry pump can also serve the facility vacuum for processing and safety systems. The first generation of dry reciprocating pumps were developed by the Tritium Process Laboratory and Mikuni Heavy Industry in Japan in 1990s and was successfully tested with tritium. Based on this 4 stage coaxial compression technology, new prototype for fusion fuel cycle was developed by Kyoto Fusioneering and Mikuni. All parts in contact with pumped gas are composed of tritium compatible materials. It is known that dry pumps often exhibit poorer compression and pumping speed for light gases, such as hydrogen isotope mixtures. One of the advantage of this pump is its less sensitivity of pumping characteristics for the species of gas. Results of pumping of various gas mixtures anticipated for fusion plasma operation from 1 Pa range will be reported. Combination of this reciprocating pump and turbomolecular pump, or metal diffusion pumps that will be an integrated pump system for plasma chamber evacuation will also be reported. This pump train shows a simplified inner fuel cycle of the fusion plasma that recycles purified DT gas back to the fueling with minimal processing inventory. The results are scalable to the 200Pam3/s throughput range, and suggest feasibility of the dry tritium compatible pumps for the continuous fusion facilities in the near future and DEMO .

Keywords

Tritium, pump, tritium compatible, vaccum, reciprocating.

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PS2-53 ABSTRACT-3b4c
C. Fuel Cycle and Tritium Processing

A magnetic compression platform for compact torus injector (central fuelling system for EAST tokamak)

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It is the key to increase the tritium burnup fraction and realize tritium self-sufficiency by fuelling the tritium particles directly into the fusion reactor core and increasing the particle confinement time. Understanding the penetration mechanism of fuelling particles in a magnetized plasma provides theoretical support for the realization of tokamak central fuelling. Based on the EAST-CTI (Experimental Advanced Superconducting Tokamak - Compact Torus Injector) system, we designed and built a magnetic compression experiment platform. The background magnetic field generated by a cryogenic superconducting coil with 2,000 turns of densely wound wire with a diameter of 50 cm, which has the capacity to generate a maximum 1.5 T magnetic field (280 mm in diameter and a stable section of approximately 30 cm in length). A longitudinal array of 2D magnetic probes is arranged in the magnetic compression platform to investigate the spatio-temporal evolution of CT during the penetration process. The penetration mechanism of the magnetized plasma is studied by injecting a compact torus into the compression platform. When the injection velocity is slow, CT cannot penetrate the background magnetic field. The amplitude and full width at the half maximum of the magnetic signal decrease significantly after penetrating the gradient magnetic field. This indicates that the CT axial compression may occur in the gradient magnetic field, and the degree of compression between CT and the background magnetic field also increases. The gradient magnetic field will reduce the CT tailing effect.

Keywords

Compact torus, Superconducting coil, Gradient magnetic field, Magnetic compression.

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PS2-54 ABSTRACT-4884
C. Fuel Cycle and Tritium Processing

Experimental results to evaluate the use of cryopumps to separate helium from hydrogen isotopes in a tokamak exhaust stream

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The adsorption of mixtures of hydrogen isotopes (Q2) and helium on an activated charcoal panel at temperatures between 10 and 20 K have been explored as a method to separate them and reduce the power consumption required of tokamak cryopumps. The aim of this work is to ascertain the suitability of activated charcoal-coated cryopanels for continuously separating helium from hydrogen isotopes at temperatures between 10 and 20 K. The results support the case of the cryopump design trapping Q2 species while continuously exhausting helium.

The trapping of Q2 and helium mixtures have been studied and successfully achieved using temperatures down to around 4 K. However, this approach does not consider the separation of Q2 from helium. A "sharp" and continuous separation process, based on existing cryopump technologies, would be advantageous for tokamak operations. A combination of experiments and Direct Monte Carlo simulations (DSMC) have been employed to elucidate the separation performance of Q2 and helium mixtures. The experiments were devised using a factorial design of experiments; while the DSMC was performed using a parallel code, PI-DSMC. All indications support that a cryopump design with cryopanels at temperatures between 10 and 20 K, without cryopanels to capture helium, is feasible albeit creating the requirement of helium having to be continuously pumped out.

Keywords

Cryopump, DIR, Recycle loop, tritium, exhaust pumping.

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PS2-55 ABSTRACT-4a21
C. Fuel Cycle and Tritium Processing

Analysis of super-permeable membrane surfaces for tritium separation applications

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Super-permeable membranes are leading candidates for separating tritium from other species in tokamak plasma exhaust. It has been established that the performance of these membranes is strongly sensitive to oxides and other impurities on the front and downstream surfaces. The present study includes a series of experiments to study hydrogen interactions with both Nb and V surfaces with varying coverage of O. We performed x-ray photoelectron spectroscopy (XPS) and low energy ion scattering (LEIS) measurements on Nb and V surfaces while dosing the surface with both atomic and molecular D. The oxide on the as-prepared Nb samples appeared to be primarily a Nb_2O_5 phase, with only a small fraction of the surface corresponding to metallic Nb. As the temperature increases, the oxide begins to decompose. At 300 °C, the NbO phase begins to dominate, and peaks associated with the pure metal begin to emerge. Once the temperature has reached 460 °C, the surface is enriched with metallic Nb, at approximately 50% coverage. Similar behavior was observed for V surfaces. In both cases, XPS peaks associated with O and C also diminished with temperature, and electron beam heating to temperatures > 1200 °C led to complete decomposition of the oxides. To study the oxide response in a more realistic environment, we applied in-situ spectroscopic ellipsometry during exposure to high-flux (1.5×10^{21} m^{-2} s^{-1}) D_2^+ plasmas. The ellipsometer was mounted to an RF plasma source with line-of-sight viewing of the sample, allowing real-time oxide thickness measurements with 0.5 nm resolution. Whereas low energy (75 eV) Ar^+ plasmas completely removed the oxide in < 5 min., much higher incident ion energies (400 eV) were required to achieve the same result with D_2^+ exposure. These studies enable us to bound the operating window for super-permeable membranes for fusion applications.

Keywords

Super permeation, tritium extraction.

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PS2-56 ABSTRACT-50cc
C. Fuel Cycle and Tritium Processing

Highly efficient and direct recovery of low-pressure hydrogen isotopes from tritium extraction gas by Pd-Y alloy membrane permeator

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Nuclear fusion power is one of the most promising solutions for the energy crisis. Tritium and deuterium are the fuel of fusion reactors. Due to very limited sources of tritium, commercialized fusion reactors need tritium breeding for self-sufficiency, where tritium is produced from lithium-based ceramics of the breeding blanket. After the breeder blanket is purged by helium doped with about 0.1mol% hydrogen from the tritium extraction system (TES), the tritium produced is transferred to the extraction gas. For the recovery of tritium in extraction gas, the early design process is mainly based on cryogenic molecular sieves or getter bed. These methods must operate in batch mode which prevent breeding tritium from participating in fusion combustion in time. The Pd membrane separation can operate continuously, and such a characteristic reduces the tritium inventory. Incelli et al. have studied numerically using a pre-concentration stage to enrich hydrogen isotopes before entering Pd-Ag permeator for such low concentration gas. However, introducing a pre-concentration stage would bring complexities.

Palladium membrane permeation technology has been widely used in hydrogen purification. However, there are few studies on the recovery of low-pressure hydrogen. In our previous work, recovery more than 99.9% have been achieved for 2 mol % D_2 in He by Pd-Ag membrane diffuser, and effects of diffuser structure on the efficiency for recovery of 1 mol % H_2 in He have been numerical simulated. Compared with the commonly used Pd-Ag, Pd-Y has higher hydrogen permeation rate and stronger mechanical properties. In this work, a Pd-Y alloy membrane permeator has been constructed. Its performance of purifying low-pressure hydrogen isotopes from helium mixture was tested. Through process optimization, the permeator with 678cm^2 effective area could process 21 Standard Liter per Minute (SLM) 0.1 mol % H_2 in He achieving around 95% recovery, up to 30.97 ml/(min·cm^2) handling capacity.

Keywords

Tritium breeding, Tritium extraction system, Palladium membrane permeator, Pd-Y alloy.

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PS2-57 ABSTRACT-5117
C. Fuel Cycle and Tritium Processing

Preliminary evaluation of NEG materials in view of CMSB replacement in HCPB TER architecture

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CMSB (Cryogenic Molecular Sieve Beds) operated at 77 K is used for tritium recovery in the reference design of HCPB TER system. However, this technology has two main drawbacks: large consumption of liquid nitrogen and the controlling of tritium stream sent to the Tritium Plant, during the regeneration of the CMSB, in such way to avoid large variations in the gas flow rate.

This paper reports on a preliminary evaluation for CMSB replacement by non-evaporable getter materials for tritium recovery. The advantages of NEG technology are the high specific pumping speed and capacity and the ability to withstand a large number of absorption/desorption cycles. In addition, NEG do not release hydrogen unless power is supplied to heat them, addressing an important safety issue related to uncontrolled release in case of power outage or subsystem failures.

Keywords

NEG technology, Tritium.

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PS2-58 ABSTRACT-5577
C. Fuel Cycle and Tritium Processing

Development of lab-scale Atmospheric Molecular Sieve Bed and generation of experimental break through curves for adsorption studies

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The effective design of Tritium Extraction System (TES), which involves proper extraction and purification of tritium in the fuel cycle of the fusion reactor, is one of the most challenging tasks. Tritium extracted from the breeder zones of Blanket is loaded with impurities like, O_2, N_2 along with tritium in combined form Q_2 and Q_2O (Q = H, D or T). It is necessary to remove all the impurities from tritium and recover tritium from its isotopes before it is refueled to the reactor. Atmospheric Molecular Sieve Bed (AMSB) and Cryogenic Molecular Sieve Bed (CMSB) are two main systems of TES. AMSB removes ppm levels of moisture (Q_2O), while CMSB removes hydrogen isotopes (Q_2) along with oxygen and nitrogen gas from Helium purge gas. Though detailed parametric Study of AMSB is very important for fuel cycle technology, such studies are very rare.

In this work, a lab-scale AMSB is designed and developed at Institute for Plasma Research, Gandhinagar, using Zeolites 4A as an adsorbent material. Experimental break through curves for the adsorption of various concentrations of moisture (H_2O) (up to ~20,000 ppm) at flow rates (1-3 LPM) from helium gas at room temperature have been generated. The moisture concentration is reduced to ~ 5 ppm at the outlet of the AMSB. Break through and saturation times of the AMSB are measured for different operating conditions. Adsorption capacity of lab scale AMSB is estimated to be ~19 % by weight. The effects of the initial moisture concentrations, pressure and flow rates on breakthrough curves are studied in detail. Simulations of breakthrough curves are performed through a numerical model. The importance and proper way of regeneration for optimizing the adsorption capacity of the bed is discussed. This paper also describes the detailed design of the experimental set up for AMSB.

Keywords

Adsorption, Breakthrough curves, Extraction, Removal, AMSB, Ppm.

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PS2-60 ABSTRACT-58d1
C. Fuel Cycle and Tritium Processing

The influence of the extraction of gas/liquid samples on the isotopic separation regime of the CECE (Combined Electrolysis Catalytic Exchange) process

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At ICSI Rm.Valcea it is a concern of the research group within the Nuclear Department for the development of the CECE isotopic separation process.

At the laboratory level, an experimental installation was created and put into operation for the development of the CECE process, both for the separation of tritium in view of waste decontamination and its recovery, and for the separation of deuterium for its recovery in various technological applications.

For the characterization of the CECE isotopic separation process, a calculation model was developed which was implemented into a calculation program.

For the analysis of the separation process, it is necessary to carry out sample extractions on the liquid/gas phase in the key points of the technological installation.

Since the isotopic separation process is a mainly non-stationary process, we considered it’s necessary to carry out a theoretical analysis regarding the influence of sample extraction frequency on the isotopic separation regime.

The paper presents a theoretical analysis carried out in order to optimize the extraction of gas/liquid samples necessary for monitoring the isotopic separation process.

Keywords

Tritium separation, PEM electrolyzer, combined electrolysis catalytic Exchange.

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PS2-61 ABSTRACT-5a68
C. Fuel Cycle and Tritium Processing

Influence of chemical form of tritium extracted from solid breeder on tritium balance in fuel cycle system of fusion

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From the viewpoint of the feasibility and safety of the fusion reactor power generation system, tritium balance evaluation in the fuel cycle system is an important issue. A water-cooled solid breeder blanket is assumed for the JA-DEMO reactor and the fuel cycle system consist of a lot of subsystems such as Tokamak exhaust processing system (TEP), Isotope separation system, bred tritium recovery system, and water detritiation system and so on. Through past research activities, it is predicted that the tritium produced in the solid breeder will be extracted by gas purging as molecular tritium (HT and T_2) and tritiated water vapor (HTO and T_2O). However, the influence of chemical form of tritium released from solid breeder on the design of a fuel cycle system is not understood significantly. In this study, the influence of the chemical form of tritium extracted from solid breeder was evaluated by applying the mean residence time method to tritium balance evaluation in the fuel cycle system.

It was assumed that molecular tritium is extracted by Pd diffuser in bred tritium recovery system and tritiated water vapor is transported to the TEP. The mean residence time in each subsystem was decided from the related literatures. In the blanket, since the tritium permeation rate from molecular tritium to cooling water through the cooling pipes is much faster than that from tritiated water vapor, tritium concentration in the cooling water increases with increasing the HT/HTO ratio in the breeding blanket. This leads to increase the safety risk of power generation system and the load of WDS but decreases the load of TEP for tritium extraction from tritiated water vapor. From the viewpoint of reducing the total inventory of TEP and WDS, it was better to increase the HT/HTO ratio under assumed conditions.

Keywords

Tritium balance, Fuel cycle, chemical form, blanket.

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PS2-62 ABSTRACT-a9a0
D. Material Engineering for FNT

An effectiveness of deuterium electrochemical charging method for a desorption study in low-alloy steel and pure Zr

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The purpose of this paper is to establish the effect of conditions of electrochemical deuterium (D) charging under galvanostatic conditions of low-alloy steel and pure Zr and to propose the optimum parameters of charging process for simple and reliable thermal desorption spectrometry study (TDS) of materials relevant to nuclear industry.

In spite of numerous investigations into the retention and diffusion of Hydrogen (H) in fusion and fission related materials, there is a large scatter in the experimental data and little agreement on the activation energy of H de-trapping. The reasons are mainly related to the complexity to execute the experiment and a lack of a clear distinction between the naturally occurring H in the materials and intentionally introduced for the experiment, thus often complementary or contradictory mechanisms of H retention and desorption have been proposed.

This work aims to determine a reliable electrochemical route to introduce D into metals, to assess the quantity of deuterium introduced into materials and to define the parameters detrimental for controllable D adsorption and desorption at galvanostatic conditions. The electrolytes were prepared from 99.99% D_2O (heavy water), and includes H_2SO_4 or $HAsNa_2O_4$. The heavy water was chosen to clearly distinguish between naturally occurred hydrogen and its isotope D introduced to the material. The charging process was carried out at current densities from 1 to 200 mA/cm^2 for various time. Complementary analysis of D depth distribution in Zr was performed by ToF-SIMS. The overall conclusion is that the current densities as low as $5 mA/cm^2$ and time approximately of 30 minutes are sufficient to introduce D into analysed materials to saturate structural trapping sites, allowing repeatability of the experiments, as well as reducing time and cost of the tests.

Keywords

Hydrogen retention,Duterium,Zr,Ferritic steel,TDS.

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PS2-64 ABSTRACT-183c
D. Material Engineering for FNT

Quantification of the uncertainties in the determination of the final dose of the HFTM samples in IFMIF-DONES

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The behaviour of the vast majority of materials is unknown under the neutron irradiation conditions expected in future nuclear fusion power reactors. The International Fusion Irradiation Facility-DEMO Oriented NEutron Source (IFMIF-DONES) aims to test materials under equivalent nuclear fusion irradiation conditions. This neutron source will be produced by a 40 MeV and 125 mA deuteron beam impinging on a thick lithium jet to produce a striping reaction D^+ + $^{6,7}Li$.

The quantification of the uncertainties in the determination of the final dose in the High Flux Test Module (HFTM) material samples is essential to analyze the damage in the specimen material. Dose in the HFTM samples will be tallied by the Self Powered Neutron Detectors (SPNDs) diagnostics located in the centre of each HFTM rig. In this work, the response of SPNDs to the IFMIF-DONES neutron spectrum has been determined, and the fluctuation of this response to variations in the neutron spectrum. Furthermore, the influence on the signal of different SPNDs due to fluctuations in beam dynamics or different distributions of the samples in the HFTM has been analyzed to quantify the uncertainties of the SPNDs detectors.

The neutron transport calculations have been performed using McDeLicious 2017 (based on MCNP6.2), developed by the KIT research institute to reproduce the IFMIF deuteron–lithium neutron source. Moreover, a computational model has been developed to determine the signal of the different kinds of SPNDs.

Keywords

IFMIF-DONES, HFTM, dose, uncertainties, simples.

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PS2-65 ABSTRACT-c0b9
D. Material Engineering for FNT

Tritium permeation through Inconel 600 under high temperature, high pressure water environment: Influence of oxidation of coexisting materials

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Strict confinement of tritium (T) is necessary to realize fusion power. In a water-cooled blanket system, T permeation in a steam generator from the primary to the secondary coolant loop is an important safety issue. The purpose of this study is to examine the T permeation through Inconel 600, a candidate material of steam generator tubes, under a high-temperature, high-pressure water atmosphere.

In addition to Inconel, metals such as reduced activation ferritic steel and stainless steel (SS) are expected to be used in the primary loop. To simulate the influence of coexisting materials, three types of high pressure containers were used: (1) SS container with fresh surfaces, (2) SS container with surfaces covered by protective oxide layers formed by prior oxidation in water, and (3) Hastelloy container. The container was divided into upstream and downstream chambers by an Inconel disk. The upstream and downstream chambers were filled with T water of 0.9 MBq/cm^3 and T-free water, respectively. After heating the container at 280 ℃ for 14-70 hours, the amount of permeated T in the downstream was measured using a liquid scintillation counter. The effect of O_2 gas addition was also examined.

The permeation rate through the Inconel 600 disk with SS container with fresh surface was initially 1 Bq/m^2s and gradually increased to 3 Bq/m^2s. The SS container with oxidized surface gave an order of magnitude smaller permeation rate, and further smaller permeation rate was observed with the Hastelloy container. The oxidation of container material by HTO generates HT. The results obtained indicate that the permeation rate can be sensitively dependent on the rates of oxidation of coexisting material and consequent generation of HT molecules. The addition of O_2 gas resulted in drop of permeation rates for all container materials probably due to oxidation of HT to HTO.

Keywords

Tritium, permeation, high pressure water, oxidation.

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PS2-66 ABSTRACT-d586
D. Material Engineering for FNT

Predicting the ductile-to-brittle transition (DBT) in BCC fusion materials

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Understanding the micromechanics of ferritic steels under irradiation can help to more accurately predict the ductile-to-brittle transition (DBT) in BCC fusion materials. Traditionally, the DBT has been obtained via impact testing (or other methods), which often require large quantities of material. Material in these quantities is difficult to obtain in the irradiated condition. This work involves developing a new methodology for studying the micromechanics of irradiated fusion steels, using small samples alongside in-situ testing. The goal of this development is to be able to understand and predict the expected behaviour of BCC fusion metals around the DBT.

High Resolution Digital Image Correlation (HRDIC) has extensive use for studying deformation behaviour in many alloy systems. The fundamental behaviour of a BCC iron matrix was investigated in this study, using interstitial free (IF) steel. HRDIC can provide quantitative information at the microstructural level about the strain distribution, which when correlated with EBSD allows the strain behaviour to be linked to the crystallography. The effect of temperature and strain level in for both irradiated and unirradiated material can be investigated. In future, the same methodology will be applied to more complex reduced activation ferritic martensitic (RAFM) fusion steels, to give insight into the expected deformation behaviour under fusion conditions.

This work will present the challenges involved with using HRDIC to study the deformation effects at temperatures down to cryogenic conditions, alongside initial results from these experiments. The technique shows great promise for developing greater understanding of the ductile-to-brittle transition in BCC fusion materials, and for improving our ability to predict it in different alloy systems.

Keywords

Ductile-to-Brittle Transition, HRDIC.

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PS2-67 ABSTRACT-377f
D. Material Engineering for FNT

Effect of Hydrogen on stress corrosion cracking behavior of Oxide Dispersion Strengthened Steel Compared with RAFMs

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Reduced Activation Ferritic/Martensitic （RAFM）and ODS alloys are primary candidates for application as structural materials in the fusion blanket structural materials. This work analyzes the effect of hydrogen on stress corrosion cracking behavior of ODS(9Cr-ODS/9Cr-Al-ODS) steel compared with RAFMs (CLF/TMT). Specimens were pre-hydrogen-charged, corrosion tests were performed in both RAFM and ODS steels at 300 under 15.5Mpa, the oxide microstructure formed on these alloys was analyzed, test results were observed that dissolved hydrogen (DH) promoted the SCC behavior of RAFMs and ODS Steels, meanwhile, the pre-hydrogen-charged specimens exhibited a higher sensibility than that of the uncharged specimen, the charged hydrogen can be trapped in the metal. ODS steel exhibited a different oxide structural morphology, the most pronounced difference was the development of a substantially thick internal oxidation layer for the ODS steel that is associated with its finer grain size. Furthermore, the addition of 4%Al is effective to improve the corrosion resistance of 9Cr-ODS steel, which was that an adequate combination between Cr and Al content ranges.

Keywords

SCC behavior; Hydrogen; RAFM; ODS.

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PS2-68 ABSTRACT-05ad
D. Material Engineering for FNT

High-throughput evaluation of primary radiation damage on combinatorial thick films

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Material selection is a challenge for plasma-facing components (PFCs) in fusion reactors, and when these components have additional functional requirements, as is the case of radio-frequency (RF) antennas requiring retained electrical and thermal conductivity, this becomes more challenging. Determining the appropriate material with traditional workflows is slow and requires many tests before and after irradiation to characterize a small number of different alloys. A better workflow would enable monitoring of property evolution straightaway after successive radiation cascades on multiple compositions at once.

In this project, we develop a high-throughput approach employing physical vapor deposition (PVD) to combinatorially create thick films, varying the composition in 2D between three different elements that comprise the alloys. We characterize the resulting thick films (3-4 μm) by rapid, non-destructive techniques, like energy dispersive spectroscopy (EDS), four-probe electrical resistivity, and transient grating spectroscopy (TGS), to establish a baseline of chemical composition, electrical, and thermo-elastic properties, respectively. Then, we ion irradiate the films and remeasure them to observe relative changes in properties after primary radiation damage. We repeat this step several times until we reach the final dose for an RF antenna. This process allows the evaluation of hundreds of alloy compositions simultaneously, tremendously increasing the workflow speed concerning conventional routes while keeping the experimental reliability of the radiation damage data. We infer primary radiation damage resistance by relative changes in thermal conductivity obtained by TGS measurements, enabling a correlation between regions with better radiation damage resistance and their composition. The decay of the TGS signal after irradiation is directly related to the presence of more vacancies in the microstructure. Other properties of interest for an RF antenna, like electrical resistivity, are also evaluated with increasing doses. We report results from the Cu-Cr-Nb ternary system, along with implications for developing Nb-free Cu alloys.

Keywords

Primary radiation damage, Combinatorial thick film, RF antenna, Transient grating spectroscopy, Ion irradiation, PVD, Cu-Cr-Nb.

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PS2-70 ABSTRACT-44fc
D. Material Engineering for FNT

scCVD diamond based µ-loss monitors for neutron detection at low temperatura

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This presentation will summarize activities in the United States over the past year to build a consensus community perspective on the requirements for a fusion prototypic neutron source. These activities involved both a 1/2 day webinar, in addition to a 2-day in-person workshop between the fusion materials and private fusion industry that led to a consensus requirement and timeline. Following the presentation of these performance requirements, including the available volume, neutron spectrum and operational timeline, the remainder of the presentation will focus on the state-of-the-art in the use of computational materials science modeling to optimally utilize the limited irradiation volumes within a fusion prototypic neutron source. In particular, the presentation will describe the challenges and opportunities associated with modeling the performance of structural and blanket materials, and divertor PFCs in a next-step fusion materials environment, and provide examples of recent progress to investigate the dramatic surface evolution of tungsten exposed to low-energy He and H plasmas, as well as the coupled He-defect evolution in bulk structural materials exposed to fusion environments.

Keywords

Micro-loss monitors, CVD diamond, cryogenic temperaturas.

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PS2-71 ABSTRACT-4502

D. Material Engineering for FNT

Deuterium permeation of low-activation vanadium alloys possible for reuse in a short time in fusion reactors

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Low-activation vanadium (V) alloys are alternative to reduced-activation ferritic/martensitic steels for the blanket structural material due to their high-temperature strength, superior irradiation properties and non-magnetic characteristics. To make the reusage of the alloys possible in a short time, new-design concepts of V alloy have been suggested in National Institute for Fusion Science.

Fuel hydrogen isotope permeation for the structural material is a key concern in points of view of reactor safety and feasibility. In the present study, the deuterium permeation properties of the new-designed V alloys were investigated.

New-designed V-4Cr-4Ti alloy samples with high and low purity, named H44 and L44, respectively, were used. For comparison, the NIFS-HEAT-2 V alloy samples, named NH2 were also used. Before the permeation experiments, the samples were electropolished and degassed in a vacuum at 1273 K. The permeation experiments were carried out with the sample temperatures of 673, 773, 873 and 973 K. To elucidate effects of surface/bulk conditions of the sample on the permeation, the permeation data were obtained when the sample temperature were raised from 673 K to 973 K and lowered from 973 K to 673 K.

It was found from the permeation experiments that deuterium permeability coefficient of L44 was smaller than that of NH2, while the permeability coefficient of H44 was comparable with NH2. Estimated activation energies of diffusion and permeation indicated that the permeation behaviors significantly depended on titanium oxides at the surface and in the bulk. Data obtained by a microscope with energy dispersive X-ray spectroscopy indicated that the concentration and composition of the oxide must change with air exposure during the maintenance and with hydrogen isotope exposure during the reactor operation. The obtained results suggest that the new-design V alloy with low titanium content would show stable permeation characteristics for hydrogen isotopes.

Keywords

Vanadium alloy, Hydrogen isotope permeation.

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PS2-72 ABSTRACT-5bc3
D. Material Engineering for FNT

Research Progress of High Strength Jacket Based on Super-Austenitic Stainless Steel for Future Fusion Magnets

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Increasing magnetic field intensity in limited space is an important strategy to obtain high parameter plasma and improve the fusion power. The next generation of fusion magnets will have a peak magnetic field greater than 15T, such as China Fusion Engineering Test Reactor (CFETR) central solenoid magnet. The development of high strength and high toughness jacket has become one of the challenges in the application of high-field cable-in-conduit conductors (CICC) for CFETR. 0.2% proof stress of jacket should be over 1500 MPa at 4.2 K for CICC, thus 316LN and JK2LB jackets developed by ITER do not meet the requirements. Nitronic 50 (N50) super-austenitic stainless steel material has great optimization potential for the development of jacket. ASIPP has developed the the modified N50 material together with China Iron and Steel Research Institute, Metal Research Institute, etc, and the CICC jacket was made some R&D work. The entire process of CICC preparation, including extrusion, bending, straightening and aging, was simulated using the N50 stainless steel jacket. The circle-in-square jackets prepared showed a yield strength higher than 1600 MPa, fracture elongation is higher than 30% and a fracture toughness K_{IC} better than 130 MPa·m$^{1/2}$ after cold work and aging at 4.2 K. This paper will focus on the preparation, cold work processing and performance test of the N50 jacket. This study will present experimental data and discuss the feasibility of modified N50 as a high-magnetic field jacket material for next-generation fusion reactors.

Keywords

High strength steel, CICC jacket, Cryogenic mechanical properties, Fusion reactor.

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PS2-73 ABSTRACT-6cec
D. Material Engineering for FNT

Integration and commissioning of the water-cooling system for the beam dump of the linear IFMIF prototype accelerator (LIPAc)

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In the frame of the IFMIF/EVEDA project, a 1.1 MW beam dump has been operated with beam, after an exhaustive characterization and commissioning stage, in the Linear IFMIF Prototype Accelerator (LIPAc). It is designed to stop deuterons accelerated up to 9 MeV with a beam current of 125 mA. The nuclear-grade cooling system is used for circulating water to remove the beam power deposited in the inner copper cone of the beam dump. The design principle of the water-cooling system (WCS) for the beam dump is mainly for circulating water with the velocity between 4–10 m/s in a shroud around of the cone without clogging nor significant corrosion by controlling pH, conductivity and the dissolved oxygen (DO) concentration. The WCS procured by CIEMAT in Spain was delivered to Rokkasho in Japan, and has been integrated into LIPAc facility by QST. After the piping was completed, the inner surface of the stainless-steel pipes was cleaned by pickling and was treated for passivation so as to avoid clogging the narrow water path of the shroud by corroded particles of welded zones. In the first stage of the commissioning, the cooling system was individually tested to confirm that values of pH (8.0 – 8.5), conductivity (0.5 – 2 uS/cm) and DO (less than 10 ppb) could be controlled, within the limits set by previous studies to minimize the corrosion rate of the copper cone. The WCS constituent devices, such as a deaeration system, have been integrated by adapting interfaces of the facility. First deuteron beam of around 20 mA accelerated up to 5 MeV at 0.01 % duty cycle could be satisfactorily introduced into the beam dump without any thermal shock observed in the behavior of water circulation as expected within acceptable ranges of the flow rate and the pressure.

Keywords

LIPAc, IFMIF, Accelerator, Beam dump, Deuteron.

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PS2-74 ABSTRACT-979c
E. Vacuum Vessel and Ex-vessel Systems

Corrosion Phenomena and Deposits in ITER Neutral Beam Test Facility Primary Cooling Circuits

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The ITER Neutral Beam Test Facility (NBTF) is hosted in Padua and includes two experiments: MITICA, the 1 MeV full-scale prototype of the ITER HNB injector, and SPIDER, the 100 keV full-size ITER Radio Frequency (RF) negative ion source. SPIDER and MITICA experiments are actively cooled by UltraPure Water (UPW) to electrically insulate in-vessel components that are biased to high voltage levels. Therefore, water conductivity is the main monitored parameter to ensure components' insulation.

First cooling plant exploitation highlighted that water degrades more quickly than estimated by design either in SPIDER and MITICA, where good water quality is a key feature to operate at 1 MeV. Careful selection of suitable materials for any in-vessel and out-vessel component is of utmost importance, as their interaction with water may affect their chemical and mechanical properties. Furthermore, this interaction can be severely influenced by water chemical characteristics.

The Primary Circuits (PCs) that showed the most severe water degradation during operation are SPIDER and MITICA power supply ones, called PC01 and PC08. This paper describes the results of specific experimental tests performed on these PCs to evaluate possible causes of water degradation and detect sources of contaminants that might compromise future experimental campaigns. The cooling circuits were sectioned and water circulation tests were performed at constant temperature and flowrate. Water samples were collected and Inductively Coupled Plasma Mass Spectrometry (ICP-MS) analyses was used to quantify the type and amount of metals released. Moreover, components samples collected along the circuits were characterized by Scanning Electron Microscope (SEM) technique to detect undesired contaminants, and immersed in UPW to study their corrosion behaviour using metal release tests.

Keywords

Ultrapure Water, Cooling Plant, Water Degradation Issues, NBTF, Fusion Reactors, Corrosion.

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PS2-75 ABSTRACT-9cee
E. Vacuum Vessel and Ex-vessel Systems

Iterative design and structural assessment of the DTT divertor interface to Vacuum Vessel

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The upcoming Divertor Tokamak Test (DTT) facility, whose construction is starting at ENEA Frascati Research Centre in Italy, plays a key role in the EUROFusion roadmap to achieve nuclear fusion. It aims at bridging the gap between ITER and DEMO, testing several divertor configurations and thus boosting the research on the problem of power exhaust in fusion reactors. The DTT divertor system, in the first configuration under development, is made of 54 cassettes. Each cassette is fixed to the Vacuum Vessel (VV) through dedicated support and alignment systems. The interface between the VV and the cassettes fixation systems is realized through dedicated components, the divertor toroidal rails, whose primary aims are to sustain the divertor cassettes, transferring their loads to the VV, and eventually allow the movement of the Cassette Toroidal Mover during Remote Handling operations. This paper describes the activities of iterative design and structural assessment of the divertor toroidal rails. After a deep state-of-the art study and a thorough analysis and definition of the system requirements, a conceptual solution with some similarities with the ITER design has been developed. Then, it has been structurally verified, through Finite Element (FE) techniques, against several load conditions (i.e., assembly condition, plasma operation, disruptive events, baking, maintenance). Particular attention has been paid on the structural assessment against Electromagnetic (EM) loads during disruptive events, namely plasma movements due to degradations of its magnetic confinement which generate strong EM forces on surrounding components. Fast and Slow Downward Vertical Displacement Events (DVDE) have been identified as guiding conditions and the effects of the different forces' components have been analyzed, firstly separately and then in combination. Following an iterative design process, different structural solutions have been designed, tested and compared, and small changes on critical regions made, basing on the results of the assessment.

Keywords

DTT, Divertor, Structural Assessment, Toroidal rails, Iterative Design, Vacuum Vessel.

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PS2-76 ABSTRACT-324b
E. Vacuum Vessel and Ex-vessel Systems

Preliminary Analysis of Phase Transition Characteristics in case of in-vessel LOCA for CFETR Divertor Primary Heat Transfer System

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In case of the in-vessel loss of coolant accident (LOCA), the coolant invaded the vacuum and flash boiling occurred, and the pressure in the vacuum vessel increased sharply. If the speed of the pressurization exceeds the operating rate of the safety device such as burst disk, the vacuum containment boundary will be damaged, leading to the risk of radioactive migration and release. The phase transition characteristics of coolant entering the vacuum vessel are investigated by using the FLUENT code in this paper. The quantitative analysis of vacuum pressure and liquid volume fraction under the accident is carried out. Meanwhile, the influence of initial coolant temperature, break area and discharge pressure on the characteristics of vacuum pressure variation is discussed. The results show that the vacuum pressure rapidly reaches 200kPa at a discharge pressure of 4MPa and multiple pipes rupture. This work provides a reference for the safety design of the primary heat transfer system of fusion reactor.

Keywords

CFETR, in-vessel LOCA, flashing boiling, Fluent.

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PS2-77 ABSTRACT-ed11
E. Vacuum Vessel and Ex-vessel Systems

Mechanical Design of ITER Radial Neutron Camera Ex-Port system

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The Radial Neutron Camera is an ITER diagnostic designed to measure the un-collided 14 MeV and 2.5 MeV neutrons from deuterium-tritium (DT) and deuterium-deuterium (DD) fusion reactions, through an array of detectors covering a poloidal plasma section along collimated Lines Of Sight (LOS). It is composed by two fan-shaped collimating structures viewing the plasma radially through vertical slots in the diagnostic shielding module of ITER Equatorial Port 1: the In-Port RNC, devoted to plasma edge coverage, and the Ex-Port RNC, devoted to the plasma core coverage.

This paper presents an overview of the mechanical design of the Ex-Port RNC at the Preliminary Design Review (PDR) stage. The Ex-port RNC is located in the port interspace and consists of a massive shielding structure hosting the detector units and two sets of collimators lying on different toroidal planes. The collimators are embedded in a stainless-steel supporting structure hosting the radiation shielding material composed by a castable borated hydrogenated mix.

The Ex-port RNC design is presented both from the point of view of functional requirements (e.g. LOS positions and angles, radiation shielding, weight limitations) and of manufacturability. The impact of the manufacturing process on the achievement of the functional requirements is discussed in terms of alignment tolerances and selection of shielding material. Finally, the Ex-port RNC structural integrity is assessed and its design validated against the main loads and load combinations.

Keywords

ITER, Radial Neutron Camera, Ex-Port, Design description, Structural integrity, Finite Element Method.

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PS2-78 ABSTRACT-25c7
E. Vacuum Vessel and Ex-vessel Systems

Conceptual design studies and preliminary analysis of the STEP Vacuum Vessel

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The UK Spherical Tokamak for Energy Production (STEP) program is carrying out concept studies to design a new spherical tokamak fusion reactor. This paper describes the initial conceptual design studies of the STEP Vacuum Vessel. The vessel design studies were carried out to meet the requirements within the restricted space constraints. The design requirements and configuration of the vessel components are described. The preliminary FE structural analysis results are presented.

Keywords

STEP, vacuum vessel, conceptual design, structural analysis.

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PS2-79 ABSTRACT-0467
E. Vacuum Vessel and Ex-vessel Systems

Thermal-hydraulic analysis of the CORC® conductor for the DEMO CS coil

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Conductors using High Temperature Superconductor (HTS) tapes are considered as a very promising solution for future high-field fusion magnets. Various HTS cable concepts, such as e.g. twisted stack cable, cross-conductor (CroCo), Roebel assembled coated conductor (RACC), conductor on round core (CORC®), HTS cable-in-conduit conductor (CICC), have been proposed. Some of them are already considered for potential use in some components of the EU-DEMO magnet system. Recently a design of a conductor based on the CORC® concept for the innermost layer of the hybrid Central Solenoid (CS) coil of EU-DEMO was proposed. In the present work we study its thermal-hydraulic performance in normal operating conditions by numerical simulations using the THEA code by CryoSoft. Taking into account heat loads due to the hysteresis losses, resulting from time evolution of the magnetic field profile along the conductor, we estimate the minimum temperature margin in the conductor. Results obtained with the 1D THEA model are complemented by quasi-3D (1D fluid + 3D solid) ANSYS APDL simulations, aimed at assessment of the radial temperature gradients in the conductor cross section.

Keywords

HTS, CORC®, EU-DEMO , temperature margin.

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PS2-80 ABSTRACT-2ae3

E. Vacuum Vessel and Ex-vessel Systems

Thermal analysis between the high-temperature nitrogen and in-vessel components in the EAST baking process

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In EAST (Experimental Advanced Superconducting Tokamak), the in-vessel components are heated by continuous nitrogen before the plasma operation, which is called the baking process. It is of great significance to explore the heat transfer relationship between the high-temperature nitrogen and in-vessel components in the baking process for the energy transmission and distribution of the baking system. Moreover, the research also provides the basis for further automatic control of the baking system. The flow and heat transfer calculation model is established for the divertor, one of the in-vessel components of baking. The convection heat transfer between high-temperature and the complex flow channel of the divertor in the baking process is simulated by the finite element method. The specific convection heat transfer coefficient, the key parameter of the thermal analysis, is also fitted. The correctness of the calculation results is verified by experiments, which lays a foundation for further thermal analysis of the in-vessel components in the baking process.

Keywords

Thermal analysis, baking process, finite element method, EAST.

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PS2-81 ABSTRACT-0803
E. Vacuum Vessel and Ex-vessel Systems

Design of a Position Monitoring System for the ITER Radial Neutron Camera

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The ITER Radial Neutron Camera (RNC) is a diagnostic system located in the ITER Equatorial Port #1, dedicated to the evaluation of the plasma neutron emissivity. It is composed by two independent collimating structures probing respectively the plasma edge (In-Port RNC subsystem) and the plasma core (Ex-Port RNC subsystem). The Ex-Port RNC contains 16 line-of-sight (LOS), distributed in two radial planes and located inside a shielding block installed on the Interspace Supporting Structure (ISS); each LOS has an associated optical path which penetrates the Diagnostic Shielding Module (DSM) of the port plug. Since the port plug and the ISS are mechanically disjointed, a Position Monitoring System (PMS) is foreseen in order to evaluate the misalignment between each LOS and the related DSM penetration, which can occur during the installation, baking and operation of the machine. Displacements data measured by the PMS will be processed to correct for the following two effects induced by misalignment:

- Vignetting, i.e. partial/total loss of direct view of the plasma along a LOS
- Mismatch between the reference LOS positions used by the emissivity profile reconstruction code and the actual LOS positions

Devices based on different technologies have been investigated and compared. A PMS based on Fabry Perot displacement sensors has been identified as the best solution to measure the displacements from reference positions along the radial, vertical and toroidal directions. Potential issues that could arise in the installation and operation of such sensors have been evaluated; MCNP calculations have been in particular performed to determine the radiation level at the position of the PMS components. Finally, mechanical tests have been carried out at ENEA to evaluate the performance of the sensors and results indicate compliance with the specifications provided by the producer and the ITER RNC design requirements.

Keywords

ITER, Radial Neutron Camera, Position Monitoring System.

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PS2-82 ABSTRACT-08fa
E. Vacuum Vessel and Ex-vessel Systems

Analysis and optimization of the secondary circuit for the option WCLL BB Direct Coupling with small ESS of the EU DEMO power plant

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The European DEMOnstration Fusion power plant (EU-DEMO) is being designed by the EUROfusion consortium to demonstrate electricity generation at the level of a few hundred MW from nuclear fusion. The Primary Heat Transfer System (PHTS) transfers heat from the reactor sources, namely: Breeding Blanket (BB) and Divertor (DIV), to the secondary circuit called Power Conversion System (PCS) in which thermal power is converted into electricity. As a result of the DEMO pre-conceptual studies two reference concepts of the BB and the respective PHTS have been selected: the Helium Cooled Pebble Bed (HCPB) and the Water Cooled Lithium-Lead (WCLL). One of problems faced by the DEMO plant designers is the cyclic tokamak operation: 2-hours long plasma pulses will alternate with dwell phases lasting 10 minutes. To cope with the drastic power drop of the reactor heat sources during dwell, an Energy Storage System (ESS) of different size is included in the considered DEMO plant configurations. According to the recent DEMO Energy Map data there are significant uncertainties regarding the partition of the fusion power among different reactor sources. Because of these uncertainties 12 possible operational scenarios are considered for each of HCPB and WCLL variants.

In 2022 we created the GateCycle model of the “maximum of maximum” PCS cycle, in which all circuit components were sized to absorb the maximum power of all the reactor heat sources. In the present work we check the flexibility of this “maximum of maximum” cycle to accommodate the most problematic operational scenario (A1-2) in which the DIV power is the highest, whereas the BB power is the lowest. We discuss the observed problems and propose layout modifications of the circuit to cure them.

Keywords

EU DEMO, Power Conversion System, Water Cooled Lithium Lead breeding blanket, Small Energy Storage System, GateCycle.

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PS2-83 ABSTRACT-3da8
E. Vacuum Vessel and Ex-vessel Systems

Investigation on the pressure drop characteristics of ITER thermal shield cooling pipes

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Thermal Shield (TS) is located between vacuum vessel/cryostat and superconducting magnets in ITER tokamak to protect the magnets by blocking the thermal radiation coming from the warm components. The TS panels has cooling pipes welded on their surface and 80 K helium gas flows inside the pipes for active cooling. The helium coolant is supplied from the cryoplant and distributed to all TS panels by manifolds. Allowable pressure drop through the TS panels is less than 1 bar. Flow test was performed for the representative TS panels at the factory to measure the pressure drop under actual operating condition of the ITER TS.

This paper presents the experimental results of the pressure drop through the cooling pipes of the ITER TS panels. High pressure and room temperature nitrogen gas is selected for the experiment instead of helium, which can be equivalent operating conditions with the 80 K helium gas flow from fluid mechanics point of view. Flow rate is controlled by a thermal mass flow controller and pressure drop between the inlet and the outlet of the pipe routing is measured by a differential pressure gauge for different flow rates. The experimental set-up is open loop type, which is connected to high pressure gas bottle. Flexible corrugated pipes are used to connect the TS cooling pipes with the measurement system line. The effect of the corrugated pipe is evaluated separately and excluded from the measurement results of the TS pipes. Test results of several TS panels are shown in this paper and are compared with the calculation using existing friction factor correlation and local loss coefficient of the pipe bends. The effect of secondary pressure loss in the bends is evaluated separately for different TS panels and compared with frictional pressure drop.

Keywords

Pressure drop, cooling pipe, friction factor, thermal shield, ITER.

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PS2-84 ABSTRACT-49bb
E. Vacuum Vessel and Ex-vessel Systems

Design and Performance Research of Baking for HL-2M Vacuum Vessel

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The vacuum vessel is a major safety barrier and provides high quality vacuum for plasma. Before plasma operation, the vacuum vessel will be baked to remove the moisture. HL-2M vacuum vessel has a D-shaped, double wall structure. The ribs between inner shells and outer shells give required mechanical strength and also form flow passages for the vessel baking. HL-2M vacuum vessel is baked with hot nitrogen gas. The design of the flow passages is done. According to the structural characteristics of HL-2M vacuum vessel, the ribs are of U-shaped types and there are 20 fluid loops. Fluid analysis is performed and shows that the distribution of ribs is reasonable. At the first plasma discharge of HL-2M device, the temperature difference could be less than 10℃ and the heating power was about 10kW when the vacuum vessel was baked to 100°C. The vacuum pressure reached 2.4×10^{-6} Pa. In 2022, the baking temperature was 135 ℃ after the upgraded first wall and divertor were installed. The vacuum vessel had good expansion uniformity performance with temperature rise. The vacuum pressure reached 5×10^{-6} Pa. It provided good vacuum condition for the discharge of plasma current exceeding 1 million amperes. These results showed that the baking system with special temperature control could make the vacuum vessel obtain the required baking temperature, good temperature uniformity and excellent vacuum performance.

Keywords

HL-2M vacuum vessel, flow passage, baking performance.

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PS2-89 ABSTRACT-7231
F. Nuclear System Design

Overview on the conceptual design of TRUST: the new nuclear fusion experiment of University of Tuscia

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TRUST (Tuscia Research University Small Tokamak) is a new small-scale tokamak currently under design and construction at the University of Tuscia (Unitus). This paper gives an overview of the TRUST conceptual design which includes a set of toroidal field (TF) copper coils, poloidal field (PF) copper coils and a copper central solenoid (CS) to achieve the target single null (SN) plasma configuration. The experiment is designed for a first stage of operation with the following main parameters: R_0 = 0.3 m, A = 2.6, I_{pla} = 0.150 MA, B_t = 1.0 T, q^* = 4. The main aim of this experiment is to realise a flexible University-class experiment that should prepare at the best the students to study the power exhaust problem on a long a time scale. The system could easily allow to the students to replace the plasma facing components (PFC) and, consequently, testing any future innovative technologies (i.e., liquid metals and meta-materials). In order to accomplish with the necessity of very long time plasma duration, TRUST is planning to use an electron-cyclotron resonance system for the plasma heating and current drive. In the first phase TRUST will be made of copper, in parallel an upgrade for the TF coil system, the CS and at least one of the PF coils is planned using windings of high-temperature superconducting (HTS) material. Therefore, the whole TRUST facility and its auxiliary systems are developed already in order to be compatible with this kind of upgrade.

Keywords

TRUST, UNITUS, Tokamak, Fusion Power Exhaust, Liquid Metal, Meta-materials, HTS.

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PS2-90 ABSTRACT-1c2f
F. Nuclear System Design

Development Progress of Advanced Neutronics Software TopMC

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The TopMC (advanced version of SuperMC) has implemented the whole process neutronics simulation, which includes the neutron/photon/electron transport, material activation and shutdown dose calculation. The coupling transport calculation of neutron, photon and electron has been newly added, which can make the dose of small structure components more accurate. The improved D1S for SDR calculation based on response of collision with nuclide was developed, which can consider all decay photons emitted during the complete transmutation process including multistep reactions and cascade decays of all nuclides under neutron radiation. For improving the efficiency and correctness of modeling of complex systems, the physical parameter visualization method and function has been developed, which realizes efficient and smooth visualization for source and tally to inspect the features of physical parameter. TopMC has been verified and validated by more than 2000 benchmark models and experiments, including ICSBEP, SINBAD, IRPhEP, etc. A neutronics analysis for European HCPB DEMO has been newly accomplished, in which the nuclear performance including neutron wall loading, tritium breeding ratio, nuclear heating and radiation loads on first walls, divertor and TFC was calculated and analyzed.

Keywords

TopMC, electron transport, improved D1S, physical parameter visualization.

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PS2-91 ABSTRACT-1fce
F. Nuclear System Design

Design and integration of Tokamak Coolant Systems based on water technology in the EU-DEMO

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One of the fundamental goals of EU-DEMO is to demonstrate the feasibility of producing net electrical power from fusion.

The design of the primary heat transport systems is dependent on a multitude of requirements and constraints. Primarily, the thermal energy that is deposited in the first wall, blanket, divertor and limiters must be removed and transferred to the power conversion system at sufficiently high temperatures to achieve economically attractive efficiency. Additionally, the heat transport systems must provide a reliable barrier against radioactivity releases and be designed to ensure integrity throughout the plant lifetime. As such, safety considerations are critical from the early phases of the heat transport system's design as the minimisation of radiation exposure and plant activity levels must be pursued.

The system design must cope with the contamination of the cooling water with activated products, and the transportation of these radiation sources outside the primary shielding by the coolant circulation. These activated products include strong gamma and neutron emitters. Finally, the significant operating temperature and pressure of the coolant results in high energy stored within the system.

This paper focuses on the requirements and constraints with the highest impact on these systems' integration within the tokamak building, whilst identifying relevant quantities associated to the aforementioned hazards. The integration concepts adopted to safely operate water coolant systems in a nuclear fusion device are outlined below.

The design principles of EU-DEMO adopt the approach implemented in fission power plants, where the entire coolant system is integrated in building areas with massive concrete shielding and segregated from other plant systems. Furthermore, the building is adapted to withstand the consequences of an accidental release through the integration of a pressure suppression system. Finally, the coolant system is equipped with purification systems to prevent the accumulation of tritium and activation products.

Keywords

DEMO, TCS, WCLL, PHTS, Safety.

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PS2-93 ABSTRACT-2560
F. Nuclear System Design

Development of a Cherenkov probe with high time resolution for runaway electron measurements in the HL-2M Tokamak

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Runaway electrons (REs) is one of the major concerns for the future reactor-scale tokamak's safe operation. The relativistic REs usually generate during low-density discharges and major disruptions of large-scale tokamaks, which may carry energy ranging up to dozens of MeV. These electrons may cause severe damage to the plasma facing components (PFC) and even ablate cooling ducts. An accurate measurement of RE's parameters is essential to understand their generation and transport mechanisms, and eventually ensure the safety of tokamaks. Recently, several diagnostics for the detection of runaway electrons based on the Cherenkov effect have been developed on different tokamaks, e.g. FTU and TORE-SUPRA. This probe has been proved to be able to provide detail information of the loss of super-thermal electrons at a specific location in the chamber of tokamaks.

In this paper a diagnostic system based on the Cherenkov effect is reported for the detection of runaway electrons in the HL-2M tokamak. Sapphire (Al2O3) is chosen as the radiator, which can measure runaway electrons of energy above 120 keV. This radiator is mounted on a reciprocating probe and the Cherenkov signal can be transmitted through an optical path to the Silicon Photomultiplier (SiPM). This probe has been installed on the HL-2M tokamak and put in application in 2022's experiments. Its validity has been proved by comparing its signals with that from other diagnostics like Mirnov probe, hard x-ray, soft x-ray and neutron. The enhancement of loss of runaway electrons during sawtooth mode and tearing mode are confirmed by this Cherenkov probe. By comparing to soft X-ray and the ECE signals, it is found that the Cherenkov probe signal are significantly enhanced during the sawtooth mode. In future, this probe will be further used to study the generation and loss mechanism of runaway electrons.

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Keywords

Runaway electron, high temporal resolution, Cherenkov probe.

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PS2-94 ABSTRACT.- cc48
F. Nuclear System Design

Nuclear analyses for the ITER Equatorial Port #8

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The ITER diagnostic Equatorial Modular Port Plugs (EPPs) are designed to host several components such as structural and shielding elements, feedthroughs, as well as different diagnostic systems. Therefore, nuclear analyses play a fundamental role in the integration process of these plugs since they have a significant impact on the expected radiation environment in the Port Interspace and Port Cell (PC).

The neutronic analyses presented in this paper are focused on the standard shielding and integration solutions on the EPP #8. This Port Plug hosts the Disruption Mitigation System and other five diagnostic systems distributed in the three Diagnostic Shielding Modules: the Lost Alpha Monitor, the Tangential Neutron Spectrometer, the Visible Spectroscopy system, the Density Interferometer Polarimeter and the Flow Monitor.

A detailed MCNP model of the EPP #8 has been developed and integrated in the proper 120° sector extracted from ITER MCNP E-Lite model. This model has been used to assess the neutron and gamma fluxes and spectra along the port from the First Wall up to the PC. Neutron and gamma heating and neutron damage on the structural components and diagnostic systems have been computed as well. The contribution due to the neighbouring ports, in particular the streaming of the NBI ports located on the EPP#8 side, has been evaluated as well.

This analysis has been performed by means of the D1SUNED v3.1.4 Monte Carlo transport code. Variance reduction techniques have been applied through the generation of specific neutron/photon weigh windows by means of the ADVANTG v3.0.3 tool, in order to achieve statistically satisfactory results in the zones of interest. The outcomes of the analyses are presented and discussed.

Keywords

Neutronics, Nuclear Analyses, MCNP, Integration, ITER, Fusion Neutronics.

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PS2-95 ABSTRACT-433d
F. Nuclear System Design

Development and Test of In-vessel Coil System for Magnetic Compression of Field Reversed Configuration Plasmas

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Field-reversed configuration (FRC) has the potential for a low-cost and low-complexity fusion reactor. An FRC experiment device named HUST-FRC (HFRC) is constructed at Huazhong University of Science and Technology. The HFRC experiments will investigate the formation, translation, collisional-merging, and magnetic compression of FRC. To compress the FRC and improve its temperature and density, a set of single-turn coils are designed with a large current rising to ~100 kA within 50 µs. The coils should be installed inside the compression chamber to avoid the shielding effect of the metal chamber wall. An improved coil conductor structure is used to achieve a low toroidal ripple of the compression field, with the main body and leads welded together instead of bending the conductor directly. In addition, feedthroughs installed on the chamber wall are designed to transfer the large current between the power supply and the in-vessel coils. During the compression process, the coils and feedthroughs operate at high voltage large current conditions, and they are subjected to large electromagnetic forces. These will challenge their structural stability and insulation performance. To verify the reliability of the design, a prototype system including the coil and feedthrough is developed, and energized by an existing high-voltage pulsed power supply. This paper presents the design, fabrication and testing of the prototype system.

Keywords

Field-reversed configuration, magnetic compression coil, low toroidal ripple, feedthrough, experimental test.

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PS2-96 ABSTRACT-4966
F. Nuclear System Design

A Two-Step Neutron Energy Regulation Method based on Multi-Dimensional Response Matrix

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In the design of nuclear fusion devices and related applications, it is often necessary to adjust the layout of the moderator to obtain the neutron spectrum of the target, and the traditional neutron spectrum regulation technology relies on manual experience to iteratively design. In this paper, optimization and neural network methods are used to adjust the layout and thickness of the moderator. First, the multi- dimensional response matrix of different moderators is calculated by neutron transport simulation, and the arrangement of moderators is optimized by differential evolution algorithm. Then, the artificial neural network algorithm is used to optimize the thickness of the moderated. The moderated collimation system of D-T neutron radiography and the TBM of ITER have been redesigned using the method developed in this paper, and the energy spectrum after regulation has been significantly improved.

Keywords

Two-step, artificial neural network, differential evolution, response matrix.

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PS2-97 ABSTRACT-5089
F. Nuclear System Design

Integration of the Beam Dynamics and Neutronics simulation tools for the IFMIF-DONES accelerator design

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In this work we present the coupling of the simulations corresponding to two different physical phenomena.
The former being the beam dynamics (BD) in the high energy line of an accelerator, and the latter corresponding with the neutron flux transport produced by beam-target nuclear reactions. This work is held within the framework of the International Fusion Irradiation Facility-DEMO Oriented NEutron Source (IFMIF-DONES) project, whose main scope is to test materials capable of withstand the extreme irradiation conditions that future fusion reactors will have to manage. In order to reach these conditions, a deuteron beam is accelerated until it reaches a 40 MeV energy and 125 mA and is shaped so it has an specific profile by the end of the High Energy Beam Transport line (HEBT).

The beam then collides with a Lithium target, causing nuclear reactions that yield a neutron flux which will produce similar radiation effects to those expected in nuclear reactors. We focus on how the beam dynamics of the HEBT, i.e. the profile shaping, affects the neutron flux distribution after the collision. Specially important will be to analyze the impact of the beam size or the effect of side peaks produced by octupoles on the ultimate neutron distribution. The BD analysis is carried out by the TraceWin software, while the neutron transport calculations are performed with McDeLicious, which is based on the MCNP6.2 code.

A coupling tool has been developed, running sequentially the BD simulations producing the beam at target, and then the McDeLicious code, computing the neutronic field distribution. This coupling serves multiple purposes. With this tool, the impact of the HEBT configuration on the neutronic distribution can be easily determined.

In the future, an optimization procedure where the TraceWin beam dynamics calculations will be optimized according to the McDeLicious results will be developed.

Keywords

Ifmif-dones, neutronics, high energy beam transport, coupling, early design, beam dynamics, tracewin, mcdelicious, mcnp, accelerator.

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PS2-98 ABSTRACT-5105
F. Nuclear System Design

Balance of Plant conceptual design of EU DEMO integrating different Breeding Blanket concepts

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The European DEMO is considered the nearest-term fusion reactor with the aim to generate several hundred MWs of net electricity, operate with a closed tritium fuel-cycle (achieving the tritium self-sufficiency) and qualify technological solutions for a Fusion Power Plant.

Two Breeding Blanket (BB) concepts - relying on different cooling and breeding technologies - are considered for the baseline design: the Helium Cooled Pebble Bed BB and the Water Cooled Lithium Lead BB.

The selection of the BB type has a strong impact on the whole DEMO plant design and, in particular, on the Balance of Plant (BoP) which is mainly composed by three systems:

- a Primary Heat Transfer System (PHTS) to remove the plasma generated thermal power;
- a Power Conversion System (PCS) to convert the thermal power in mechanical and finally in electrical energy;
- eventually a Energy Storage System (ESS) introduced with the purpose of buffering some energy produced by the pulsed plasma source to allow a continuous production of electricity.

The aim of the present work is to investigate a possible concept design of the BoP accommodating two different BB: the Driver Blanket (the basis configuration) which provides the largest part of the power (P) generated in the Tokamak (15/16 of P), whereas the Alternate Blanket (namely an advanced configuration) provides the complement (1/16 P). In this study, the WCLL BB is postulated as Driver BB while the HCPB BB is considered as Alternate Blanket.

In the proposed configuration, the HCPB BB tokamak sector is connected to the ESS in order to store thermal energy during pulse to be used in dwell time to keep synchronized the turbine and warm-up the PCS components.

A flexible tool made with Matlab® was developed to study this configuration and allows to propose several uses of the heat fluxes.

Keywords

DEMO, Balance of Plant, Power conversion system.

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PS2-99 ABSTRACT-6d3b
F. Nuclear System Design

Analysis on fluid-induced acoustic resonance coupling in series T-structure fusion device

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In the pipe system of the fusion device, there are several T-junction pipe structures connected in series and parallel at the distribution console. When TCWS (Tokamak Cooling Water System) performs blow and bake operations, the flow induced acoustic resonance will occur in the vocal cavity of the closed T-junction pipe branch within a certain flow rate range of the main pipe. In this paper, referring to the structure of EAST (Experimental Advanced Superconducting Tokamak) gas-water distribution console, a series and parallel connected closed T-junction acoustic resonance coupling experiment is designed. Simulation and experimental research are carried out. The basic elements of acoustic resonance in single T-channel are summarized and the coupling relationship of acoustic resonance in multi-series parallel T-junction is explored. The accuracy of the simulation coupling model is verified by experiments. It provides basic design basis for avoiding acoustic resonance in T-junction pipe design process.

Keywords

EAST, Tokamak Cooling Water System, acoustic resonance.

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PS2-101 ABSTRACT-aa67
G. Safety Issues and Waste Management

Requirements analysis for fusion reactor safety analysis software development based on existing one for fission reactor

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Systematic accident analysis is a significantly important and necessary phase for reactor primary heat transfer system (PHTS) design. Over the past several decades, many systematic accident analysis software and modules, such as RELAP5, CHATHARE, MELCOR, etc., have been developed to be applied on fission reactor accident simulation. However, there is no dedicated systematic safety analysis software for fusion reactor at present, most fusion reactor PHTS safety analysis research are just carried out on the existing original fission safety analysis software, which could cause deviation between real situation and simulation result due to the significant differences on the functional components and operation environment between the two kinds of reactor. Therefore, it is essential to develop a dedicated fusion reactor systematic safety analysis software. At the moment, a nuclear system safety code is being developed with the support of Chinese fusion community. The baseline version of the code was extracted from COSINE, which is the software platform for fission power plant design and safety analysis being developed in China.

To develop this software, the adaptability analysis of this software for the fusion reactor is needed in the initial phase. In this paper, the different characteristics of PHTS accident between fission and fusion reactor is analyzed, and the related modules to be modified or developed were then sorted out, including heat transfer and material properties module, etc. This part of work will provide demands for the following development of the software and references for the fusion reactor safety analysis.

Keywords

Water-cooled divertor, Accident analysis, loss of coolant accident(LOCA).

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PS2-102 ABSTRACT-c750
G. Safety Issues and Waste Management

Electronic effects of aryl groups in modified polysilane liquid scintillator on its fluorescence properties

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Radiation detection is the key technology for the construction and safe operation of fusion reactor. Liquid scintillator is widely used in this field but its fluorescence properties still need to be improved for high sensitivity detection at present. Polysilanes are unique luminescent materials, mainly because of the continuous Si-Si single bond in the main chain and the large radius of the Si atom with 3d orbitals, which creates a localized delocalized electron channel along the Si-Si main chain. Further enrichment of the delocalized system can effectively enhance the fluorescent properties of material. In this paper, the optical properties of modified polysilanes under the influence of several groups were calculated by model building and molecular simulation in Gaussian software, and several typical materials were successfully synthesized by aryl Grignard reagent routes. Then the optical properties of the materials were evaluated and compared with the calculations to determine the electronic effect of aryl groups on the electron delocalization in modified polysilane scintillator. The scintillation counting performance in solvent was also tested and the scintillation counting rate under natural background γ radiation was determined. It could be concluded that polysilanes modified aryl group can be used as a new type of liquid scintillator and the induction effect of electric absorption could enhance the fluorescence properties more effectively, which providing meaningful guidance for the further improvement of liquid scintillator.

Keywords

Polysilane, aryl group, electronic effects, liquid scintillator, fluorescence.

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PS2-103 ABSTRACT-dc97
G. Safety Issues and Waste Management

Neural network-based source biasing to speed-up challenging MCNP simulations

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Nuclear analysis of nuclear fusion facilities results of paramount complexity, particularly in the case of ITER. The need to consider high-fidelity modelling of the geometry and the radiation sources make the Monte Carlo the most suitable approach. MCNP stands as the most widely used code.
A high variety of variance reduction techniques (VR) have been developed to speed-up the calculations, enabling the application of MCNP to the analysis of fusion devices with unprecedented accuracy. However, a challenge remains to simulate the coexistence of deep penetration and intense streaming, a most common situation in nuclear fusion. Source sampling bias is a suitable VR technique for such situations. The spatial, energy and angular resolution considered to describe the bias greatly affects the achievable efficiency, what sometimes results limiting.
In this work, the application of neural networks to bias the source sampling is explored. This method samples the phase space of the source according to a prediction of the potential contribution to a given tally. Thus, the computational load is spent only in those histories judged of relevance. The goal of this article is to show the capability of the method to optimize the prediction of the source contribution to the tally, improving the expected performance of the MC simulation. It has been tested in a simple but representative source-geometry arrangement. The computational time of the MC method to reach acceptable statistics is greatly reduced with promising performance for extrapolation to more complex cases compared to the state-of-the-art techniques. The good performance encourages further development of the method to be applied
in ITER-like complex cases.

Keywords

MCNP, variance reduction, machine learning, neural networks.

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PS2-105 ABSTRACT-51a6
G. Safety Issues and Waste Management

Qualitative safety analysis for the Divertor Tokamak Test (DTT)

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The Divertor Tokamak Test (DTT) facility will emit ionizing radiation during its operation, both due to D-D (deuterium-deuterium) reactions and to accelerating electrons and other particles within the plasma. Therefore, the plant must be designed and must operate according to the specific legislation of the Italian state.

The licensing process provides for the execution of safety analyses dedicated to demonstrating compliance with the legal limits in the various operational phases and in the incidental/accidental conditions that could occur.

Accordingly, safety analyses begin at an early stage of project development. Evaluations of source terms such as tritium, activated dust in the plasma chamber, and activated corrosion products (ACPs) in the cooling circuits are ongoing. In parallel, failure mode and effect analyses (FMEAs) at the functional level and at the component level allow the identification of the full set of possible initiating events and the selection of the reference accident sequences. Subsequently, deterministic assessments of the reference accidents will evaluate the possible consequences and will demonstrate the compliance with the safety limits.

In this work, the following investigations are presented:

- the functional analysis, that defines the process and safety functions provided by systems, structures and components (SSC) in DTT;
- The functional FMEA in terms of methodology used and results obtained when the design was only in a pre-conceptual stage;
- The FMEA at component level both in terms of results obtained and design improvements needed to reduce safety risks

Keywords

FMEA, DTT, Accident análisis.

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PS2-106 ABSTRACT-5412
G. Safety Issues and Waste Management

Mechanical design of a Fast Shutter for the Disruption Mitigation System

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ITER is a large-scale experimental nuclear fusion reactor currently under construction. To protect the first wall components of the machine against damages due to a plasma disruption event, ITER needs a Disruption Mitigation System (DMS). The ITER DMS is based on the Shattered Pellet Injector (SPI) technology. One of the issues with SPI injection is that the propellant gas may overtake the pellet and start plasma edge cooling before material deposition by pellet fragments took place. Edge cooling may increase runaway electron generation, therefore should be avoided. To address this problem, the ITER Organization has launched a project to develop Fast Shutter which serves an important role by closing the orifice right after the pellet passed the component. The shutter must be closed within a few milliseconds to block enough gas. It will be installed close to the torus, where the equipment will be exposed to a high magnetic field, and neutron irradiation. Due to limited accessibility for maintenance, the shutter must operate thousands of cycles. This paper describes the detailed design of the shutter and its present capabilities. The proposed concept addresses the issue of high dynamic loads resulting from the fast-closing time of the device, which can cause fatigue due to cyclic loading. The unique environmental conditions involved in the operation of the device necessitate the use of special actuators and materials. The design space available for the implementation of the concept is relatively small, which requires the optimization of the arrangement of device components to minimize the overall loads and maximize the device's efficiency. The optioneering [1] and the laboratory testing of the actuator [2] are also presented at the conference to give a whole picture of the project.

[1] A. Zsákai et al at this conference

[2] D.I. Réfy et al at this conference

Keywords

DMS, fast shutter, SPI.

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PS2-107 ABSTRACT-545a
G. Safety Issues and Waste Management

Overview of detritiation processes applicable to tritiated soft housekeeping waste

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During its operating and dismantling phases, the ITER facility will generate different types of tritiated radiological waste for which suitable outlets must be provided. Depending on national waste management strategy and waste acceptance criteria of final repositories, the management of this tritiated waste could require a preliminary tritium removal phase (interim storage for natural tritium decay, tritium removal treatment or combination of treatment and interim storage) prior the waste transfer to final repository.

The objective of this paper is to describe the detritiation processes that could possibly be used on solid tritiated soft housekeeping waste, and to investigate the technical relevance of implementing a preliminary treatment whenever possible.

This study concludes that there are a number of efficient detritiation processes. Their potential benefits rely in positive impacts on options of tritiated waste management scenario with a quantifiable safety, technical and economic effects (e.g. radioprotection issues, duration and surface area of possibly required interim storage for tritium decay, waste volume to be transferred to final repository, overall tritium discharge levels of the waste management scenario…). It must nonetheless be remembered that all tritium removed from this waste represents a secondary waste which also needs to be managed. The entire issue of detritiation is thus based on finding the right balance between process efficiency and the production of a minimum volume of secondary waste.

Furthermore, though solutions exist, the implementation and operation of these processes can be costly, especially for irradiating waste such as LL-ILW. The safety, technical and economic relevance of the entire waste management system remains to be assessed, but first the chosen processes will need to be optimized either via engineering studies if the process is sufficiently mature or via R&D if the impact of the different operational parameters on the process performance levels is poorly known.

Keywords

Waste treatment, fusion, soft housekeeping, tritium removal.

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PS2-109 ABSTRACT-56aa
G. Safety Issues and Waste Management

LIFUS5/Mod3 Series-E Experiment Test 6.1 for SIMMER-III code validation

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Several research activities and experiments are being conducted at the University of Pisa and the ENEA Brasimone Research Center to explore the dynamics and responses during a hypothetical in-box Loss of Coolant Accident (LOCA) in the water-cooled lead-lithium breeding blankets (WCLL-BB) and during the system's protective response. Many projects are underway to improve the predictability of computational tools and evaluate the safety analysis system codes, modeling methodologies, and effective implementation.

SIMMER-III is a system code widely used for analyzing the behavior of nuclear systems during transients and accidents. It is particularly well-suited for multi-fluid flow and interaction analysis, which makes it a valuable tool for investigating the behavior of systems such as WCLL-BB. One of the strengths of SIMMER-III is its ability to simulate the behavior of fluids in various conditions, including at high temperatures and pressures and during multi-phase flow. This makes it particularly useful for studying the behavior of fluids in nuclear systems, where these conditions are often present. The University of Pisa has successfully upgraded the SIMMER-III code to account for the chemical interaction between the two fluids (lead-lithium and water).

In the current investigation, the SIMMER-III algorithm numerically reproduced the experimental Test E6.1 performed in the separate effect test facility LIFUS5/Mod3. The study presented in the paper focused on improving the SIMMER-III system code validation and verification for the multi-phase flow of lead-lithium and water and their chemical interaction using the LIFUS5/Mod3 facility and test E6.1 experimental data. Overall, this work aims to increase the accuracy and reliability of simulation tools for properly predicting WCLL-BB safety analyses.

Keywords

LIFUS5/Mod3 experimental campaign, SIMMER-III code validation, In-box Loss of Coolant Accident; water-cooled lead-lithium breeding blankets (WCLL-BB); chemical interaction between the two fluids (lead-lithium and water);.

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PS2-110 ABSTRACT-5888
G. Safety Issues and Waste Management

Eddy current actuated fast valve development for disruption mitigation applications

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A disruption mitigation system (DMS) is critical for the safe and reliable operation of magnetically confined fusion devices. DMS aims to prevent or minimize the effects of disruptions by rapidly controlling the plasma and dissipating its energy. Two common techniques used in DMS are Massive Gas Injection (MGI) and Shattered Pellet Injection (SPI), both aiming injection of considerable amount of material into the plasma, the former in gas while the latter in cryogenic pellet form. In both cases, a fast-acting valve is needed for operation, but with different features: an MGI requires a larger stroke length of the piston to fully empty the gas reservoir, while an SPI requires a shorter stroke length and quick opening-closing to reach a sufficient pressure peak at the pellet for launching and minimize the expelled gas amount.

Recently, we developed an eddy current actuated high-pressure fast valve at the Centre for Energy Research (EK-CER) dedicated to our SPI test bench. As a continuation of this work, we have developed a Tritium compatible valve which is suitable for both MGI and SPI purposes.

In this contribution, we discuss the results of the development and laboratory testing of our eddy current actuated fast valve.

Keywords

Fast valve, Eddy current actuated valve, High pressure valve, Disruption Mitigation System, DMS, Massive Gas Injection, MGI; Shattered Pellet Injection, SPI.

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PS2-111 ABSTRACT-58c0
G. Safety Issues and Waste Management

A Proportionate Approach to Regulation for Fusion Power Plants

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Concepts for fusion energy prototype power plants are now being developed around the world by research organisations and private companies, some targeting deployment in the 2030s and 2040s. This move, from research to power generation, will see fusion devices on a much larger scale in terms of characteristics such as power, fuel throughput, neutron production and operational time, leading in most instances to an associated increase in the level of hazard and waste arisings. Fusion power plants will need to be regulated appropriately and proportionately to maintain public and environmental protections, and to provide public assurances.

The Fusion Safety Authority at UKAEA reviewed existing studies on the safety and waste aspects of early concept fusion power plant, publishing the "Technology Report – Safety and Waste Aspects for Fusion Power Plant". The report outlines some of the proposed fusion technologies, identifies key radiological hazards of fusion and discusses the nature and magnitude of these in the context of a fusion power plant.

The report formed part of the supporting documentation for the UK Government's proposal for fusion regulation. In June 2022, the UK Government published its decision on the regulation of future fusion energy facilities which will continue to be regulated by the Health and Safety Executive and the Environment Agency (or devolved regulator). The UK Government does not consider fusion energy facilities to be nuclear installations, and so will not require a nuclear site licence.

This presentation summarises the findings of the Technology Report with regard to the radiological hazards and potential public dose consequence from indicative fusion power plant accident scenarios, described through consideration of the radiological inventory (e.g. tritium and activated dust) and multiple layers of inventory confinement.

Keywords

Safety, Regulation, Fusion Power Plants, Tritium.

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PS2-113 ABSTRACT-b35e
H. Models and Experiments for FNT

ITER relevant conditions of the closed-water activation loop at the JSI TRIGA

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Numerous computational analyses of the water activation process have been performed for ITER, but the understanding of cooling water as a radiation source is still inadequate due to a lack of experimental nuclear data and inconsistencies among major nuclear data libraries, inaccurate computational methods/codes considering time- and spatial dependent radiation sources (such as the flow of activated water in the cooling system), experimental facilities to validate the methodology and, most importantly, the lack of water activation experiments under fusion-relevant conditions. For this reason, a closed-water activation loop is being built in the Jožef Stefan Institute (JSI) at the research reactor TRIGA, which will serve as a well-defined and stable 6 MeV-7 MeV gamma ray and neutron source. Such a high-energy radiation facility will enable various experiments based on water activation, which are essential to fill knowledge gaps and improve existing experimental nuclear data sets, study detector response to high-energy gamma rays, explore short-lived moving radiation sources, validate computational codes and methods, etc. The aim of this work is to present the current status of the closed water loop and its relevance to the conditions of ITER. Furthermore, a replacement of the inner irradiation part is planned for the second phase of operation, with the features of the first wall of ITER being preserved as far as possible. In addition, a modification of the non-irradiated downstream components (decay tank, collectors, etc.) is also being considered.

Keywords

Water activation, TRIGA, ITER.

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PS2-114 ABSTRACT-4d88
H. Models and Experiments for FNT

Effect of argon additive in Cs-free negative hydrogen ion source TPDsheet-U

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The progress toward realizing a high-performance cesium (Cs)-free negative-ion source based on volume production in a magnetized sheet plasma device (TPDsheet-U) is reported [1-3]. In the experiments, H^- ions were successfully extracted from sheet plasma using single-aperture grids when argon is added to hydrogen plasma at the external magnetic field strength of 38 mT. The experimental results of the single-aperture grid show that the negative hydrogen ion current density ($J_C(H^-)$) performance of Cs-free H- ion beams in TPDsheet-U is about one-fourth that of Cs-containing H^- ion sources in the ITER negative-ion neutral-beam injector (NNBI)[4]. (i) The $J_C(H^-)$ was ~7.5 mA/cm^2 at an extraction voltage of 10 kV, a discharge current Id of 90 A, and a gas pressure of 0.3 Pa without argon. (ii) Co-extracted electrons $J_{EG}(e)$ are successfully suppressed by setting a soft magnetic filter (SMF) on plasma grid. The $J_{EG}(e)/J_C(H^-)$ ratio was reduced to below 0.5 for a single-aperture grid with the SMF. (iii) The $J_C(H^-)$ was increased by a factor of 1.18 when adding argon at the base pressure below 0.3P and the Id of 80 A.

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Keywords

Negative ion,TPDsheet-U,sheet plasma.

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PS2-115 ABSTRACT-d064
H. Models and Experiments for FNT

Preliminary Electromagnetic Analyses of the Effects of Plasma Disruptions on SPARC Vacuum Vessel for the Construction of a Synthetic VDE Database

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On its path to fusion power, Commonwealth Fusion Systems (CFS) will build the SPARC tokamak, a compact high-field burning plasma experiment, with the goal to demonstrate net fusion energy gain. SPARC is designed to operate at 12.2 T with a plasma current of 8.7 MA in a double null configuration [1]. The combination of high toroidal magnetic field and high plasma current make SPARC less tolerant to disruptions than currently operating research devices. For this reason, the need to estimate the Electro-Magnetic (EM) behaviour of the SPARC components under disruption loads and the correlation with the vertical instabilities has emerged, due to the necessity of consistent information for the future operation phase. Starting from a set of analyses to estimate the EM response of the SPARC Vacuum Vessel (VV) during a plasma Vertical Displacement Event (VDE) reported in [2], different scenarios were considered to characterize the disruptions loads. The results of [2] were reproduced and validated using an alternative methodology, which involves the combined use of the MAXFEA code and ANSYS-APDL for EM and structural analyses [3]. The purpose is to use this coupling tool to support the development of synthetic diagnostics system capable of estimating the loads applied to the SPARC vessel and in-vessel components from the limited set of VV displacement measurements and integral plasma observations such as plasma current and effective plasma position. This paper is focused on the collection of different predictive VDE scenarios simulated with MAXFEA by exploring a set of characteristic parameters. In addition, studies concerning different growth rate conditions characterizing the SPARC plasma vertical instability were performed through a simplified RZIp model [4]. These analyses are used to prepare a set of predictive VDE simulations as synthetic experiments to generate and collect relevant data for future SPARC synthetic diagnostics.

Keywords

SPARC Tokamak; Disruptions Loads; VDE; MAXFEA Code; Vertical Instability; Synthetic Experiments.

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PS2-116 ABSTRACT-ed0
H. Models and Experiments for FNT

Low-power testing of the EU-DEMO Steam Generator mock-up

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Within the fight against climate change that commits all Europe, nuclear energy is a clean, flexible and low-carbon source that can reduce greenhouse gases emission and limit the world dependence on high-carbon and high-polluting fossil fuels. Nuclear fusion reactors are still far from realization because of the high complexity not only of the machine itself but also of the many auxiliary systems required to correctly and safely operate the plasma. In particular, the pulsed plasma regime foreseen for the reactor normal operations determines atypical pulsed operative regimes, that not only can cause high thermal stresses on the components but can also generate instabilities. R&D activities are therefore required to characterize the components behavior not only in the fusion extremely severe conditions of pressure and temperatures but also during this sudden power transitions. ENEA, as part of the EUROfusion consortium, has recently planned the construction of STEAM, an experimental facility envisaged for the qualification of the DEMO Steam Generator (SG) during the pulse-dwell-pulse transitions integrated into the W-HYDRA platform. A dedicated experimental campaign will reproduce the minimum operative power phase associated with the only material activation (dwell), the full power phase of the pulse and the transient phase where the transition from pulse to dwell operation occurs. This paper aims at supporting the low power phase characterization by realizing a set of simulations with the RELAP5/Mod3.3 code, investigating the thermal-hydraulic behavior of the system at 1%, 5% and 10% of the nominal power, hence in very challenging conditions which are far from the fission standard practice. The simulation outcomes provide important feedback for the regulation strategy to be adopted for the SG test section, as well as for the correct operation of the rest of the loop.

Keywords

DEMO reactor, STEAM facility, DEMO Balance of Plant experimental investigation, Steam Generator (SG), RELAP5 for fusion applications.

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PS2-117 ABSTRACT-262d
H. Models and Experiments for FNT

Validation of the GETTHEM Tritium Transport and Permeation model Against Experimental Data

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A key aspect for the realization of the EU DEMO fusion reactor is the demonstration of tritium balance sustainability, which will be investigated also during the Test Blanket Module (TBM) programme of ITER. To achieve tritium self-sufficiency, the Breeding Blanket (BB) and Tritium Extraction and Removal (TER) system of DEMO must be designed to minimize tritium inventory. Inview of this, both DEMO design and TBM programme require reliable modelling to prepare experiments and study different concepts. One of the candidate designs for DEMO BB is the Water Cooled Lithium-Lead (WCLL), adopting flowing PbLi as a breeder material. To close the fuel cycle, tritium carried within the PbLi flow must be extracted in the TEU (Tritium Extraction Unit); one of the proposed technologies is the Permeator Against Vacuum (PAV), which is based on the phenomenon of tritium permeation through a membrane that divides the PbLi flow on one side and vacuum on the other. The aim of this work is to validate the model of the tritium permeation through the membrane in the PAV, involving both transport phenomena in the wall and surface processes. The model has been recently implemented in the GETTHEM system-level code, that already included a model of the WCLL TER assessing the transport and removal of Activated Corrosion Products, and will allow to simulate all the possible operating conditions of the EU DEMO TEU. The model validation has been performed against experimental data coming from tests performed in TRIEX-II facility at ENEA Brasimone Research Center, where TEU technologies are investigated.

The PbLi loop model has been adapted and simulated to suitably mimic the components and operative conditions of the TRIEX-II PbLi loop, and results obtained by the code are compared with data obtained during TRIEX-II experimental campaign on the PAV mock-up.

Keywords

EU DEMO, PbLi, modelling, permeator against vacuum, TRIEX-II, validation.

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PS2-118 ABSTRACT-2696
H. Models and Experiments for FNT

Design of the ENEA Water Loop facility in support of the design of the DEMO Water Cooled Lead Lithium Breeding Blanket

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The design of the DEMO Breeding Blanket (BB) has recently entered its conceptual design phase and the Water-Cooled Lead-Lithium (WCLL) is one of the two candidates to become the driver blanket of the machine. With the aim of investigating and assessing different design aspects of the WCLL BB, the Water-thermal-HYDRAulic (W-HYDRA) experimental platform is being designed and built at the ENEA Brasimone Research Centre.

The W-HYDRA platform consists of three different experimental facilities aimed at the investigation of different WCLL BB features: the Water Loop, focused on the investigation of both BB and ITER Test Blanket Module (TBM) thermal-hydraulic aspects and mock-up testing; the LIFUS5/Mod4, aimed at the study of the PbLi/water interaction; STEAM, focused on the study of the once-through steam generator planned to be used for the DEMO WCLL Balance of Plant.

The Water Loop is a high pressure/high temperature loop operating in PWR conditions (15.5 MPa and 295°C-328°C). Its layout and architecture are based on that of the ITER Water Cooling System to have an integral test facility for testing the performances of both specific components and the overall circuit, testing procedures and acquiring significant data for the validation of numerical codes and models for system safety analysis. It is equipped with an Electron Beam gun and a Vacuum Chamber to reproduce the loading conditions expected for both t TBM and BB mock-ups.

The present work details on the main rationale behind the Water Loop design together with a set of thermal-hydraulic and thermo-mechanical calculations performed to support its design.

Keywords

Water Loop, design, WCLL, DEMO, Breeding Blanket.

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PS2-119 ABSTRACT-2940
H. Models and Experiments for FNT

Thermal-hydraulics characterization of the Steam Generator mock-up during operational transients in STEAM facility in support of the design of the DEMO WCLL BoP

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The design of the European DEMO is progressing towards its Conceptual phase foreseen at the end of 2027 and the Water Cooled Lithium Lead Breeding Blanket (WCLL BB) is one of the two concepts candidate to become the driver blanket of the machine. With the aim of investigating and assessing different water and lithium-lead technologies applied to the WCLL BB and Balance of Plant (BoP) systems and components, the Water-thermal-HYDRAulic (W-HYDRA) experimental platform is being designed and built at the ENEA Brasimone Research Centre. Among the facilities constituting the new W-HYDRA multipurpose infrastructure, STEAM is going to experimentally investigate the DEMO WCLL BoP thermal-hydraulics, focusing on the Steam Generator (SG) of the Primary Heat Transfer Systems (PHTS), in order to qualify its performances and suitability under its unconventional operation. The paper aims at supporting the thermal-hydraulic characterization of the Steam Generator mock-up during the sudden power variations typical of a pulsed fusion reactor. The analyzed selected scenario is the operational transient dwell-pulse-dwell, which determines high thermal cycling and correspondent high thermo-mechanical stresses on the primary side components. Two control logics with their relative drawbacks have been analyzed on the RELAP5/Mod3.3 1-D model, the first regulating the primary side average temperature, the second monitoring the minimum one. The comparison of the two systems has been performed and preliminary feedbacks on the control strategy to be adopted have been provided.

Keywords

STEAM, Steam Generator, WCLL, DEMO, Balance of Plant.

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PS2-122 ABSTRACT-38bd
H. Models and Experiments for FNT

A Tungsten-wall Sputtering Model for the Plasma Start-up Simulation in Tokamaks

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Tungsten (W) is the most probable material for the plasma-facing components of fusion reactors due to the excellent thermal and physical properties. A W-wall sputtering model has been set up for simulating the plasma start-up with the 0D simulation code DYON, adopting the revised Bohdansky formula to calculate the sputtering yield. The start-up was found to fail with the formula used in calculating the energy of ions impacting the ITER-like wall of JET, which is in contradiction to the experimental experiences from the tokamaks having W walls such as AUG, WEST, EAST, etc. Modification was made to the formula for the wall-impact energy in a way to reflect that sputtering occurs near the plasma edge and the temperature at the edge actually matters in sputtering. The edge temperature was assumed to decay with rising of the plasma current and finally reach room temperature at the end of the plasma burn-through phase. Using the new model, predictive simulations were tried for KSTAR with an Ohmic plasma start-up scenario and two types of W walls; a full W wall and a 5%-carbon (C)-mixed W wall. The full W wall was found considerably preferable to the C-mixed W wall from the start-up performance point of view, which could be attributed to influx of the sputtered-out W impurity by the ions originating from the C limiters. The model will be verified when the KSTAR Thomson scattering diagnostic system is enabled to measure the edge electron temperature during plasma start-up.

Keywords

Tungsten wall, Sputtering model, Plasma start-up, Tokamak.

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PS2-123 ABSTRACT-4c11
H. Models and Experiments for FNT

Technology trade-off for remote lithium jet thickness monitoring system

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A lithium jet thickness monitoring system is being developed as part of the DONES-FLUX project (Misiones CDTI MIP-20221017). The system will be designed to perform live 3D measurements with a depth resolution of 0.1mm over an area of a few 100s of cm2 of IFMIF-DONES' flowing lithium curtain. It will be used as part of the active control of the lithium target flow parameters and the interlock system of IFMIF-DONES. The technology needs to be compatible with applicable environmental conditions of radiation, vacuum and temperature, and with the applicable geometrical restrictions of IFMIF-DONES. In these early stages of design, a technology trade-off for this measurement system is presented which will be used to guide the design up to TRL4 by the end of 2024.

Keywords

DONES, lithium jet, remote measurement system.

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PS2-124 ABSTRACT-4062
H. Models and Experiments for FNT

Development of liquid metal key technology for advanced nuclear system

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Liquid metal including Lead bismuth eutectic (LBE), lead-lithium and lithium is one of the most potential candidate coolant materials for advanced nuclear reactor system such as the fusion reactor and Lead cooled reactor, due to its attractive nuclear, thermos-physical and chemical properties. To promote the successful application of nuclear energy, detailed studies on key issues of the lead bismuth alloy, lead-lithium and lithium such as thermal hydraulics, corrosion behavior and components performance test were performed.

To provide an experimental platform for non-nuclear key technologies investigation and components performance test for advanced nuclear system, a series of liquid metal loops were designed and developed under the research program in FDS Consortium. It was constructed in 2021, which consists of the forced convection thermal-hydraulic loop and material loop. The thermal-hydraulic loop was built to study the R&D of liquid metal technology, such as the liquid metal heat transfer, flow resistance characteristics and measurement technology etc. The high temperature material loop was designed to investigate the material corrosion with LBE or lithium. The test section is made up of refractory alloy and the operating temperature of the hot leg is 1250℃. And these liquid metal loops can also provide an integrated test platform with international advanced level for engineering verification and basic research of liquid metal cooled reactor technology. Up to now, a lot of experiments have been performed to investigate the liquid metal technologies, such as corrosion behaviors of structural materials under at 700℃ and the reliability tests of key equipment including heat exchange, pump, electromagnetic flowmeter and oxygen pump. The results will support the R&D of the key techniques and the engineering design for advanced nuclear system.

Keywords

Nuclear reactor system, Lead-Lithium, Lead bismuth alloy, Lithium, Loop.

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PS2-125 ABSTRACT-47d6
H. Models and Experiments for FNT

Nuclear irradiation rig design for the Irradiation of ITER prototype bolometers in SCK-CEN BR2 reactor

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Bolometers are sensors designed to measure the power of incident electromagnetic radiation, which causes a rise in temperature in the sensor that is detected with a temperature-dependent resistance. They are an important diagnostic in ITER to detect the total power radiated by the plasma. A number of prototype bolometer sensors need to be tested in a nuclear (neutron) environment up to 0.1 dpa, which is only possible in a fission research reactor. The test includes on-line monitoring of the electrical properties and periodic in-situ calibrations during the irradiation. In order to introduce the bolometers in SCK CEN's BR2 reactor, a dedicated rig has been developed.

Bolometers must be integrated into a rig that ensures proper thermal and dose conditions during testing while monitoring. The design of the nuclear needle in which they are contained has been mainly driven by fitting, routing and properly sealing all the required cables while allowing a homogenous temperature field for the bolometers' critical parts. This needle contains a pressurized helium capsule consisting in a central aluminum piece in which four 4-channel bolometer sensors are placed. Temperature resulting from gamma heating is mainly controlled by the conduction through the exterior via a finely machined 0.4 mm gap. In addition, 6 heaters have been placed inside the capsule to create temperature excursions, compensate for possible deviations of the reactor cycle and thermal gradients due to the different materials of the bolometers' holders. The main novelties of these tests are the number of prototype sensors that will be irradiated at the same time, the number that will be electrically characterized during irradiation, the types of prototype sensors and the level of temperature control. Proposed design has been able to overcome the challenges and successfully integrate the bolometer-sensors into the nuclear reactor for online monitoring during irradiation.

Keywords

ITER, bolometers, online measurements, design, irradiation, reactor, SCK-CEN, temperature control, dose.

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PS2-126 ABSTRACT-4a47
H. Models and Experiments for FNT

Combined analysis of laser interferometer and microwave reflectometer for improved density measurement on HL-2A tokamak

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In order to acquire a consistent and accurate electron density profile for HL-2A tokamak, a combined analysis of the laser interferometer and microwave reflectometer based on the mapping of density profiles to a common equilibrium magnetic coordinate has been proposed. In this approach, the edge density profile measured locally by microwave reflectometer is taken as a supplementary boundary condition in the procedure of inverting line-integrated data from interferometer diagnostic. The density profile inside that boundary is reconstructed by a Gaussian Process tomography (GPT) method within the Bayesian framework, by which the uncertainty of the result can be quantified via the confidence interval of a posterior probability. In the simulation benchmark, two representative cases of density distribution with different profile shapes, e.g. peaked and flat, are tested and the factors that may affect the accuracy of the results are investigated, providing a useful reference for practical applications. The application of this approach to actual data proves its ability to resolve the dynamic evolution of density profiles. Specifically, the effect of Electron Cyclotron Resonance Heating (ECRH) on electron particle transport observed in HL-2A experiments has been confirmed. The combined analysis approach developed is expected to provide a robust means to study the confinement performance and particle transport on HL-2A, with the focus on the core region.

Keywords

Combined analysis, Bayesian probability theory, Tomography method.

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PS2-127 ABSTRACT-c9ce
I. Repair and Maintenance

An obstacle avoidance path planning algorithm to simulate hyper redundant manipulators for tokamaks maintenance

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Due to their high flexibility and manoeuvrability, hyper redundant manipulators result to be the best solution for remote inspection and maintenance tasks in narrow spaces, as tokamaks. During the iterative design process of such manipulators, virtual simulations are conducted to verify if the concept design is coherent with the ideated strategy, in terms of reachability and collision avoidance. For this reason, simulations are programmed offline, with an Inverse Kinematic (IK) approach, aiming to find (if it exists) at least one feasible path for the manipulator, in compliance with a series of geometric constraints. The selection of the IK algorithm for simulations is still crucial. Currently, path planning of hyper redundant manipulators is an important area of research, and a significant effort must be put into the improvement of the efficiency of the algorithm's optimization criterions. In light of this, the present work proposes an obstacle avoidance path planning algorithm for virtual simulations of hyper-redundant manipulators, with customizable optimization criterion. For test purposes, the effectiveness of the proposed algorithm has been tested by simulating the Remote Maintenance (RM) tasks conducted by the HyRMan: the Hyper Redundant Manipulator developed for the Divertor Tokamak Test (DTT) project. The algorithm has been employed to simulate some critical handling tasks of the First Wall (FW) modules, with a specific optimization criterion as example of its potentialities.

Keywords

Inverse Kinematics, Path Planning, Collision Avoidance, Hyper-Redundant Manipulator, Remote Maintenance, Nuclear Fusion, DTT.

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PS2-128 ABSTRACT-a1cd
I. Repair and Maintenance

Interactive simulations for logistics and maintenance of DONES

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Virtual reality simulations of the logistics and maintenance processes have proven useful for identifying potential design issues as well as planning operations during an early design phase of the facility. But VR simulations can also be used to deeply explore the feasibility of these procedures in a more interactive manner, so that we can identify risks and study different strategies to assist the operator during the procedure.

To create a simulation, we begin with a preparatory phase where essential information about the procedures and CAD models is compiled into a comprehensive document (VTD). Once the VTD is agreed, we implement an animated sequence for our system (automatic simulation). Then, we implement the interactive version of the virtual environment, where the different equipment are controlled by the operator. Finally, we analyse the data generated by both simulations. In parallel, we implement different tools that can be applied to multiple simulations, such as a visualization system, a custom camera configuration, an user-machine interface to control every piece of equipment and monitoring/measuring tools inside the virtual environment and detection systems for clashes.

We have used this technology to develop two simulations for the installation process of the SRF Linac modules in DONES. In one simulation, the movement of the equipment is controlled by the own simulator in an automatic way, while in the second one the operator is controlling the movements. In the automatic simulation the user can reproduce the different procedures as they are planned, while in the interactive one an operator controls the movements of the moving cranes, grab and release equipment, move some platforms, etc. This interactive simulation allows us to assess the difficulty of the tasks to be performed, study the best user interfaces for the operator and the most convenient configuration of cameras to prevent risk of clashes.

Keywords

Virtual Reality, DONES, logistics, maintenance, simulation.

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PS2-129 ABSTRACT-2ecf
I. Repair and Maintenance

Material Flow Planning in Fusion Test Facilities

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Fusion test facilities are large, complex buildings used to qualify materials for use in fusion power plants. To ensure a smooth test environment, the machinery and equipment in these facilities must be regularly maintained and efficiently replaced in the event of failure. This results in the need to move large and heavy machinery and equipment through the narrow corridors, doors, hatches and shipping bays of such test facilities. Using the DEMO oriented neutron source (DONES) as an example, we will show the challenges of material flow planning in such facilities, how these challenges can be overcome, and how efficient and safe material flow planning can be organised. A methodology is presented on how the different steps of material flow planning can be performed in such facilities and how uncertainties affect this planning. The main challenges are how to collect data on machinery and equipment and the layout of buildings, how to translate this data into transport devices and routes, how to provide detailed transport instructions and how to ensure the feasibility of the material flow. A multi-stage solution to all these challenges is proposed and applied at DONES. This solution includes data collection methods, the development of a modular transport system, a universal model for transport processes in intralogistics, and a simulation to quickly and efficiently check for collisions between transport devices and machinery and equipment and the building itself. A key finding is that automation of this process is highly desirable, as uncertainties lead to many changes in data, requiring updated material flow planning.

Keywords

Material Flow Planning, Transportation, Transfer Device, Clash Analysis.

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PS2-130 ABSTRACT-310d

I. Repair and Maintenance

Remote Maintenance Study for the DEMO

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Basic concept of the DEMO fusion reactor has been endorsed by the Korea government on February 2023. Four of top level requirements and four of major design have been selected. Technical specification details will be managed by DEMO design Task Force which will be built in KFE with the experts from industries, universities, and institutes.

Accordingly DEMO conceptual design activities including machine assembly and installation, remote maintenance, and system integration will be accelerated. Remote maintenance of breeding blanket and divertor are crucial for the reactor operation and the design shall be performed in parallel with the components design and the building architecture. Remote maintenance study for the DEMO has been launched in 2022 and the strategy aims at the high plant availability through efficient maintenance operations. Based on the R&D plan, three studies, DEMO remote maintenance concept, multi-functional robot arm & manipulator, and small in-bore laser cutting & welding device, were selected. Staged approach was applied on the studies. Visual inspection of the damaged in-vessel components (IVC) and picking out of the damaged small part inside of vacuum vessel during the shot-term maintenance are the first functions defined on the robot arm system. Multi functions like cutting, bolting, welding, grinding, and dust collection and high radiation environment will be considered on the second phase. Cutting and welding of the IVC pipe needs to be performed using the in-bore laser cutting and welding device operated remotely and the reliability shall be demonstrated with mock-up. In this study, DEMO remote maintenance concepts including the scenario, requirements, conceptual design and the progress on the R&D are introduced.

Keywords

DEMO, remote maintenance, robot.

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PS2-131 ABSTRACT-31dd
I. Repair and Maintenance

Maintenance and optimization of the TCV power supply system

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The TCV tokamak is powered by a flywheel generator to supply the magnetic coils and the auxiliary heating systems. The generator has just undergone its fourth major overhaul to make it ready for the next ten years, after more than thirty years of almost trouble-free operation.

In the context of the energy crisis in Europe, we took advantage of the generator outage to evaluate the power consumption of the TCV power supply system and implement some changes to increase the energy efficiency.

After a brief description of the generator overhaul, this article presents the operational mode of the TCV tokamak, focusing on the losses in the power flow from the electrical network to the tokamak coils. Then, the changes implemented in the generator plant are discussed in terms of reducing the energy cost and improving the availability of the TCV experiment.

This work was supported in part by the Swiss National Science Foundation.

Keywords

Power supply, flywheel generator, overhaul, power losses, energy reduction, tokamak operation.

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PS2-132 ABSTRACT-3635
I. Repair and Maintenance

The Electro-hydraulic Mix-drive Robotic Platform Used for General Technology Research of the Remote Handling in Tokamak Devices

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The remote handling of Tokamak requires many general functions, such as manipulating the cantilever arm, screwing and unscrewing bolts, and cutting metal pipes in a toric vessel. This paper aims to describe a robotic platform for general technology research. The platform consists of radiation tolerant materials and joints, a long cantilever arm, a rapid control prototype, a vacuum vessel model, a bolts screwing tool, cutting tools, and virtual reality software. The design of this platform is partly presented: 10 degrees of freedom are chosen, the length of articulated arm reaches 3.15m, the load capacity of arm's end is 10kg, the radiation tolerance of an electro-hydraulic joint reaches to a gamma dose of 4*10^6Gy. The motion planning was tested in a vacuum vessel model. The primary process was studied with the end effectors for functional researches. Screwing and unscrewing bolts was successfully realized without the arm, but pipe cutting from outside still displayed some force issues, and the laser cutting tool for inner pipe working space brought about waste contamination problems, which is to be improved. This remote handling platform is the first visible success for SWIP to struggle for unmanned maintenance for the fusion devices.

Keywords

Remote handling,Cantilever arm,Articulated,Electro-hydraulic,radiation tolerance,Robot platform,Tokamak.

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PS2-133 ABSTRACT-d583
I. Repair and Maintenance

Linear IFMIF Prototype Accelerator (LIPAc) Radio Frequency Quadrupole's (RFQ) RF couplers enhancement towards CW operation at nominal voltaje

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Under the Broader Approach (BA) agreement the Accelerator Facility validation activities aim at demonstrating the acceleration of 125 mA D+ beam up to 9 MeV. This is the main goal of the Linear IFMIF Prototype Accelerator (LIPAc) under installation and commissioning in Rokkasho [1] [2].

The LIPAc commissioning by accelerating the beam through the entire line is currently on-going at Rokkasho Fusion Institute [3] [4] [5]. Other than the beam commissioning, in 2021 and 2022 extensive experimental campaigns have been carried out on the Radio Frequency Power System (RFPS)-RFQ system. The RFPS-RFQ is composed of 8 RF-RFQ tetrode based chains injecting RF power into the RFQ. The target of the RFQ conditioning campaigns is to reach CW operation at nominal vane-voltage of 132kV. The RFPS could achieve CW injection in the RFQ at 105kV at the end of 2021.

During the RFQ conditioning campaigns in 2022 conditioning was stopped because of abnormal increase of vacuum pressure due to a leak from one O-Ring of the couplers. RF power is fed in RFQ by 8 couplers which are equipped with a ceramic window (vacuum barrier) using O-ring for sealing purposes. Extensive simulation and design studies have been carried out and an upgraded design of the parts near the RF window of RFQ coupler has been proposed and validated for implementation.

This paper will focus on the RFQ couplers technical issue, analysis of the design upgrade, implementation of the upgraded solution and perspective for future improvements of the RFQ couplers designs in view of FNS and DONES application.

Keywords

LIPAc, IFMIF, Accelerator, RFQ couplers.

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PS2-134 ABSTRACT-3af3
I. Repair and Maintenance

Research on gas-liquid two-phase counter-current flow limitation Mechanism for Fusion Reactor Airflow Drying Behavior

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The occurrence of gas-liquid two-phase counter-current flow limitation (CCFL) process in the airflow drying of fusion reactor has a great impact on the removal of tritium coolant in the coolant channel, and thus affects the operation safety of fusion reactor. Based on the complex pipeline structure of the EAST primary heat transfer system, this paper deeply explores the mechanism and characteristics of the CCFL phenomenon in the coolant channel under airflow drying, develops a CCFL correction model matching the complex coolant channel structure based on RELAP-5, and verifies the reliability of the application of this model, combined with the characteristics of the pipeline structure of the ITER primary heat transfer system, reuses RELAP-5 program to calculate the pressure in the coolant channel on the basis of the CCFL correction model The distribution of thermal hydraulic parameters, such as flow rate, is used to evaluate the impact of CCFL physical mechanism on the drying efficiency of tritium contained coolant in the ITER cooling channel, which provides a scientific basis for the safe operation of ITER, and also provides a theoretical basis for the design and construction of the future fusion reactor drying system.

Keywords

Fusion reactor; Airflow drying; Gas-liquid two-phase counter-current flow limitation; Physical mechanism.

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PS2-136 ABSTRACT-20f7

J. Burning Plasma Control and Operation

Design of a Cherenkov gamma-ray counter for fusion power measurements in ITER

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Thermonuclear fusion is attractive as it provides a controllable, carbon-free energy source devoided of the risks associated with fission power plants. Moreover, deuterium and tritium, the isotopes needed to fuel the reaction, would be sufficient for thousands of years. A parameter that will always be measured in a reactor is fusion power. The standard technique for measuring the fusion power relies on the counting of the neutrons produced by the de-excitation of the 5He* nucleus produced in the fusion reaction (D + T → 5He* → α + n). An alternative technique is based on the counting of 17 MeV gamma-rays emitted when the 5He* de-excites to the ground state (D + T → 5He* → γ + 5He) whit a probability of about 10-5. In this work, we present a Cherenkov-based gamma-ray counter optimized to provide fusion power measurements every 0.1 s with a 1% counting error on ITER. The diagnostic will be installed with a tangential line of sight on ITER equatorial port plug and able to work in the presence of scattered neutron fluxes of 1010 n/s/cm2. The detector relies on the dependence of the Cherenkov threshold on the particle mass and energy to be insensitive to neutrons and neutron-induced particles impinging on the detector. We present the results of a set of Genat4 simulations confirming that the detector is able to measure the signal produced by the fusion gamma rays when operating in a much more intense scattered neutron background.

Keywords

Cherenkov Detector, Fusion Power, Gamma ray.

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PS2-137 ABSTRACT-57a6
J. Burning Plasma Control and Operation

Design of the In-Vessel Componens of ITER In-Vessel Neutron Flux Monitor Equipped with Microfission Chambers to Withstand High Thermal Load

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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, especially high thermal loads such as nuclear heating and the stray load of Electron Cyclotron Heating (ECH). The MI cables and exhaust pipes, which are part of the in-vessel components of the MFC, are secured to the vacuum vessel (VV) by bolting them to bosses welded to the VV using clamps. Therefore, those components, heated by the high thermal load, need to be cooled only by thermal contact with bosses.

In this study, the design of the in-vessel components was developed to optimize their contact thermal conductance. MI cables and exhaust pipes, both surfaces made of stainless steel, are coated with copper of appropriate thickness to avoid absorption of ECH stray power and to improve heat transfer. In addition, copper interpolation material with appropriate shape and thickness is inserted between the clamp and the MI cable/exhaust pipe to improve the contact thermal conductance of these components.

Thermal stress analysis was performed on this design. The results show the structural integrity of the in-vessel components even under high thermal load conditions (For example, the nuclear heating rate of 1.4 MW/m^3 and ECH stray power of 80 W/m^2). Further, the experiment to evaluate the contact thermal conductance was conducted using prototypes of in-vessel components. As a result, it was demonstrated that the evaluated contact thermal conductance was sufficient to maintain the integrity of the in-vessel components.

As described above, the design of the in-vessel components of the MFC, which can withstand high thermal loads, has been optimized.

Keywords

ITER, In-vessel Components, Neutron diagnostics, Fission chamber, copper coating.

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PS2-138 ABSTRACT-7583
J. Burning Plasma Control and Operation

Experimental Environment for Testing the Shattered Pellet Injection of KSTAR

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The dual shattered pellet injection (SPI) systems installed at the KSTAR (Korean Superconducting Tokamak Advanced Research) have been operational since 2019 to conduct disruption mitigation studies in support of ITER DMS (Disruption Mitigation System) design activities. The system has the advantage of firing pellets simultaneously from the two identical pellet injectors installed at 180-degree opposite positions on KSTAR. Each injector is capable of fabricating and firing three pellets of different sizes (diameter, length). The characteristics of the initial pellet (geometry, velocity, etc.) can be verified using a microwave cavity (MWC) system and an imaging optical pellet diagnostics (OPD) system.

Each SPI system must achieve specific conditions related to pellet fabrication and firing prior to plasma experiments. We realized that we didn't have the sufficient time and conditions to do off0line testing until 2022, so we have developed a suitable solution. First, we have to separate the SPI system from the KSTAR vacuum vessel and build a separate test chamber. Inside the new test chamber, we have to install the same shatter tube used in the current experiments and equip it with measuring devices to observe the fragmented pellets. We also have to switch the fabrication process of the pellets from manual to semi-automatic using a feedback pressure controller on the barrel pressure. In this presentation, we will describe the new experimental environment for independent SPI testing and the initial results.

Keywords

Disruption, shattered pellet, dual injectors, KSTAR.

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PS2-139 ABSTRACT-1bc9
K. Inertial Confinement Fusion Studies and Technologies

Design of High Power Supply Protection System for CRAFT Sensor Test Platform

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In order to develop equipment for measuring current up to 100 kA , a test platform for high power supply of Comprehensive Research Facility for Fusion Technology (CRAFT) sensor is established. there are many diverse and serious faults in practical engineering applications of high power power supply systems, so it is necessary to pay attention to the analysis and design protection systems. The inverter power supply of the sensor test platform adopts the cascaded H-bridge topology structure. According to the structural characteristics of the cascaded H-bridge inverter system, the whole topology structure is divided into three parts: AC input converter, DC bus and AC output inverter, and the possible faults of each part are analyzed; A variety of hardware protection and integrated protection system based on DSP software protection are designed. The simulation and experimental results show that the power supply protection system can ensure the safe and reliable operation of the power supply in case of failure, and effectively protect the components of the power supply system from damage, thus verifying the feasibility and effectiveness of the design scheme.

Keywords

CRAFT power supply, H-bridge cascade, Protection system, System fault, protection circuit.

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PS2-140 ABSTRACT-7fc9
K. Inertial Confinement Fusion Studies and Technologies

Experimental studies of laser-driven Ions and extreme plasmas Interaction

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"*Experimental studies of laser-driven Ions and extreme plasmas Interaction*"

The study of Ion Stopping Power close to the Brag peak in Warm Dense Matter is an open field of investigation strongly related to Inertial Confinement Fusion due to the importance of the ion energy deposition in the burning phase of nuclear fusion. Theoretical models give different and controversy predictions in this regime and the experimental results are still rare in the field. In addition, proton stopping power close to the Bragg peak require low energy (below 1 MeV) proton beams with a very thin targets (~ 1 um). Such extreme conditions together cannot be satisfied if using long pulse (~ ns) lasers such as the one in the large energy laser facilities because of the short lifetime of the heated target compared to the duration of the heater. The limited lifetime (~ hundreds of ps) of the Warm Dense Matter Target reduce drastically the free time of investigation forcing to reduce both the time of the probe and the time of measurement. High Power and ultra-short (from ps down to fs) have opened the possibility to generate very short (few ps) proton beams with which is possible to probe the WDM that can also be generated with similar duration laser pulses.

Here we report on:

1. Recent an experimental campaign performed at the 30 fs 200 TW system VEGA 2 at the Centro de Laseres Pulsados and results are published [Malko2022]
2. An experimental plan to investigate more regions of the ion stopping power in plasma with the aim to benchmark theoretical predictions

[Malko2022] S Malko, et al, "*Proton stopping measurements at low velocity in warm dense carbon*" Nature Communications 13 (1), 1-12

Keywords

Laser Fusion, ion stopping power, HED, WDM.

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PS2-141 ABSTRACT-28a5
K. Inertial Confinement Fusion Studies and Technologies

Novel analysis method for BNCT pharmacokinetic evaluation by use of a neutron source based on Inertial electrostatic confinement fusion and particle detector

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Boron neutron capture therapy (BNCT) is one of the medical applications of neutrons. The ^{10}B atoms located in cancer cells capture thermal neutrons and release α-particles with ^{7}Li nuclei associated with gamma rays based on the following nuclear reactions:

$^{10}B + n \rightarrow \alpha$ (1.47 MeV) + ^{7}Li (0.84 MeV) + γ (0.48 MeV) (94%)

$\rightarrow \alpha$ (1.78 MeV) + ^{7}Li (1.01 MeV) (06%)

The conventional analysis method for BNCT is detecting gamma rays through the prompt gamma ray analysis (PGA) technique, which in principle, requires pure thermal neutron flux and low background gamma-ray environment. The high neutron flux could be generated using nuclear reactors or powerful accelerator-based neutron sources. Both sources generate massive background radiation that requires sophisticated shielding and data processing for accurate results.

Using nuclear reactors and accelerator-based neutron sources for BNCT becomes complex and remarkable, while alternative methods to accelerate the development of BNCT research and treatment aiming for higher tumour selectivity and accumulation are crucial issues. Establishing a cancer treatment method with an extremely low burden that can be applied to elderly patients is an urgent issue in realising a society of health and longevity. This work aims to develop a novel quantitative analysis method of ^{10}B for BNCT. The proposed method is based on detecting a-particles generated from the nuclear reaction between ^{10}B and thermal neutron, which differs from the conventional method that investigates gamma rays. In the proposed method, the thermal neutrons will be generated from a compact neutron source based on the inertial electrostatic confinement fusion (IECF), while the α-particles will be detected using the centrifugally tensioned metastable fluid detector (C-TMFD) that is entirely blind to gamma rays. The proof of principle, experimental layout, conditions, and preliminary results will be discussed in the meeting.

Keywords

BNCT, PGA, Compact neutron source, CTMFD.

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PS2-144 ABSTRACT-7054
K. Inertial Confinement Fusion Studies and Technologies

Spectroscopic characterization of core conditions in highly magnetized cylindrical implosions

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The use of external magnetic fields in inertial confinement fusion has been identified as a promising way to assist ignition, by improving the hot-spot performance, reducing thermal losses and enabling higher fusion yields. To facilitate the investigation of the magnetic-field compression mechanism, a cylindrical geometry is particularly appropriate. Here, we present the application of Ar K-shell spectroscopy to characterize the core conditions in magnetized cylindrical implosion experiments performed at the OMEGA laser facility. Targets filled with Ar-doped deuterium were symmetrically imploded by a 40-beam, 14.5 kJ, 1.5 ns laser drive. According to 2D numerical simulations using the extended-MHD code Gorgon, the seed B-field of ~30 T delivered by the MIFEDS pulsed-power device is compressed with the target up to ~10 kT, which should be strong enough to alter the hydrodynamic evolution and therefore the conditions of the compressed core. Recorded Ar K-shell spectra show highly reproducible differences in the line ratios, indicative of a higher temperature in the magnetized implosions compared to the non-magnetized case. Forward-directed simulations of the spectra, obtained by post-processing the MHD output using detailed atomic-kinetics and Stark-broadened line shapes, reproduce reasonably well the observations. However, extracting representative values of core conditions from the time- and space-integrated emission spectrum is difficult, particularly in the magnetized case where large gradients in both temperature and density are expected according to the MHD simulations. In this regard, we discuss the use of advanced minimization algorithms that show potential for a quantitative spectroscopic analysis. The methodology allows to extract a coarse-grained representation of the core conditions radial profile at stagnation and supports the formation of a hotter central region in the magnetized scenario. Additionally, we assess a dual dopant (Ar and Kr) spectroscopy technique to achieve an effective spatial resolution and a more robust diagnostic of the magnetized compressed core.

Keywords

X-ray spectroscopy, cylindrical implosions, magnetized inertial confinement fusion.

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PS2-145 ABSTRACT-737a
K. Inertial Confinement Fusion Studies and Technologies

Inadequacy of magnetohydrodynamics to model shock waves in some Inertial Confinement Fusion settings

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Inertial Confinement Fusion Studies features the propagation of shockwaves in collisionless magnetized plasmas. These phenomena are usually described using magnetohydrodynamics (MHD). Yet, MHD entails the same hypothesis than fluid theory with respect to collisions, which raises some doubts on its relevance to collisionless plasmas. We will show how this is especially true in strongly magnetized plasma, where MHD can greatly overestimate the density jump of a shockwave. Numerical simulations will be presented confirming our theory of such processes.

Keywords

ICF, MHD, Shockwaves.

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PS3-2 ABSTRACT-4e23
A. Plasma-Facing High Heat Flux Components

Investigation of Erosion/Deposition by Charge-exchange Neutrals in Far-SOL Region

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In tokamak fusion devices, plasma-facing materials are damaged by charged particles, which hit the surface of the material following the magnetic field lines [1]. However, erosion and deposition of PFCs are observed in a shadowed area, since charge-exchange (CX) neutrals can migrate to a magnetic shadowed area where ions can hardly reach [2-3]. It implies that CX neutrals play an important role in fuel retention and the erosion of wall material. This study aims to investigate the erosion/deposition effects of CX neutrals on the plasma-facing materials, cavity samples with amorphous hydrogenated carbon film (a-C:H film) were exposed to KSTAR plasmas. To distinguish CX neutrals from incident particles, the structure of the cavity sample has a different two-slit opening direction. The cavity sample made of a stainless steel box and there is a deposited film. Deposited films, especially in the so-called amorphous hydrogenated film (a-C:H film), were grown by 13.56MHz RF power glow discharge. The thickness and optical properties of a-C:H film were measured by ellipsometry before and after experiments. A net deposition rate of 0.17 - 0.89 nm/s during Ohmic and H-mode discharges was obtained at KSTAR far-SOL region. The slit opening parallel to the magnetic field line was the toroidal gap, it found that the thickness profile in the toroidal gap was broadening than that in the poloidal gap due to the contribution of both ions and CX neutrals at the condition of H mode plasma.

[1] P.C. Stangeby, The Plasma Boundary of Magnetic Fusion Devices, Institute of Physics Publishing, Bristol, 2000.

[2] S. Krat et al., J. Nucl. Mater. 456 (2015) 106-110[3] Suk-Ho Hong et al., J. Nucl. Mater. 438 (2013) S698-S706

[3] Suk-Ho Hong et al., J. Nucl. Mater. 438 (2013) S698-S706

Keywords

Plasma-facing material, PFC damage, erosion/deposition, Charge-exchange Neutrals.

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PS3-3 ABSTRACT-6340

A. Plasma-Facing High Heat Flux Components

Thermal-hydraulic assessment of the EU DEMO Upper Limiter

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The EU DEMO tokamak foresees the integration of a limiter system composed of four discrete limiters (i.e. upper, outboard midplane, outboard lower, inboard midplane) for wall protection during different transients, either operational (i.e. ramp-up/down) or incidental (i.e. vertical displacement events, VDE). In particular, the Upper Limiter (UL) is designed to protect the wall in case of upward VDEs, and is composed by two subcomponents, i.e. the Plasma-Facing Wall (PFW) and the Shielding Block (SB), which has structural function in addition to shielding the Vacuum Vessel and the magnets from the neutrons. The design of the UL requires integration of different pieces of physics, namely neutronics, thermal-hydraulics, electromagnetism, and thermal-mechanics.

This work presents the thermal-hydraulic assessment of the EU DEMO UL. The analysis is performed with a 3D Computational Fluid Dynamics (CFD) model developed with the STAR-CCM+ commercial software. The SB and PFW are analysed independently under the nuclear heat deposition computed by neutronic calculations, verifying the overall pressure drop in the component, and ensuring that the water coolant distribution is uniform. The layout of the cooling coils in the SB is checked to be sufficient to keep the EUROFER structure within a safe temperature window under the nuclear heating. The temperature distribution within the PFW castellated structure is also assessed with a detailed model. Finally, the thermal coupling between the PFW support structure and the SB is checked.

Keywords

EU DEMO, Upper Limiter, thermal-hydraulics.

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PS3-4 ABSTRACT-64a5
A. Plasma-Facing High Heat Flux Components

Molecular dynamics modelling of the stress effect on diffusion behavior of hydrogen in tungsten

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It has been proved by experiments that the heat load and implantation of hydrogen atoms from plasma of fusion reactors introduce significant stress in plasma facing material, tungsten. However, the stress effect on the diffusion of hydrogen atoms remains unclear. In this work, molecular dynamics simulations combined with the nudged elastic band method were performed to investigate the influence of hydrostatic, uniaxial and shearing loads on the site preference, diffusion path and diffusion barrier of hydrogen atoms in tungsten lattices. The simulated results show that the deformation of lattices under external stress changes the layout of tungsten atoms around interstitial hydrogen atoms, and therefore changes the formation energy. Under hydrostatic and uniaxial loads, the alteration of preferable sites between tetrahedral and octahedral sites was observed. Meanwhile, the variation of formation energy also results in the change of both the diffusion path and the diffusion barrier. However, no monotonic relationship between the diffusion barrier and the strain was obtained. Both compressive and tensile stresses can increase the diffusion barrier. Under uniaxial and shearing loads, diffusion properties of hydrogen atoms differ in directions. For uniaxial compression, it is much easier to diffusion in the loading direction, while for shearing load, it is easier to jump in the direction perpendicular to the shearing plane.

Keywords

Molecular dynamics, diffusion, tungsten, hydrogen, stress effect.

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PS3-6 ABSTRACT-048b
A. Plasma-Facing High Heat Flux Components

Engineering strategy for the European DEMO divertor: challenges and approaches

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Since 2021, conceptual design activities for the European DEMO divertor have been conducted in the framework of EUROfusion Consortium following the preceding pre-conceptual design phase (2014-2020). A major outcome of the pre-conceptual phase was the "Baseline" design validated as the current reference design. This Baseline design is based on a dual cooling circuit concept where the plasma-facing targets and the cassette body are cooled by pressurized water via a separate cooling circuit, respectively. The inlet coolant temperatures of both circuits are relatively low (target: 130 °C, cassette: 180 °C). While ensuring a large heat removal capacity, such a low operational temperature raises a serious concern with regard to the mechanical reliability of the structural materials (target: copper alloy, cassette body: EUROFER97 steel) being embrittled under neutron irradiation up to a substantial damage dose. These contradicting aspects pose a fundamental trade-off issue between the thermal and structural performance.
For risk mitigation, an alternative cooling option is under investigation in the conceptual design phase. In this option, a higher inlet temperature (295 °C) is considered for the steel cassette body so that its structural design is exempted from embrittlement issue. However, the latter option is also afflicted with diverse design issues (e.g. softening of the overheated steel parts, extensive coolant nucleate boiling, etc.), which potentially could be critical.
In this contribution, we present our updated design strategy and new engineering approach. Comparative evaluation of performance is presented between the Baseline and the alternative concept on the basis of comprehensive multi-physics analyses. Focus will be placed on the permissible thermohydraulic design space, impact of each cooling condition on the structural integrity and the technology options pursued for the targets and cassette body. High-level design

requirements such as maintainability and waste reduction are also revisited and addressed in the current design revision process.

Keywords

DEMO, Heat exhaust, Divertor, Cooling, Design, Materials.

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PS3-8 ABSTRACT-710b
A. Plasma-Facing High Heat Flux Components

Experimental investigation on the MHD effects of liquid metal film flow relating to the plasma-facing components

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In nuclear fusion, the liquid metal film plasma-facing component (PFC) is considered to be one of the most promising ways to realize a PFC capable of operating for long periods. However, due to the existence of a magnetic field, the flow status of liquid metal became complex due to the magnetohydrodynamic (MHD) effects. In the present study, by adopting the modeling liquid metal, Galinstan, as a working liquid, we experimentally studied the MHD effects of the liquid metal film flow with considering the influence of the intensity of the magnetic field, the flow rate of liquid metal, the electrical conductivity of side wall and back plate. The laser profile meter accompanied by the high-speed camera was utilized to obtain the thickness of the liquid metal film, the free surface wave and the surface contour. Results show that the surface wave on the liquid metal flow can be smoothed by the transverse magnetic field, and the average film thickness increases greatly with the increases of the magnetic field. The electrical conductivity of the side wall has an important influence on the MHD effects of liquid metal film flow. Meanwhile, a liquid metal hydraulic jump appears in the region near the position of the film generator can be changed by the magnetic field. Moreover, to realize a suitable liquid lithium limiter with a high coverage ratio of flowing lithium on the solid back plate, we designed a limiter with typically designed microgrooves on its surface which can improve the wetting of liquid lithium on the stainless-steel surface. Our experimental results show that 97.32% of the limiter surface has been covered by the flowing lithium, which is the largest one reported. Moreover, once the lithium spreads on the surface, it can last for a long time in different situations.

Keywords

Liquid metal, film flow, magnetohydrodynamics, plasma-facing component.

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PS3-10 ABSTRACT-78cd
A. Plasma-Facing High Heat Flux Components

Design of in-Vessel Mirror of ITER Poloidal Polarimeter

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ITER Poloidal Polarimeter (PoPola) is laser-aided diagnostics that measures the change of polarization of injected far-infrared laser beam (wavelength of 119 μm) in order to identify plasma current profile. PoPola consists of thirteen measurement chords, and each chord uses two in-vessel mirror assemblies for the laser propagation. Diagnostics in-vessel mirrors receive high neutron heating loads and high radiation heat loads because it directly faces the plasma. The PoPola mirror assembly consists of three layers; tungsten mirror, copper-alloy heatsink and stainless-steel mount. The copper-alloy heatsink is actively cooled by stainless-steel pipe. The tungsten mirror is attached to the heatsink with contact pressure of 3.5 MPa and is cooled by thermal conduction to the heatsink. The mirror is not rigidly fixed to the heatsink but is able to slide on the surface of the heatsink because thermal stress in the dissimilar materials during baking needs to be avoided. Loads applicable to in-vessel mirrors are nuclear heat load, radiation heat load, heat load by charge-exchanged neutral particles, water pressure, electro-magnetic force and acceleration load during plasma disruption and earthquake. The structural integrity under these loads was evaluated according to RCC-MRx code. The highest primary load to the mirror was the acceleration load during plasma disruption, and the highest secondary load was the baking temperature (240°C). All design criteria specified in RCC-MRx code are satisfied. In addition, the design of the in-vessel mirrors was developed to facilitate adjustment of mirror angle with accuracy of 0.1°. A prototype mirror was made to confirm manufacturability. Surface deformation owing to the assembly load was 2.2 μm and was within the limit of deformation of 20 μm.

Keywords

ITER, in-vessel mirror, polarimeter.

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PS3-11 ABSTRACT-8736
A. Plasma-Facing High Heat Flux Components

Manufacturing of advanced target mock-ups for the DEMO Divertor

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Within the roadmap of the DEMO reactor design, R&D activities have been promoted for the technological development of Plasma Facing Units. A dedicated task in the WP-DIV deals with the consolidation and verification of the current target concepts envisioned for DEMO, i.e., the ITER-like concept and the back-up concept with tungsten-fiber reinforced copper (Wf-Cu) pipes. A research activity has been undertaken at ENEA to support the back-up concept, finding alternative technological solutions for monoblock-pipe joining in order to reduce the use of materials having high activation and/or degradation under neutron irradiation. Among the brazing alloys tested for the monoblock/Wf-Cu pipe joint, the Gemco commercial alloy has been selected as the most suitable, thanks to its good joining capability and low content of Nickel. Despite the effectiveness of the joining capabilities proven by Gemco with CuCrZr pipes, some issues arise when the monoblocks have to be joined with Wf-Cu pipes. A possible cause could be the low coefficient of thermal expansion of the pipe due to the relevant content of W, that makes difficult recover the gap between the surfaces to be joined (necessary for the assembly) during the brazing process. In order exclude the eventual influence of the joining process parameters, an experimental sensitivity analysis has been carried out by realizing joining tests between W monoblocks and Wf-Cu pipes using Gemco, changing in each test one parameter at a time (i.e., Cu interlayer, monoblock-pipe free gap, thermal cycle, positions of the items in the furnace). Consequently, small mock-ups with four W monoblocks have been manufactured by brazing procedure using Gemco, changing the Cu interlayer thickness of the monoblocks and the and Wf-Cu pipes external surface finishing. The results of the experimental activity are reported through destructive and non-destructive examinations performed on samples and small mock-ups realized.

Keywords

DEMO, Plasma Facing Units, Divertor target mock-ups, tungsten-fiber reinforced copper pipes for PFCs, brazing alloys testing, nondestructive analysis.

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PS3-12 ABSTRACT-8bf5

A. Plasma-Facing High Heat Flux Components

Characteristics of detached/attached plasma formation on ICR heating in a linear divertor plasma simulator TPDsheet-U

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Most divertor simulators cannot reproduce the actual detached plasma in a divertor because the ion temperature of the simulated plasma is low (a few eV) compared to the ion temperature of the divertor plasma in fusion devices (10 eV or higher). Since magnetized sheet plasmas can be efficiently heated by the ion cyclotron resonance (ICR) method [1], the main objective of this study is to investigate the effect of ICR heating on the detached/attached plasma production process. In this paper, the effect of the detached plasma production process on the ion temperature using a linear divertor simulator (TPDsheet-U) is investigated by applying RF power in the ion cyclotron frequency band (1.0-1.3 MHz) to high-density hydrogen sheet plasmas (n_e~10^{19} m^{-3}, T_e~10 eV) in the uniform magnetic field regime [2]. The ion temperature increased from 2 eV to 8 eV as the RF power for heating was increased from 0 W to 400 W at the uniform magnetic field of 0.08 T. Using a Langmuir probe and visible spectroscopy, the formation of the detached plasma was verified by measuring the electron temperature, electron density, and Balmer series emission intensity ratio Hγ/Hα. The stored energy of the heated plasma was measured using a magnetic loop coil, and the ion temperature was approximated using these values. When the ion temperature was increased from 3 eV to 6 eV by applying a resonance frequency ω_{RF}, the electron temperature of the detached plasma was increased from 0.5eV to around 2 eV and the Balmer series emission intensity ratio Hγ/Hα was found to be smaller. Therefore, the electron temperature does not decrease sufficiently, which is thought to suppress the formation of detached plasma.

[1] Y. Ohara, et al., J. Plasma Fusion Res. SERIES, 8 (2009) 888.

[2] N. Okada, et al., Fusion Eng. and Des.192(2023)113596.

Keywords

Detached/attached plasma, ICR heating, linear divertor plasma simulator.

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PS3-14 ABSTRACT-d12e
A. Plasma-Facing High Heat Flux Components

A calorimetric evaluation method for beam targets with IR imaging: implementation for the negative ion source BATMAN Upgrade

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Infrared (IR) imaging of beam targets yields time-resolved high-resolution spatial temperature footprints. Conversion of temperature footprints into power density profiles is not straightforward. A comprehensive calorimetric evaluation method is presented for the beam target installed in the testbed BATMAN Upgrade (BUG). BUG hosts the prototype RF-driven negative ion source for the ITER Neutral Beam Heating System. In nominal conditions, BUG features an array of individual (5 horizontal × 14 vertical) negative ion beamlets with a 20 mm spacing among them. Each individual beamlet is accelerated electrostatically up to a maximum of 45 kV and transports currents in the range from 0.01 to 0.05 A. BUG has two beam targets at different distances from the beam acceleration system. The furthest (2.09 m downstream) is a water-cooled diagnostic cw-calorimeter, which can measure the beam profile with a resolution of 20 × 20 mm² over a surface of 600 (H) × 800 (V) mm². The closest (0.87 m downstream) beam target with submillimetric resolution, consists of a (142 × 376 ×20 mm³) tile of a composite anisotropic material out of carbon fibers oriented in the thickness (1D-CFC). It can only be exposed to beams for limited time, therefore it is retractable. The proposed calorimetric evaluation method of the IR images is based on extensive thermal FEM simulations of the beam target for relevant beam scenarios. It reconstructs the impinging heat flux of the negative ion beams to derive individual beamlet properties under specific conditions. Some error sources are discussed and quantified to assess systematic error estimation. The method is experimentally validated in BUG with the electrical measurements and data from the cw-calorimeter for all beamlets together, showing good agreement among them.

Keywords

Ion Beam optics, 1D-CFC, Thermography, Calorimetry, BATMAN Upgrade.

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PS3-15 ABSTRACT-03a1
A. Plasma-Facing High Heat Flux Components

Wrapping the Sun in a Blanket and putting it in a Box - The Design and Manufacture of Plasma Facing Components

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Jacobs' fusion energy heritage goes back more than four decades. The highlights include supporting the Joint European Torus (JET) at UKAEA; serving as ITER's construction management-as-agent contractor as part of MOMENTUM, and coming right up to date with our work to design and build the UKAEA's Chimera facility for plasma facing components (PFC) testing.

Jacobs has been involved in the design, development, and manufacture of PFCs from the mid-1990s. Our presentation will chart the evolution of PFC design from early development to full scale prototypes of ITER First Wall Panels (FWPs). Mock-Ups to be discussed include:

- **Thermal Fatigue Mock-Up:** Demonstrated beryllium tile bonding on a water-cooled composite Stainless Steel – Copper heatsink
- **Semi-Scale Prototype:** 1/6th section of a First Wall Panel (FWP)
- **Full Scale Prototype:** The first panel manufactured to meet the design and QA requirements of ITERs FWPs
- **Advanced Design Mock-Up (ADMU):** Designed to reduce manufacturing costs of the ITER FWPs and improve manufacturing efficiency

During the journey we will examine the evolution of PFC design before progressing to the development and testing of manufacturing techniques used for the successful production of these components. This includes the evolution of the Hot Isostatic Pressing (HIP) process; used for the joining of heatsinks and the joining of beryllium tiles to plasma facing surfaces; and improvements of the thermal quenching applied to copper-chromium-zirconium (CuCrZr) to ensure heat sink performance.

Finally, we will discuss the transition to tungsten-covered components and our support to UKAEA's STEP Programme. Here, we will outline our concept designs of the breeder blanket and divertor, studies focusing on improvements to the tiling methodology and material joining and conclude with the design, development, and production of the Thermal Barrier Mock-up.

Keywords

Plasma Facing Components, PFC, Hot Isostatic Pressing, HIP, Diffusion Bonding, UKAEA, ITER, F4E, Jacobs, Prototype, Thermal Barrier Limiter, Tungsten, Beryllium, First Wall panel, Quenching, Copper - Chromium -Zirconium, CuCrZr, Heat Sink, STEP, JET, Breeder Blanket, Divertor.

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PS3-16 ABSTRACT-9791
A. Plasma-Facing High Heat Flux Components

Mock-ups fabrication by HRP technology with advanced W-alloy monoblocks for DEMO divertor target

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The current consolidated process for the fabrication of a target divertor for ITER foresees pure tungsten monoblocks as armour material, copper as interlayer joint to pure W, CuCrZr-IG alloy as heat sink material joint to the W-Cu monoblocks by hot radial pressing (HRP).

ENEA developed the HRP technique for manufacturing plasma-facing units of the ITER reactor which consists of W armour monoblocks and a cooling CuCrZr-IG pipe joined together through a pure copper interlayer which acts as a coupling and stress mitigating layer. Tungsten is also the primary candidate armour material for the divertor target in the EU-DEMO demonstration fusion power plant. However, during operation at high temperature pure W is subject to recrystallization which results in a loss of strength and thermal properties, while loss-of-coolant accidents with simultaneous air ingress can generate volatile and radioactive tungsten oxides.

Within the EUROfusion roadmap for the DEMO reactor design, dedicated tasks in the Work Package Materials deal with the development and fabrication upscaling for advanced W based plasma facing materials, with the aim to find the best advanced W-alloy candidate as armour material of the DEMO reactor divertor target.

This paper reports the fabrication of several DEMO-like divertor target mock-ups using W-alloy monoblocks. The interlayer inside the monoblock hole was manufactured by Cu casting method. For each mock-up, Cu-Cast advanced W monoblocks were joined to a CuCrZr-IG pipe by HRP. Both manufacturing processes required optimization of the fabrication parameters according to the different thermo-mechanical and chemical characteristics of each type of W-alloy monoblock. Mock-ups should be tested at high heat flux (HHF) for performance comparison.

For quality control, non-destructive examination by ultrasonic technique (UT) was done on the monoblocks as received and after Cu casting on hole monobloocks. The UT examination was performed also after the HRP process and after HHF testing.

Keywords

DEMO Plasma Facing Components, Divertor target mock-ups, Divertor materials, advanced W-alloy, hot radial pressing, casting, nondestructive analysis, ultrasonic technique.

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PS3-17 ABSTRACT-98cd
A. Plasma-Facing High Heat Flux Components

Development of Divertor Heat Removal Component Using Tungsten-Copper Alloy Bonding with SPS Method

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As a divertor heat removal component of a fusion device, tungsten (W) and chromium-zirconium-copper (CuCrZr) joints were fabricated by the discharge plasma sintering (SPS) method. The thermal stress relaxation layer of W-Cu mixed powder was confirmed to have sufficient density. A test piece was fabricated using the proposed method and installed in the divertor section of a large helical device LHD. The results of the infrared camera during the experiment and the damage after the experiment are also reported.

Keywords

Divertor, Tungsten, Copper Alloy, Spark Plasma Sintering, Heat Load.

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PS3-19 ABSTRACT-7419
B. Blanket Technology

Preliminary Design and Analysis Activities of the Deuteron Accelerator Target for Tritium Breeding Unit Test

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A breeding blanket (BB) is a key component that multiplies neutrons and generates tritium in the fusion demonstration reactor (DEMO). Evaluation of the performance and reliability of the BB is a very important engineering task, and it is going through the ITER TBM program. The Korean DEMO program needs to complement the ITER TBM program to validate the BB, and the test facility is necessary to validate the BB in the continuous wave operation. Based on this need, the Korea Institute of Fusion Energy (KFE) has initiated a pre-conceptual study to verify the long-term performance and reliability of the BB.

This study deals with neutronics and engineering analyses of a solid beryllium (Be) target that generates neutrons through a deuteron accelerator for the fusion neutron source. In the neutronics analysis, the yield and energy spectrum of neutrons generated according to the incident particles and energy were analyzed. As a result, in the case of 40 MeV deuteron, the yield of neutrons with 10~20 MeV similar to the fusion neutron was the highest in the forward direction of 1.08E+15 n/s. In addition, vanadium (V) was selected as the blistering mitigation layer (BML) and the thickness of the Be target and the BML were optimized through neutronics analysis to minimize the blistering. In the engineering analysis, a preliminary conceptual design of the target to cool the nuclear heating caused by the nuclear reaction was performed. To improve the cooling capacity of the target composed of Be-V-CuCrZr with the pressurized water coolant, the design of the target was optimized and its thermal integrity was evaluated. Next, thermal stress was evaluated by coupling the temperature distribution of the target obtained through thermal analysis with structural analysis. Through this, the preliminary design concept of the target that satisfies thermal and mechanical integrity requirements was derived.

Keywords

Target, Accelerator, Fusion Neutron Source, Neutronics Analysis, Engineering Analysis.

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PS3-20 ABSTRACT-789e
B. Blanket Technology

Two-way coupled CFD-DEM investigation on gas and powder flow characteristics of pebble bed of Fusion Blanket

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The breeding blanket, which is core component for the fusion device, plays an important role in tritium breeding and energy conversion. Ceramic particles are usually used as tritium breeders (e.g., Li_4SiO_4, Li_2TiO_3 particle) and neutron multipliers (e.g., Be/Be_{12} particle) inside the blanket, and pebble bed structure is utilized to hold ceramic particles. The helium pressure drop inside the ceramic pebble bed is closely related to the design of the tritium purge structure of the blanket, the flow characteristics of helium and powder directly affect the tritium breeding and extraction efficiency. Hence, there is a great need to promote research on the flow behavior of purge gas and the dynamic behavior of the crushed particles. Based on the two-way coupled Computational Fluid Dynamics and Discrete Element Method, the velocity distribution, pressure distribution and inlet and outlet pressure drop of helium in pebble bed structure under different inlet flow rates were obtained. The velocity and quantity distribution of lithium powder at different moments and inlet flow rates were numerically investigated. The results indicated that flow velocity and pressure drop agree well with Ergun's formula and dimensionless velocity of helium is independent of helium inlet flow rate, the difference in powder velocity between different inlet flow rates is not significant. In the large inlet flow condition, larger powder is more easily blown out of the pebble bed than smaller powder. The results in the paper can provide technical recommendations for fusion ceramic blanket design.

Keywords

Helium Cooled Blanket; Pebble Bed; Computational Fluid Dynamics; Discrete Element Method; Fusion Energy.

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PS3-21 ABSTRACT-78ec
B. Blanket Technology

Weldability of F82H for WCCB TBM Application

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Water-cooled ceramic breeder (WCCB) TBM fabricated by reduced activation ferritic/martensitic steel of F82H is being developed by National Institutes for Quantum and Science and Technology in Japan. Since the welding process shall be applied in WCCB TBM, an assessment of the validity of the design and manufacturability is urgently necessary based on the requirement of French regulations and construction codes.

Different welding methods including Tungsten Inert Gas (TIG) were applied to F82H and its weld properties were reported in the previous studies. Based on the achievements, plates, Φ10 mm with a thickness of 0.75mm pipes of F82H was welded and examined by a series of non-destructive and destructive examinations. French regulation related to nuclear pressure equipment (ESPN) and the construction code of RCC-MRx were applied.

A maximum thickness of about 5 mm is expected for welding based on the WCCB TBM. Post-weld heat treatment with a condition of 720℃＋15℃/－0℃ for 1 hour was conducted. Mechanical properties of the welds were figured out by hardness, tensile, bending, and toughness tests.

Non-destructive testing performed that cold and hot cracking were not observed in welded pieces. In the case of tensile tests, regardless of base metal and weld metal, total elongation performed magnitude higher than that of 14%, also tensile strength of TIG welds was not exceeded 800 MPa, at room temperature and 550℃. In the case of Charpy impact tests at RT and -20℃, base metal, weld metal, and heat affected zone showed higher absorbed energy than the requirement of Level N1 ESPN. As a preliminary evaluation of weldability of F82H, full penetration for nuclear pressure equipment by limited TIG welding process and post-weld heat treatment condition satisfied the requirements of ESPN and RCC-MRx.

Keywords

F82H, TIG welding, Mechanical property.

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PS3-22 ABSTRACT-7d51
B. Blanket Technology

Design and Analysis of the HCCB Breeding Blanket with Casing Structures for DEMO reactor

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Blanket is one of the core components of the DEMO reactor, and the basic functions are tritium breeding, energy extraction and nuclear shielding. Focus on higher Tritium Breeding Ratio (TBR), a newly helium cooling ceramic breeder (HCCB) blanket with casing structure was proposed for DEMO reactor. The blanket is characterized by the using of two concentric pips, the inner pipe center is lithium silicate, the annular gap between the concentric pipes is helium cooling channel, the outside of outer pipe is Be pebble bed.

According to the design of blanket of DEMO, an optimization analysis of neutronic, thermo-hydraulics and mechanics were completed. the neutronic analysis shows that the TBR is around 1.2 considering the windows of diagnostic system and heating system of plasma. The thermal-hydraulic results show that the temperature of Be pebble bed (561℃), lithium orthosilicate(870℃) and RAFMs(538℃) are lower than the temperature limit of each material. Thermal-mechanical analysis results show that the maximum stress is around 265MPa during the normal operation. The maximum stress in the first wall is 371MPa for the in-box LOCA (Loss of coolant accident), in other words, the integrity of the blanket can be achieved for the accident. According to the above key analyses, a preliminary conclusion can be draw that the new design blanket satisfies the mainly safety requirements and has potential to have a good TBR. The new design has references for the structural design of solid tritium breeding blanket.

Keywords

DEMO, TBR, Casing Structure.

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PS3-23 ABSTRACT-7fc5
B. Blanket Technology

Basic study on Simulating Moving Bed chromatography for separation of lithium isotopes

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Lithium-6 is used as a blanket material in DT fueled fusion reactors. A technology to enrich lithium-6 over 90 isotopic percent is, then, necessary to establish the fuel cycle. Though the lithium-mercury amalgam method had developed in practical use, an environmentally preferable method such as ion-exchange method needs to be developed. One of the challenges of chromatographic methods is scaling up for mass production. In the present study, the application of Simulating Moving Bed (SMB) chromatography was investigated as a lithium-6 enrichment process. SMB chromatography is applied as a separation technique in the pharmaceutical industry, fine chemical manufacturing, and bioengineering. SMB chromatography has the potential to separate substances continuously and also has the advantage of reducing the amount of waste liquid to be treated. The main parameters of SMB chromatography such as column length, switching period, flow rates were estimated for separation of lithium isotopes. The results of the study indicated the properties required for the adsorbent and the optimal combination of adsorbent and eluent. The development status of novel adsorbents is also introduced.

Keywords

Simulating Moving Bed, chromatography, lithium isotopes, isotope separation, blanket.

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PS3-24 ABSTRACT-7fee
B. Blanket Technology

Influence Of Various Gases and Water Vapors on The Processes of Tritium Release from Two-Phase Lithium Ceramics

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The generation and release of tritium in breeder lithium ceramic blankets will depend on numerous factors, for example the presence of impurities in the purge gas. As shown before, the most affecting gas impurities here are oxygen, water vapor and hydrogen (the latter is planned to be added specifically to the purge gas mixture to facilitate the release of tritium).

The study presents data from reactor experiments on the irradiation of two-phase lithium ceramics (65 mol% Li_4SiO_4/35 mol% Li_2TiO_3) under conditions of feeding various gases and water vapors into the chamber with samples.

Sections of experiments are analyzed in which a successive stepwise injection of water, oxygen, hydrogen, and deuterium vapors was carried out into a continuously evacuated chamber with samples (during the steady-state quasi-equilibrium release of tritium from ceramics). The gas composition in the chamber was recorded using a mass spectrometer and, in particular, the features of the release of tritium-containing molecules were recorded.

The main parameters of the experiments were as follows:

- Mass of irradiated ceramics: ~ 5 g;
- Flux of thermal neutrons in the irradiation zone ~ $5{\cdot}10^{13}$ n/(cm^2·s);
- Pressure of residual gases in the chamber ~ 10^{-6} torr;
- Pressure range of supplied gases and water vapors (10^{-4}-10^{-3} torr).

Based on the data obtained, mechanisms of interaction of studied gases and water vapor with tritium on the ceramic surface, as well as their influence on the processes of tritium release from ceramics, were also determined.

Keywords

Tritium, lithium ceramics, water, oxygen, hydrogen.

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PS3-25 ABSTRACT-876c
B. Blanket Technology

Updated conceptual design and analysis of breeding blanket modules integrated with divertor for CFETR

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Achieving tritium self-sufficiency (tritium breeding ratio (TBR)≥1.1) is one of the most important missions of the Chinese Fusion Engineering Test Reactor (CFETR). The space occupied by the auxiliary heating systems and diagnostic systems of CFETR leads to the decrease of the number of the breeding blanket modules. It would lead to decline of TBR and failure of tritium self-sufficiency. Therefore, a breeding blanket module concept extended to the divertor region and integrated with the divertor was proposed to increase TBR in the previous study. The updated design of the proposed blanket modules is presented in this paper. The feasibility of the blanket design is evaluated considering tritium breeding, neutron shielding, thermal-hydraulic and mechanical performances.

Keywords

CFETR, breeding blanket module, divertor.

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PS3-26 ABSTRACT-91bf
B. Blanket Technology

Characterization of a Permeator Against Vacuum mock-up with niobium membrane in lithium-lead eutectic at 350°C

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Tritium extraction from molten lithium-lead eutectic alloy (LiPb) is one of the biggest challenges to be solved for the exploitation of the WCLL (Water Cooled Lithium-Lead) as the Breeding Blanket (BB) of EU DEMO reactor. The Permeator Against Vacuum (PAV) is one of the most promising technology to reach this goal, but it needs a bigger amount of experimental data to demonstrate its viability and to support the scale-up towards DEMO relevant size. For this reason, an experimental campaign was performed at ENEA Brasimone Research Centre in TRIEX-II facility, where a PAV mock-up based on niobium U-tubes was installed and characterized in flowing LiPb. This paper describes the results obtained at a temperature of 350°C and with a hydrogen partial pressure in LiPb in the range between 110 and 230 Pa. These operative conditions are considered relevant for the WCLL BB. The LiPb mass flow rate was kept constant at 1.2 kg/s for the entire duration of the campaign. Furthermore, the results at 350°C are compared with those previously obtained at 450°C on the same mock-up and with the same flow rate, showing a reduction in the extracted hydrogen flux of about one order of magnitude. Finally, an analytical model has been set up to reproduce the experimental results in terms of permeated flux and hydrogen partial pressure decrease across the PAV mock-up. The data obtained from the model are compared with the experimental ones in the final section of the paper.

Keywords

Tritium extraction, LiPb, WCLL BB, Permeator Against Vacuum.

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PS3-27 ABSTRACT-91d7
B. Blanket Technology

Main nuclear responces of the DEMO tokamak with different in-vessel components configurations

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The DEMO breeder blanket (BB) research and development work has been performed in recent years based on a pre-defined DEMO tritium breeding ratio (TBR) requirement, which determine a loss of wall surface by non-breeding in-vessel components (IVCs). It is important to consider the integration in the tokamak of IVCs other than the breeding blanket and diveror, which consume plasma-facing wall surface and do not contribute to the breeding of tritium. Recently a very preliminary DEMO tokamak configuration was defind to arrange plasma limiters, neutral beam injectors, electron cyclotron launchers and diagnostic systems. The integration in tokamak of these IVCs requires cut-outs in the BB resulting in a loss of the breeder blanket volume, TBR and power generation, respectively.

The neutronic analyses presented here have a goal to provide an assessment of the TBR losses associated with each IVC. Previously performed study on this topic was done with simplified, homogenized BB geometry models. To address the effect of detailed heterogeneous structure of the BBs to the TBR losses due to inclusion of the IVCs in tokamak a series of the blanket geometry models was developed to integrate them in the latest DEMO base model. The assessment was performed for both types of BB, being currently developed within the EUROfusion project, the Helium-Cooled Pebble Bed (HCPB), the Water-Cool Lead-Lithium (WCLL) concepts and for the Water-cooled Lead and Ceramic Breeder (WLCB) hybrid concept of the BB.

The neutronic simulations were performed using MCNP6.2 Monte Carlo code with Joint Evaluated Fission and Fusion File(JEFF)3.3 data library. For each BB concept a 22.5^{o} toroidal sector of the DEMO tokamak was developed to assess the TBR and nuclear power generation in the breeder blankets. The TBR impact of all IVCs and the losses of the power generation were estimated as a superposition of the individual effects.

Keywords

DEMO, Breeder Blanket, Neutronics, MCNP.

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PS3-28 ABSTRACT-0add
B. Blanket Technology

Numerical and experimental analysis of magneto-convective flows around pipes

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Liquid metal flows exposed to intense magnetic fields play a fundamental role in the development of blankets for nuclear fusion reactors, where they serve to produce the plasma-fuel component tritium and to transport the generated heat. When the liquid metal circulates towards external ancillary systems for purification and tritium extraction, it interacts with the plasma-confining magnetic field leading to induction of electric currents. The related electromagnetic forces and thermal buoyancy affect velocity and pressure distributions. The resulting magneto-convective flow has peculiar features that have to be taken into account in order to evaluate heat transfer properties in the liquid metal.

In the water-cooled lead lithium blanket concept, which is one of the driver blanket designs for a DEMO reactor, in the breeding zone the volumetric heat released in the liquid metal is removed by water flowing in circular pipes. The latter represent obstacles for the liquid metal flow and their presence leads to the occurrence of a typical velocity distribution where the largest velocities are confined into thin layers tangent to the pipes and aligned with the intense magnetic field.

For improving the understanding of the underlying physical phenomena and obtaining a database for code validation, model experiments have been performed to investigate magneto-convective flow and heat transfer at two differentially heated parallel tubes immersed in a liquid metal filled box. Among other properties, electric potential has been recorded at more than 400 points on the surface of the test-section and compared with 3D numerical simulations. The latter computational results provide the basis for interpretation of measured data, since they allow differentiating between flow-induced electric potential and thermoelectric effects

Keywords

WCLL blanket, magneto-convection, simulations, model experiments.

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PS3-29 ABSTRACT-9e1b
B. Blanket Technology

EUROfusion and F4E collaboration for the European Test Blanket Module

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In 2017, a technical and programmatic assessment of both the EUROfusion DEMO Breeding Blanket and the Fusion for Energy Test Blanket Module programmes was undertaken with the objective of streamlining them and ensuring full consistency.

Fusion for Energy and EUROfusion have agreed on the need to rationalise and coordinate the use of European resources to facilitate the achievement of the Fusion Roadmap objectives, proposing an enhanced R&D programme.

This agreement allows to avoid duplication of work while, at the same time, exchange of data and technical requirements insure coherence between the two projects. Furthermore, specialised technical competences and dedicated test facilities are mutualised, selected activities are prioritised to secure the R&D needed for the Test Blanket Module programme.

The paper describes the joint execution of such R&D activities and the ongoing EUROFER97 structural material qualification for both the Helium-Cooled Pebble-Bed and Water-Cooled Lead-Lithium Test Blanket Module systems.

This joint effort focuses on the development and qualification of functional materials, predictive tools, and sensors, integrated testing of the main relevant technologies, safety analyses as well as EUROFER97 base material and weld qualification and experimental campaigns in support of design rule validation. These activities have been then grouped in four packages: (i) System Modelling, (ii) Technology R&D, (iii) Safety and (iv) Materials.

An overview of the main activities performed in the period 2021-2022 is presented.

Keywords

TBM, R&D activities, EUROFER97 qualification.

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PS3-30 ABSTRACT-9e3f
B. Blanket Technology

Laser powder bed fusion additive manufacturing of ODS-RAFM steel by prefabricated multi-component nano-oxides

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Enhancing the operating temperature of blanket modules can improve the thermodynamic efficiency of fusion reactor. Oxide dispersion strengthened reduced activation ferritic/martensitic (ODS-RAFM) steels are potential candidate materials for blanket modules operating at higher temperature. The complex structure of the blanket modules is a challenge for powder metallurgy manufacturing of ODS-RAFM steels and its joining process. Laser powder bed fusion additive manufacturing is a promising method for the fabrication of ODS-RAFM steels. The addition of nanoscale oxides, defect control in additive manufacturing materials, and homogeneous dispersion of nanophases in the metallic matrix are the keys to obtaining high-quality additively manufactured ODS-RAFM steels.

In this paper, nano-scaled $Y_4Zr_3O_{12}$ oxide powders were synthesized by sol-gel method, and $Y_4Zr_3O_{12}$ nano-oxide powders were adhered on the gas-atomized micro-scaled RAFM steel powders through dielectrophoretic controlled adsorption method. The nano-enhanced RAFM steel powders containing nanoparticles was consolidated with selective laser melting. A stable process window was determined by optimizing the parameters. The microstructure, nanoparticle evolution and mechanical properties of the high-density samples were investigated. The as-deposited ODS-RAFM steel was an epitaxial elongated grain consisting of ferrite and martensite. The oxide phase was stabilized in the melt pool and formed in-situ a homogeneously distributed $Y_4Zr_3O_{12}$ phase with a number of 2.4 $10^{24}/m^3$ and a size ranging from 5 to 65 nm. Preliminary mechanical tests demonstrated ultimate tensile strength, yield strength and elongation of 1134 ± 6.5 MPa, 986 ± 5.8 MPa and 10.8 ± 0.32 %, respectively. Grain boundary strengthening from martensite, dispersion of nanoparticles and load-bearing strengthening through strong interfacial bonding between nanoparticles and the matrix, solid solution strengthening from dissolved alloying elements and dislocation strengthening dominated by high density dislocations are the main strengthening mechanism. This study provides a potential solution for manufacturing blanket modules using ODS-RAFM steels.

Keywords

ODS-RAFM steel, Additive manufacturing, Microstructure, Mechanical properties..

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PS3-31 ABSTRACT-a755

B. Blanket Technology

Preliminary thermo-mechanical assessment of the Top Cap region of the Water-Cooled Lead-Ceramic Breeder Breeding Blanket alternative concept

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The Water-Cooled Lithium Lead (WCLL) and the Helium-Cooled Pebble Bed (HCPB) breeding blanket concepts are the two candidates currently poised to be chosen as the driver blanket for the EU DEMO reactor. Nevertheless, new variants are emerging with the aim of overtaking the potential showstoppers arisen during the WCLL and HCPB BB pre-conceptual design phase. Then, an intense campaign of exploratory studies has been launched in EU to investigate the potential performances of such alternative concepts. Among them, the Water-Cooled Lead-Ceramic Breeder (WLCB) BB concept is one of the most promising. It has been conceived as a trade-off (i. e. an hybrid concept) between WCLL and HCPB BB concepts, trying to take and integrating the best features of both. Since in the reference WCLL geometric layout the Top Cap (TC) region was identified as particularly critical from thermal and mechanical standpoints, so requiring specific design studies, also in the WLCB conceptual design activities a dedicated campaign of analysis has been necessary to preliminarily size the TC region components. Hence, in this paper, the preliminary thermo-mechanical assessment of the TC region of the WLCB BB alternative concept is presented. First, an initial geometric layout has been set-up on the basis of the main features of the reference WCLL BB TC region. Then, thermal analysis under nominal conditions has been performed and the results have shown that modifications were necessary to ensure the compliance with the thermal requirements of the structural material. Finally, once obtained an acceptable geometric layout, mechanical analyses were carried out in order to verify the fulfilment of the RCC-MRx structural design criteria, also considering an off-normal load case. The results showed that this variant is very promising in terms of structural behaviour, whereas further design iterations are necessary to lower the temperature of the breeder zone.

Keywords

WLCB, DEMO, breeding blanket, top cap, thermomechanics, FEM analysis.

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PS3-33 ABSTRACT-a928
B. Blanket Technology

Thermal-hydraulic study of the EU-DEMO Water Cooled Lithium Lead Breeding Blanket Primary Heat Transport System

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The EUROfusion consortium is developing the design of a DEMOnstration fusion reactor (EU-DEMO) that will follow ITER on the path to harnessing fusion energy. The EU-DEMO is designed to supply net power to the grid. Therefore, proper critical assessments of the cooling and energy conversion systems of the tokamaks are required because they play a key role in the design and licensing of the overall facility.

The EU-DEMO reactor will be based on the tokamak concept and as such it can rely on well-known intrinsic safety features (e.g. low decay heat, huge inertia of the structures, etc.). Nevertheless, transients involving under-cooling of the structures could challenge the in-vessel components and their cooling systems due to the practical difficulties of suddenly stopping the fusion operations. Indeed, a rapid plasma termination would quickly arrest the fusion reaction but would result in the loss of plasma control and potential damage to several plasma-facing components, while a fully controlled shutdown could take approximately 100 to 200 seconds, thus making it necessary to ensure a high cooling capability of the in-vessel components for a longer period of time to avoid endangering the components themselves and the overall safety of the plant.

In this context, the University of Palermo, in collaboration with EUROfusion, has launched a research campaign to study the thermal-hydraulic behaviour of the Primary Heat Transport System (PHTS) that feeds the Water Cooled Lithium Lead Breeding Blanket (WCLL BB) during an anticipated under-cooling transient. Particular attention has been given to evaluating which parameters and system features most influence the transient evolution in order to draw useful lessons for system design. The activity has been carried out following a theoretical-computational approach based on the adoption of the TRACE thermal-hydraulic system code. The models, assumptions and results of the analyses are reported and critically discussed here.

Keywords

DEMO, Breeding Blanket, WCLL, PHTS, Thermal hydraulics.

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PS3-34 ABSTRACT-a909
B. Blanket Technology

Experimental Investigation of Heat Transfer Performance of the Helium-cooled Annular Gap in the Breeder Zone of the EU-DEMO HCPB Breeding Blanket

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The conceptual design of the Helium-Cooled Pebble Bed (HCPB) breeding blanket has been modified recently via introducing a new design of the breeder zone that is called the fuel-breeder pin. A mock-up of this fuel-breeder pin has been tested in the Helium Loop Karlsruhe (HELOKA) facility in order to investigate its thermal-hydraulic performance under the HCPB operating conditions (i.e., coolant helium at 300 °C and 8 MPa pressure). The focus of this study is on the heat transfer performance of the coolant annular gap (15 mm) of the fuel-breeder pin mock-up. In particular obtaining the experimental Nusselt (Nu) numbers and investigating the possibility of enhancing the heat transfer of the coolant gap by changing the surface characteristics of its inner surface, which is formed by what is called the mock-up insert. Therefore it was planned to test the mock-up with several inserts that have different conditions of surface roughness and structure. The experimental Nu numbers as a function of Reynolds numbers are presented and compared with those calculated by relevant Nu correlations from literature. Particularly, the impact of the surface characteristics on the Nu numbers is discussed.

Keywords

Breeding blanket; HCPB; breeder unit; helium cooling.

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PS3-35 ABSTRACT-ad6c
B. Blanket Technology

Solubility of Chromium in Lead-Lithium Eutectic Alloy

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The reduced activation ferritic-martensitic steel EUROFER is considered as a baseline structural material for different concepts of European DEMO breeding blanket (BB), including the Water Cooled Lithium Lead (WCLL) concept. In the BB concepts using as a tritium breeder liquid metal Pb-16Li, which is in direct contact with the structural material, dissolution is the primary cause of corrosion. The rate of dissolution within the Pb-16Li flow channels depends on liquid velocity, geometric arrangement and solubility of the EUROFER constituents in the liquid. Successful transfer of experimentally obtained EUROFER corrosion rates to real operating conditions of DEMO BB as well as predictions of corrosion rates from fundamental laws of mass transfer require the knowledge of solubility data of the main constituents of EUROFER in Pb-16Li, namely Fe and Cr. Existing solubility correlations are based on fitting observed corrosion data using a dissolution/precipitation model, which leads to large uncertainties given by the particular geometry and/or temperature regime of the experimental loops.

This contribution reports on experimental measurement of chromium solubility in liquid Pb-16Li. The Pb-16Li alloy in the as-received form was introduced into a molybdenum crucible. During the experiment, the liquid metal was agitated by an impeller made of pure chromium, which was dissolving in Pb-16Li at given temperature. Pb-16Li samples were taken into ceramic sampling tubes after a sufficient amount of time had passed for the liquid to become fully saturated with chromium. Spectroscopic techniques were used to determine the chromium and other impurities concentrations in the Pb-16Li alloy.

Keywords

Breeding blanket, Pb-16Li, corrosion, chromium.

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PS3-36 ABSTRACT-bc8a
B. Blanket Technology

Electrochemical extraction of nitrogen impurities in liquid lithium using chloride molten salt

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In fusion blanket, liquid breeders have some advantages over solid breeders, such as simpler blanket structure, no material deterioration due to neutron irradiation, and continuous tritium recovery during fusion reactor operation. Liquid lithium (Li) is a candidate liquid breeder material, and easily contains light element impurities such as nitrogen and hydrogen. These impurities have negative influences on the compatibility with structural materials. Typical methods for impurity removal include cold-trap method, which uses the solubility change of impurities, and hot-trap method, which uses a titanium alloy as an absorbent for nitrogen in Li. However, these methods cannot monitor the concentration of impurities. Therefore, our research group has focused on electrochemical method that can continuously remove impurities and measure their concentration on-line. In this study, nitrogen impurities in Li were electrochemically extracted. The experimental setup was an electrochemical cell that has double liquid layer of Li and chloride molten salt (LiCl 58.5 at.%–KCl 411.5 at.%). Li was used as a counter electrode (CE). The reference electrode (RE) material was also Li. The working electrode (WE) material was a nickel wire. Lithium nitride (Li_3N) was directly added as nitrogen impurities to Li. Then the change of cyclic voltammograms (CV) caused by redox reaction at WE was obtained. Furthermore, chronoamperometry (CA) was performed sweeping with argon gas. A portion of gas was fractionated and analyzed by gas chromatography (GC) with a column for nitrogen gas detection. Additional GC analysis for ammonia gas detection are in progress. The results of these analysis will be presented at the conference.

Keywords

Lithium, liquid metal, nitrogen imprities, molten salt.

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PS3-37 ABSTRACT-bd5a
B. Blanket Technology

Evaluation of the impact of various operation parameters on isotopic exchange process for TER HCPB DEMO

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The tritium generated in the pebbles beds of the breeding blanket and its tritiated water portion are adsorbed in the Tritium Extraction and Removal System (TER). In the Reactive Molecular Sieve Bed (RMSB) case, where tritiated water is adsorbed, the tritium shall be recovered via isotopic exchange with a H2/D2 stream that is circulated in closed loop with Tritium Plant during the regeneration process. During the regeneration the vapours are flowing from one section to the following one. The isotopic exchange between the trapped tritiated water and the swamping gas, H2 and D2, is determined by the temperature variation along the molecular sieve bed as resulted from heat transfer between the molecular sieve and the purge gas that realize the heating of the molecular section.

In order to support the development of the operation and regeneration pattern of the RMSB from the TER of the Helium Cooled Pebbles Bed (HCPB) concept, a software that simulates the isotopic exchange process between the tritiated water trapped on the RMSB from TER and a swamping gas has been developed.

The software will support the development of the regeneration pattern, i.e. swamping gas flow-rate, temperature evolution inside the RMSB sections by incorporating the design features of the in-built heat exchanger in the bed sections. Several runs have been performed aiming to show the impact of various operation parameters on the tritium extraction efficiency as a basis for the development of the RMSB regeneration pattern.

This paper aims to presents the results of the evaluations related to the swamping gas flow-rate impact on the regeneration time of the RMSB, the impact of the regeneration time on efficiency of the tritium extraction and the temperature evolution inside the RMSB during the regeneration process.

Keywords

Reactive molecular sieves beds, isotopic exchange process, swamping gas, temperature evolution.

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PS3-38 ABSTRACT-c077
B. Blanket Technology

Design and manufacturing of tritium-carrier process pipes and associated guard-pipe for ITER Test Blanket Systems

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As part of the ITER TBM Program, two equatorial ports (#16, #18) of the ITER machine are dedicated to the testing of mock-ups of four different tritium breeding blankets called Test Blanket Modules (TBMs). Each TBM is part of an independent Test Blanket System (TBS) that includes also several ancillary systems, in particular the cooling system, the Tritium Extraction System (TES), the Neutron Activation System (NAS), the Tritium Accountancy System (TAS), the Instrumentation and Control (I&C) system.

The main equipment of each TES, each NAS and each TAS are located in different rooms of the tokamak and tritium nuclear buildings. The pipes connecting such equipment are called Connection Pipes (CPs). They carry tritiated fluids and cross buildings rooms where circulation of maintenance workers is expected to be frequent and regular. Because of the tritium permeation through the stainless steel pipes bare walls, a second confinement barrier has been envisaged and developed. For this purpose, the design team decided to encapsulate the process pipes in larger pipes called guard-pipes.

The paper addresses the rational and benefits of implementing guard-pipes around the process pipes. It describes also the outcomes of the design development phase carried out over the last few years to reach the stage of manufacturing design, with the fabrication of some components, and also to complete the construction design ready for execution. The design accounts for the specific sequence of assembly and welding operations during first, the pre-fabrication steps in the workshop, and then the installation operations in the various rooms. It will focus on some technical complexities which were tackled, triggered in particular by the required compliance with French regulations for nuclear safety and for nuclear pressure equipment.

Keywords

TBM, tritium, piping, guard.

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PS3-39 ABSTRACT-604b
C. Fuel Cycle and Tritium Processing

Tritium Compatible Pump Development

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All-metal scroll pumps have become the standard for tritium processing, and turbomolecular pumps have been used when higher vacuum levels are required. Alternative all-metal scroll pump options were of high interest following the liquidation of Normetex. New turbomolecular pumps are being brought to the market as older models are discontinued, which prevents the ability to replace-in-kind. Savannah River National Laboratory has been working with Air Squared to develop an all-metal scroll pump similar to the Normetex with comparable flow characteristics. Two pump designs were developed: a stainless-steel version and an aluminum version. Various coatings were evaluated for the aluminum version to decrease the wetted-aluminum tritium surface contamination, with the best performing being used in the final pump fabrication. Pumping characteristics were evaluated on both versions and compared directly against a Normetex pump using the same test system. SRNL has also been evaluating multiple high vacuum pumps for potential tritium service compatible models. Turbomolecular pumps were evaluated from multiple manufacturers, which included models with standard greased bearings and magnetic levitation bearings. Compression ratio pump curves were measured for the pumps before being burst tested and autopsied to identify components of concern when exposed to tritium. Results of the Normetex/Air Squared comparison pump performance tests and turbomolecular pump evaluations will be presented.

Keywords

Tritium, pumps, vacuum, processing.

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PS3-40 ABSTRACT-64d8
C. Fuel Cycle and Tritium Processing

Planning Tritium Recovery System for Integrated Breeding Test Facility in South Korea

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The fusion power plant can be commercialized through experimental reactors and demonstration reactors. The ITER is one of the most useful experimental reactors to verify the DT nuclear fusion power production to project the commercial power plant scale. The integrated breeding test facility is planning to test the various breeding blanket concepts under continuous neutron irradiation environment for the blankets of the demonstration reactor in future.

The integrated breeding test facility will produce a bit amount of tritium to be purified, concentrated, and stored safely in itself. The conceptual design of the tritium recovery system is developing with various tritium handling technologies. Some of the technologies are selected for the system design with considering the applicability toward the design of the demonstration reactor. In this study the on-going results of the conceptual design will be addressed with including its functions and requirement, selection of technologies, and overall process flow diagram.

The results of the conceptual design will be identified the R&D items to be developed and applied the tritium recovery system for the next design stage. The overall experience with the design, the fabrication and assembly, and the operation will be one of the most important step to develop the design of demonstration reactor.

Keywords

Tritium Recovery System, Breeding Blanket Concept, Integrated Breeding Test Facility.

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PS3-42 ABSTRACT-752a
C. Fuel Cycle and Tritium Processing

Development of a semiconductor hybrid pulse switch module with high repetitive rate for compact torus injector

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Compact Torus Injector (CTI) is one of the most promising central fuelling methods for the future fusion reactor, which must be implemented in a repetitive mode. The realization of CTI repetitive operation mainly depends on its power supply system, and the pulse switch is the technical bottleneck which restricts its further development. A pulse switch module is designed, which is composed of two semiconductor switches in series, GTO-like thyristor and Fast Recovery Diode (FRD). The switch module topology is a low-inductance coaxial structure. The GTO-like thyristors are used to control the main circuit turn-on, and the FRDs are used to protect the GTO-like thyristors by turning them off in advance. A pulse power discharge test platform of the semiconductor hybrid switch module is designed. Under the applied voltage of 10 kV, the peak current of discharge circuit reaches 200 kA, the half-wave width of the current waveform is 15 μs, the repetitive rate is 10 Hz (under the peak current of 100kA), and a maximum pulse number of 100 have been achieved. This shows that the FRDs can effectively prevent the switch module damage caused by the reverse overvoltage, and the switch module can improve the parameters and reliability of the power supply.

Keywords

Compact torus injector, Pulse power supply, Low-inductance coaxial structure, GTO-like thyristor, Fast recovery diode.

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PS3-43 ABSTRACT-76ca
C. Fuel Cycle and Tritium Processing

Upgrade and test of the compact torus injection system for the EAST central fuelling

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The compact torus (CT) is a self-organized plasma with high speed and high density, which is considered as the most promising solution for central fuelling in the future fusion reactor. The new compact torus injector system (EAST-CTI) is being developed on the Experimental Advanced Superconducting Tokamak (EAST). The EAST-CTI is composed of a CT injector, fast gas valve system, power supply system, vacuum pump system and characteristic diagnostic system. Using the ignition as a high voltage switch, the power supply system has been upgraded, so that the EAST-CTI has a certain capability to fire repetitively, and the power supply system provides a high performance for CT formation and acceleration. The discharge current in both formation and acceleration bank units of CT exceeded 330 kA. At the same time, the auxiliary power supply system, acquisition system and control system are modified to be suitable for repetitive mode. The maximum fire rate achieved on EAST-CTI was 2 Hz. A pulse switch module composed of two semiconductor switches in series, GTO-like thyristor and Fast Recovery Diode (FRD) is designed. Under the applied voltage of 10 kV, the peak current of discharge circuit reaches 200 kA, the repetitive rate is 10 Hz (under the peak current of 100kA), and a maximum pulse number of 100 have been achieved. A magnetic compression platform for CT plasma has been developed, where the CT plasma interacts with the external gradient magnetic field. The motion of CT in the magnetic field was measured by the 2D magnetic coil array and electrostatic probe, and the results indicate that the CT plasma penetrated the magnetic field strength of 0.6 T. At present, the EAST-CTI is in the engineering testing phase under different discharge parameters. The injection velocity of CT was 50-180 km/s and electron density was 0.1-2×10^{22} m^{-3}.

Keywords

Compact torus, Central fuelling, Repetitive operation, External magnetic field.

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PS3-44 ABSTRACT-793f
C. Fuel Cycle and Tritium Processing

On the possibility to recover Plasma Enhancement Gases from DEMO tokamak exhaust stream

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Fusion machines, like DEMO, are characterized by low tritium conversion rate that defines the composition of the tokamak exhaust throughput. Although a reference plasma scenario is not finalized yet, the unburnt fuel is expected to be more than 95% of the tokamak exhaust stream and must be recovered and purified in the Exhaust Processing System (EPS). The residual throughput features a complex mixture of multiple species. Among these, there are the Plasma Enhancement Gases (PEGs), comprising inert noble gases for the enhancement of the radiation from core plasma and divertor. In the most recent DEMO plasma scenario, Xe and Ar are assumed as core radiator and for divertor seeding, respectively, although the required amount of these gases is still an open point and varies over several orders of magnitude.

The present paper deals with the investigation of the opportunity to recover and separate PEGs for the reuse in the machine, being economics the primary driver for the development of a viable recycling strategy. The objective is to assess possible technologies for PEGs recovery and separation and to individuate ranges of PEGs throughput for which the recycling approach is economically favorable. Issues related to the activation of noble gases are also considered in the assessment of the recovering strategy. Focusing on cryogenic distillation for separating the PEGs, the analyses are carried out with a process simulation tool, investigating the effect of relevant parameters on the design and the economics of the system. Outcomes of the analyses are useful to support the ongoing discussion on PEGs amount and composition.

Keywords

Fuel Cycle, Exhaust Processing System, Cryogenic Distillatio, Process Simulation.

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PS3-45 ABSTRACT-7a71
C. Fuel Cycle and Tritium Processing

Engineering Design of Multiple shattered Pellet Injection System for HL-2M

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Disruption is a destructive accident event which may occur during high-performance operations in tokamak. Disruption mitigation is one of the high-priority subject in fusion research. Injecting large quantities of high Z materials is a generally method to mitigate the disruption. Mostly, massive gas injection (MGI) and shattered pellet injection (SPI) are the most common techniques for disruption mitigation and are also regarded as the most prospective schemes. Due to deeper penetration and higher efficiency, SPI has more advantages than MGI when injecting the same number of particles.

The HL-2M is a new tokamak device in Southwestern Institute of Physics (SWIP). Comparing with the HL-2A tokamak, the plasma parameters of HL-2M are greatly improved. By the end of 2022, 1 MA plasma current has been achieved in HL-2M . In order to alleviate the destructive effects and study the physical mechanism and operation strategy of disruption, it is necessary to establish a multiple pellet injection system for HL-2M tokamak.

In-situ condensation and pneumatic acceleration technologies are adopted for the multi-SPI prototype, 3 pellets can be prepared and injected in the same time, one of pellet is 7 mm and the other two are 5.5 mm in diameter. The prototype has 3 individual gas supply circuits but shares same one propelling circuit. As the key component of multi-shoot SPI, the rationality of pellet preparation structure is related to fulfil the requirements of experiments. The thermal analysis by ANSYS is performed to simulate he heat and temperature distribution in different operation conditions. The thermal shock caused by propellant gas is also evaluated. The results show the structure design of the pellet preparation system can satisfy the design requirements.

Keywords

HL-2M tokamak; Disruption mitigation; Shattered pellet injection.

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PS3-46 ABSTRACT-86c7
C. Fuel Cycle and Tritium Processing

Empirical model for the time dependence of the radio-induced D-T equilibration in a turbulent Flow

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Whenever tritium and the stable hydrogen isotopes are combined, a radio-induced equilibration process starts. Although the concentrations in chemical equilibrium can be derived from first principles, the time evolution of the equilibration reaction towards the steady state is not fully described yet. In particular, detailed knowledge of the reaction speed is essential to accurately model the behavior in tritium gas streams and tritium process systems. In earlier works, either purely theoretical or experimental investigation have been published; however, up to now, no studies combining an empirical model and systematic experimental studies were performed.

We developed an empirical reaction model, that is benchmarked to experimental data using the TRIHYDE facility. This facility consists of two vessels of accurately known volume, pressure and temperature. These vessels were filled with pure T_2 and D_2 respectively and the gases are subsequently circulated using a metal-bellows pump. By varying the initial fill pressures, the time constants of DT-formation in dependence of (i) T_2-concentration and (ii) absolute pressure was determined experimentally. The isotope fractions are *in-situ* monitored using Raman spectroscopy, which allows unambiguous tracking of the individual hydrogen isotopologues. Because of the continuous circulation of the gas stream, the observations are applicable for turbulent conditions, similar to those in an expected fusion fuel processing application.

In this contribution, we will introduce the empirical reaction model and compare the model predictions with measured time constants for varying D to T ratio at constant pressure. We will also give an outlook to future model refinement with respect to system geometry and total pressure.

Keywords

Tritium fuel cycle, tritium analytic, chemical equilibrium, tritium processing.

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PS3-47 ABSTRACT-898c
C. Fuel Cycle and Tritium Processing

Tritium sorption measurement of an elementary tungsten sample by BIXS

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Beta Induced X-ray Spectrometry (BIXS) is a versatile tool in tritium analytics for activity measurement of gases, liquids and solids. Its prime advantage is the non-destructive, high-sensitivity, in-situ measurement of contaminated samples. While the electrons from tritium beta-decay are comparatively short-range particles in solid matter, due to the maximum energy of 18.6 keV, they induce characteristic long-range fluorescence and Bremsstrahlung in the X-rays during the absorption process. These X-rays can be measured outside of a tritiated sample volume if an X-ray transparent window is used. Interpretation of the resulting X-ray spectra can be difficult, since they depend on a variety of factors.

In this work, a new BIXS cell has been used for tritium sorption measurements of an elementary tungsten sample. The sample was repeatedly exposed to high-purity tritium between 100 Pa and 1000 Pa. During and after exposure, BIXS activity measurements were performed. After tritium loading two different decontamination measures were investigated, bake out at 200°C and long-term evacuation over ≈ 3 days.

Similar to previous BIXS measurements of gold and stainless-steel samples the tungsten sample showed no saturation effect but a decreased tritium sorption capacity after around 50,000 Pa h of tritium exposure. The surface activity after 200,000 Pa h of tritium exposure was estimated with $3.6 \cdot 10^6$ Bq/cm^2. A total of five bake outs over 90 hours (each/total?) resulted in a surface activity reduction of ≈ 20%. The count rate trend showed an exponential decrease over the bake out runs. Long-term evacuation showed no significant count rate decrease.

Keywords

BIXS, tritium, surface contamination, TRIADE, TLK, activity measurement.

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PS3-48 ABSTRACT-8bda

C. Fuel Cycle and Tritium Processing

Tritium Extraction and Accountancy Systems Design for ITER HCCP-TBS

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One of the main objectives of the ITER project is to test the generation of tritium through the test blanket systems (TBS) program. The Helium-Cooled Ceramic Pebbles test blanket system (HCCP-TBS), jointly developed by F4E and ITER KOREA, is being used to generate tritium in a solid breeder, where pebble beds made of a mixture of lithium ceramic compounds are employed. The scope of this paper is to give a general overview of design aspects of the Tritium Extraction System (TES) and the Tritium Accountancy System (TAS), including a description of the main functions and associated components and a preliminary 3D layout.

TES is responsible for extracting tritium from the Test Blanket Module, storing the daily tritium production in getter beds in the form of Q_2, and delivering the Q_2 to TAS. The Tritium Accountancy System uses ionization chambers to measure the amount of tritium and stores the Q_2 in getter beds until the Short-Term Maintenance period, during which the Q_2 is sent to the Tokamak Exhaust Processing or Detritiation System for further processing.

TES is comprised of about 90 process components and about 70 I&C components, while TAS is comprised of about 100 process components and about 90 I&C components. The getter beds, which must operate in both adsorption and regeneration modes at different temperatures, are key components of both TES and TAS. Sizing the getter beds is a critical activity for the design of the systems, especially given the stringent space constraints for their implementation. Furthermore, both TES and TAS must comply with a tritium confinement function, which makes compact layout development essential.

Keywords

Tritium extraction, Tritium accountancy, HCCP-TBS, ITER.

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PS3-49 ABSTRACT-8b5a
C. Fuel Cycle and Tritium Processing

Overview of Tritium Management in WCLL Test Blanket System of ITER

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A key aspect for the achievement of the fuel self-sufficiency of future fusion power plants is the management of tritium in the different stages of the reactor fuel cycle. In particular, one of the greatest challenges for the success of liquid metal breeder concepts such as the Water-Cooled Lithium-Lead (WCLL) is the extraction of tritium from the liquid LiPb eutectic alloy (15.7 at. %Li), which fulfils the functions of tritium breeder, tritium carrier and neutron multiplier, and the processing in a form suitable for an accurate tritium accountancy. Within this frame, significant advancements have been carried out in the last years as regards the WCLL Test Blanket System (TBS) of ITER. Tritium is mainly produced inside the LiPb as a by-product of the Li(n,α)T nuclear reactions, and extracted through the Tritium Extraction Unit (TEU). The reference technology for the TEU is the Gas-Liquid Contactor (GLC) in the packed-column configuration, which uses helium as stripping gas, with the possibility to add a small amount of hydrogen to allow the isotopic exchange with tritium enhancing the extraction efficiency. Tritium is then concentrated in Zr-Co getter beds before to be sent to the accountancy system. The accountancy is performed through static and dynamic technologies in order to provide precise and reliable tritium measurements, which are an essential need in view of an application to the European DEMO reactor. In this paper, the present status and the design solutions foreseen for tritium management in WCLL-TBS are presented and discussed. The implementation and optimization of the conceptual design of the LiPb loop, the Tritium Extraction System (TES) and the Tritium Accountancy System (TAS) have been performed in order to manage tritium concentration in LiPb and in the ancillary systems.

Keywords

Tritium management, tritium extraction, tritium accountancy, lithium-lead, ITER.

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PS3-50 ABSTRACT-9cff

C. Fuel Cycle and Tritium Processing

Superhydrophilic Metal-Organic Frameworks Film Modified Surface for Tritium Removal from Tritiated Heavy Water

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Tritium extraction from heavy water moderators with catalytic exchange reactions had hydrogen explosion, fire risk, and complicated system design, which restricted the industrial scale usage. Herein, we reported a safer and simpler tritiated heavy water distillation process with metal-organic frameworks (MOFs) modified surface. MOFs with microstructure, microporous pores, and hydrophilic nature showed superhydrophilic properties, which was a benefit to better interface contact properties between water drop, vapor, and packing. High separation efficiency was achieved with height equivalent to a theoretical plate (HETP) of 1.53 cm under optimized operation conditions. Pilot-scale two-column tritiated heavy water distillation system was simulated and optimized based on experimental results with a high separation factor and recovery rate of tritium. This work demonstrates the first attempt at MOFs engineered surface for tritiated heavy water treatment, advancing a new concept in the design of tritiated heavy water distillation with intrinsic safety for industrial applications.

Keywords

Superhydrophilic, metal-organic frameworks, hydrogen isotopes, tritiated heavy water, water distillation.

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PS3-51 ABSTRACT-6d3d
D. Material Engineering for FNT

Fracture toughness evaluation by the 3 points bending for pure tungsten

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Evaluation of structural integrity of fusion reactor is a crucial issue, and necessary to consider material deterioration during using environment. Fracture toughness is an indicator of the resistance of the material to fracture and is one of the base properties required for the evaluation of structural integrity. However, since tungsten, which is expected to be used in a divertor or blanket of a fusion reactor, it is difficult to introduce fatigue pre-cracks for evaluating fracture toughness because it is a brittle material. Therefore, fracture toughness tests using specimens without fatigue pre-crack or high temperature pre-cracking specimens are being considered. Though, when cracks are introduced at high temperatures, accurate evaluation becomes difficult because plastic deformation occurs at the crack tip. This study considers the technique of fatigue pre-cracking at room-temperature and testing the fracture toughness by the 3 points bending for pure tungsten. In addition, the validity of the data was checked with the fracture toughness standard ASTM E399 and key technical issues based on the obtained knowledge are described.

The two kinds of pure tungsten with different manufacturing processes were fabricated by A.L.M.T. Corp., Japan. Specimen length, width, and thickness were 25mm, 5mm, and 3mm, respectively. Fatigue pre-cracking is started by using axial compression fatigue and then grown by 3 points bending fatigue.

Specifically, 6 violation items out of a total of 12 criteria items could be identified, and it is believed that most items that have not passed can be solved by changing the size of the specimen or cracks introducing with careful. However, it is thought that the branching or diffraction of fatigue cracks, which are difficult to solve in this way, is due to the characteristics of tungsten materials.

Keywords

Tungsten, Fracture toughness, pre-cracking.

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PS3-52 ABSTRACT-6d59
D. Material Engineering for FNT

Thermal Cycling of Titanium Beryllide to Simulate its Operating Conditions in DEMO HCPB Breeding Blanket

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The helium-cooled pebble-bed (HCPB) breeding blanket concept of the European DEMO will use solid titanium beryllide ($TiBe_{12}$) blocks for neutron multiplication. However, the limited ductility and fracture toughness of titanium beryllide make it prone to cracking and fracture due to pulsed reactor operation with fast heating and cooling. Thermal cycling tests have been proposed to validate computer simulation predictions of temperature distribution and assess the suitability of $TiBe_{12}$ blocks for the HCPB breeding blanket. The first experiment involved heating and cooling $TiBe_{12}$ blocks in the 350–930°C range for 205 cycles. The blocks retained their structural integrity without cracking, but microstructural analysis revealed porosity due to the evaporation of free beryllium from areas near the outer surface exposed to the highest temperatures. In 2022, a second campaign was launched with refined $TiBe_{12}$ block sizes and chemical composition, subjecting the blocks to 40% of end-of-life cycles in an increased temperature range of 440–1000°C. As of 650 cycles, one of six blocks had cracked and was replaced, while the others retained their shape and density. The experiment is ongoing with 800 cycles, and the results will provide valuable insights into the performance of $TiBe_{12}$ blocks under thermal cycling conditions.

Keywords

Titanium beryllide, neutron multiplier, blanket, DEMO, thermal cycling.

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PS3-53 ABSTRACT-1925
D. Material Engineering for FNT

Development of the design of a resistivimeter and an electrochemical sensor for impurities detection on liquid lithium for IFMIF-DONES

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One critical point for the operation of the Impurity Control System of IFMIF-DONES is the online impurities control. Current solutions are in a low readiness level of industrial development. This work presents a development from the previous basic designs (TRL1-2) to a medium-advanced design (TRL2-4) of a resistivimeter and an electrochemical sensor mock-ups to be tested at CIEMAT.

The resistivimeter is a system devoted to the measurement of the total concentration of non-metallic impurities present in lithium. It is based on the increase in electrical resistivity produced by the various anions dissolved in the lithium compared to that of pure lithium. The design of the resistivimeter derives from the one used in Lifus 6 by ENEA, which is based on the four-point measurement.

One significant development is filling the resistivimeter through the Bernouilli effect instead of using a pump, improving the quality of the measurements by reducing electromagnetic noise. Alternative heating systems are being analyzed in order to obtain a homogeneous temperature with a tolerance of ±0.5℃ necessary for accurate measurements.

The electrochemical sensor is used to determine the hydrogen isotopes concentration in the liquid metal. It is based on the measurement of the difference in potential (V) due to the different concentrations of hydrogen between two electrodes. The sensor's electrodes are: the reference electrode made of Li+LiH with a known concentration of hydrogen and the working electrode made of $CaCl_2$+5 at-% CaH_2 (hydrogen conducting electrolyte).

In the design of this sensor, niobium was selected for the capsules for its good permeability of hydrogen and good corrosion resistance to liquid lithium at 300℃. A sealed furnace filled with argon will be used to anneal the working electrode at 875℃. EBW is selected as joining technique and the assembly will be performed inside a glove box with inert environment.

Keywords

IFMIF-DONES, Liquid lithium, Electrochemical hydrogen sensor, resistivity meter.

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PS3-54 ABSTRACT-71b4
D. Material Engineering for FNT

Diffusion bonding of cds composite with al2o3 and 316l stainless steel at a gleeble 3800

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CDS-316L and CDS-316L/1 wt% Al_2O_3 specimens were successfully diffusion bonded with standard 316L stainless steel rods with a Gleeble 3800 physical simulator. The bonding parameters were within the temperature range of 950-1000 °C and axial mechanical pressures of 20 – 30 MPa, experiments were established based on earlier works [1]. The goal was to investigate the bonding applicability of CDS-316L specimens. It was found that the creep properties of the CDS specimens at that high temperature and high current density were much higher compared to the standard 316L steel. We supposed a high electrical resistivity inside the CDS specimens with larger volumetric heating during the welding process.

The CDS 316L specimens were prepared by powder metallurgy via spark plasma sintering (SPS) similarly to [2] with the following composition: Fe–16.8% Cr–12% Ni–2.5% Mo–1.5% Mn–0.6% Si (wt.%). The addition of Alumina particles increased the microhardness of the 316L CDS steel. No diffusion zones have been observed within the investigated magnification for all composites, where the interfaces between the different zones were well defined at all bonded specimens.

The strain rate and the flow stress were investigated. It was found a very slow and continuous axial deformation at the given stress, where the slow deformation can be the results of the diffusional creep.

[1] Baross, T., et al. (2020) Diffusion bonding experiments of 316L steels in a Gleeble 3800 thermomechanical simulator for investigation of non-destructive inspection methods, Fusion Engineering and Design, Vol. 160, 111768, ISSN 0920-3796, https://doi.org/10.1016/j.fusengdes.2020.111768

[2] C. Balázsi, et. al. (2011) Preparation and structural investigation of nanostructured oxide dispersed strengthened steels. J Mater Sci 46, 4598–4605. https://doi.org/10.1007/s10853-011-5359-1

Keywords

Diffusion bonding, 316L, CDS 316L, ceramic dispersion strengthened steel, Gleeble 3800.

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PS3-55 ABSTRACT-72b0
D. Material Engineering for FNT

Properties of CLAM Steel after Irradiation in Fission and Spallation Neutron Environment up to 21dpa

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CLAM steel is the primary structural material of fusion testing blanket material in China. It is important to understand the neutron irradiation effects of CLAM steel for fusion engineering application, especially in an environment close to the service condition in future fusion energy systems. To promote the application process of CLAM steel, neutron irradiation experiments of CLAM steel with highest dose 21dpa have been carried out under fission and spallation neutron environment. In this work, current status of the study on the neutron irradiation properties of CLAM steel will be introduced, and the results from the neutron irradiation data analysis for nuclear engineering application also be included. The results show that CLAM steel have good irradiation resistance. Based on current results and considering the engineering application requirements, the future research and development strategy on neutron irradiation damage of CLAM steel will be proposed to contribute to its industrial applications in nuclear energy systems.

Keywords

CLAM, Neutron irradiation, Properties, Fusion energy.

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PS3-56 ABSTRACT-74b1
D. Material Engineering for FNT

Development of thick nanostructured tungsten coatings for ionic radiation protection

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First wall materials in nuclear fusion reactors are directly exposed to the plasma and then, high thermal stress and radiation. Their function is the protection of structural materials of ions created in the explosion of deuterium-tritium capsules. They have to be capable of maintaining the protection under high flux of high energy ions (in the range of MeV).

In this work a protective coating for sensor covers have been developed. According to ITER calculations, 6 microns of W are necessary to protect the sensors from the ionic radiation. Additionally, it is neccesary to provide electrical conductivity, having a Faraday cage.

Growing thick W coatings by sputtering uses to be difficult due to residual stresses accumulated in the coating. Playing with process parameters, W films changes quickly from highly compressive to highly tensile stresses, which makes difficult the control. In this work, a semi-industrial PVD system has been used, with 550x150 mm targets. Substrate (sensor covers) is made of AlN. A first layer of Cr (300 nm) has been deposited, to promote adhesion.

The coating is based on Tungsten and is applied by DC sputtering: 4 kW and 500 V bias. The growing process is assisted with heat, setting the temperature at 500ºC. The growing process takes 330 minutes. Thickness is measured in silicon samples (cross section in SEM) and verified in the covers by resistivity measurements. Adhesion of the coatings is tested on samples by means of a pull-off test (>50 MPa required). After process development, sensor covers have been produced to be installed.

Keywords

Coatings, tungsten, ionic radiadion protection.

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PS3-58 ABSTRACT-a0e3
D. Material Engineering for FNT

A "proof of principle" experiment quantifying helium nucleation and transport parameters in lead-lithium eutectic (CAVITEST)

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A proof-of-principle experiment quantifying helium nucleation parameters in LLE appears as compulsory. A new test is proposed to: (1) demonstrate empirically as “proof-of-principle” the sufficiency conditions for bubble nucleation under radiogenic He local super-saturation conditions, (2) produce quantitative data as… atoms per bubble, number of nucleated bubbles, frequency of nucleation, (3) provide direct experimental data on bubble transport characteristics (migration and isothermal stability); velocity, implosion and coalescence, (4) generate data on H_2 or D_2 mass-transfer coefficient into bubbles; (5) provide empirical information on the possible build-up of LM/He/steel interface. A conceptual experimental reference set-up is initially proposed in order to directly proof sufficiency conditions under He super-saturation from the ionic breeding. Few mg of an (Am-241 powder) alpha source (^{241}Am (126,8 GBq/g) -> ^{237}Am + $^{4}{}_{2}$He (5,5 MeV)) generates the local oversaturation conditions (> 10^{-3} mol m^{-3}) in the projected range (few microns) volumes. It is an ion implantation and atomic diffusion problem. If ionic breeding operates as “sufficient condition”, bubbles with a given size will be nucleated. Once nucleated, the bubbles if stable should start to rise and grow-up (isothermally), due to the hydrostatic pressure difference at the column (proof-of-principle). With a good statistics on the number and sizes of bubble at the top of the ½ m column; the concentration at saturation breeding bubbles at the bottom and then the value of K_H (the Henry´s solubility constant in the LLE) could be determined together with the average number of He atoms nucleating into a bubble. If H_2 is injected in the molten LLE (ex. by using a thin PdAg permeation membrane) and stable He bubbles are formed; H_2 mass-transfer coefficient rates into He-bubbles (hm) could be also determined. This h_m value would be crucial to justify the impact of bubbles on tritium transport at LM BB channels.

Keywords

Helium, Bubbles, lead-lithium, breeders.

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PS3-59 ABSTRACT-8d97
D. Material Engineering for FNT

Corrosion Behavior of CLAM steel in Liquid Lead-Lithium Alloy

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The corrosion of materials in lead-lithium alloy is one of the important factors to maintain the integrity of components, which is one of the key engineering problems to be resolved in the development of first wall materials in future fusion reactor. The first wall materials of future fusion reactor are mainly made of various types of low activation steel, so it's necessary to study the corrosion behavior of low activation steel in lead-lithium alloy. In this work, the corrosion behavior and corrosion rate of China low activation martensitic (CLAM) steel in the stagnant and flowing lead-lithium alloy were studied. In the stagnant and flowing lead-lithium alloy, the corrosion of CLAM steel was mainly dissolved corrosion, and no obvious oxide particles were found on the surface of CLAM steel. Elemental analysis results showed that the dissolution corrosion on the surface of CLAM steel was mainly caused by the dissolution and diffusion of Fe and Cr. Due to the promotion of flowing factors, the degree of thinning of CLAM steel was greater in the flowing lead-lithium environment, and a few corrosion pits were formed locally. As the flow velocity increases, the depth of the corrosion pits increased, and the overall thinning thickness increased. In addition, as the ambient temperature of flowing lead-lithium increased, the thinning rate of CLAM steel increased.

Keywords

China low activation martensitic (CLAM) steel, corrosion, lead-lithium alloy.

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PS3-60 ABSTRACT-8e61
D. Material Engineering for FNT

Investigation of applicability of long-distance LP system for Li target diagnostics in Fusion Neutron Source

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The beam target for generating a fusion neutron field in the Advanced-Fusion Neutron Source (A-FNS) is currently designed as a one-side free surface liquid lithium (Li) jet flowing along a vertical concave wall. Characteristics of its thickness fluctuation were diagnosed in the EVEDA (Engineering Validation and Engineering Design Activity) Lithium Test Loop (ELTL), which has almost the same scale as the actual target excepting its channel width. In the diagnostics, the Laser Probe (LP) system was used. However, in an actual fusion neutron source, the diagnostics need to be conducted from over 10 meters away with no condenser lens at a downstream point from the final mirror for directing the incident laser to the Li surface from the perspective of Li vapor and radio activation. Whereat, the long-distance LP system for the actual Li target was conceptually designed for the IFMIF, and its precision error was also verified using a specular-reflection object and a water loop. In this previous study, the precision of measurement was evaluated as sufficiently high compared with the required limitation of surface variation (±1 mm). Then, in this study, we aim to verify the possibility of long-distance measurement of thickness variation for the free-surface Li flow. In the measurement, the incident laser head was set at a distance of about 10 m from the Li surface. And, while the previous measurement setup employed the focusing lens system with a focal length of 300 mm, some focusing lenses with longer focal length were used in this study. As a consequence, the Li surface variation was successfully detected and measured though the signal intensity of the reflected laser became low. Furthermore, it was confirmed that the impact of focal lengths on the detected signal was small.

Keywords

Liquid Li target, surface diagnostics, Long-distance LP system.

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PS3-61 ABSTRACT.-8f74

D. Material Engineering for FNT

Engineering design status of ifmif-dones high energy beam transport line and beam dump system inside the tir and riz

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[6]KIT
[7] Fusion for Energy

IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO Oriented Neutron Source) is a fusion materials testing facility that is currently being designed under the framework of a work package of the EUROfusion Consortium. It will use a 40 MeV at 125 mA deuteron beam to generate a high neutron flux through Li(d,xn) nuclear reactions in a liquid lithium target. The High Energy Beam Transport line (HEBT), the final system of the IFMIF-DONES accelerator, is responsible for the guidance and shaping of the beam. Additionally, during commissioning periods, the HEBT is also responsible for diverting the beam, through the Beam Dump Transport Line, to the Beam Dump for testing purposes. The HEBT is spread along different rooms and zones: the Accelerator Vault, the Radiation Interface Zone (RIZ), and Target Interface Room (TIR). The engineering design of the HEBT components situated within the TIR and RIZ has been updated to satisfy new requirements, with a focus on ensuring the durability of the Fast Safety Isolation Valve (FSIV) under intense irradiation. These modifications include the relocation of the FSIV outside the TIR, the configuration of an inert enclosure for the FSIV, the placement of a lead shutter close to the wall, and the addition of local shielding to protect actuators placed inside the RIZ and TIR. This work describes the current status of these TIR and RIZ engineering design, including radioprotection, commissioning and maintenance plan, beam diagnostics devices, beam dynamics and new remote handling approaches, as well as the layout and integration of the required components along the beamline. The TIR and RIZ are critical areas for IFMIF-DONES, and their design and operation must ensure maximum efficiency, reliability, and safety. The updated design addresses potential issues and enhances the facility's overall functionality.

Keywords

IFMIF, DONES, Early neutron source, Linear particle accelerator, High energy beam transport line, Remote handling, Modular design, argon enclosure.

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PS3-63 ABSTRACT.-99af
D. Material Engineering for FNT

Strength Evaluation of Oxide Scale on FeCrAl Alloys by Micro Double-Notch Shear Test

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In the D-T fusion reactor, a design of the liquid LiPb breeder blanket has been proposed, where LiPb acts as a coolant and a tritium breeder material. When using liquid LiPb as the coolant material, the compatibility of the piping material is the most important issue that directly affects the system's lifetime. We propose the use of FeCrAl alloys, which self-form a protective alumina scale on the surface, as a piping material behind the blanket. From the material application viewpoint, checking the oxide film's resistance against frictional stresses from the flowing LiPb is very important. In this study, the resistance of the alumina scale formed on the surface of FeCrAl-based alloys by pre-oxidation treatment to shear stress was clarified by micro double-notch shear (DNS) tests and microstructural observations.

The specimens were hot-extruded FeCrAl-ODS alloy (Fe-12Cr-6Al-0.5Ti-0.4Zr-0.5Y_2O_3, hereafter, in wt.%) and Kanthal®APMT (Fe-21Cr-5Al-0.08C-0.7Si-0.4Mn). Strip-shaped samples were cut from each specimen and pre-oxidized at 1000°C for up to 100 hours. The oxide/alloy interface was exposed by cutting and polishing the samples, and DN-machined pillars were fabricated by the focused ion beam. An indentation shear test was performed on the pillars with a flat indenter.

The average values of the rupture strength of the interface measured were 1037 MPa for APMT and 1162 MPa for ODS after 64 h of pre-oxidation treatment: both are comparable to or higher than the UTS of the base metal alloys. The rupture strength tends to increase with increasing pre-oxidation time.

Keywords

FeCrAl alloys, LiPb blanket, oxide scale, shear strength test, micropillar technique.

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PS3-64 ABSTRACT-5895
E. Vacuum Vessel and Ex-vessel Systems

Electromagnetic Analysis on Conceptual Design of the EU DEMO CE equatorial Laucher

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The EU DEMO Tokamak is foreseen to be equipped with Electron Cyclotron (EC) launching systems for plasma heating, MHD control and thermal instability suppression. Up to six launchers will be installed into equatorial ports with the aim to inject a maximum of 130 MW millimeter wave power at dedicated positions into the plasma. The integrated design of the EC launcher is made of two equatorial port plug modules mounted into the equatorial port extensions of the vacuum vessel that mainly serve as housing for the eight optical system´s mirrors.

To guarantee reliable operation of the launcher, in the latest year, a structural system has been designed which provides secure fastening and alignment of the optical components, sufficient heat dissipation and protection of sensitive areas against nuclear loads from neutrons by adequate shielding capability.

To prove the fastening concept of the port plug modules and mirrors, a set of mechanical loads acting on these components is essential. EM loads caused by plasma instabilities are major contributions of mechanical load set for components close to the plasma.

The present paper describes the EM analysis carried out to support the assessment of the pre-conceptual design solution. A dedicated FE model was developed using ANSYS® solid236 element type (edge-based formation). Transient electromagnetic analysis was performed under centred plasma disruption event. Total force and moment due to Lorentz and ferromagnetic loads on the EC launcher port plugs and mirrors were computed. The resulting EM loads can be used for the preliminary design assessment.

Keywords

EU-DEMO, EC equatorial Launcher, Integrated design, Electromagnetic analysis.

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PS3-65 ABSTRACT-6660
E. Vacuum Vessel and Ex-vessel Systems

Preliminary Assessment of Vertical Motions of HL-2M Vacuum Vessel during 1MA Operation

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Abstract: The HL-2M vacuum vessel is supported at the mid-plane of the outboard of the torus by 5 rod supports, and designed to bear the dead weight and the electromagnetic forces loaded on the vacuum vessel at plasma disruption. Dynamic responses have been observed to occur after disruptions of 1.0MA plasmas operation. Firstly, the vacuum vessel is relatively flexible, so these events showed up first as vessel movements. Therefore the finite element model was developed to analyze the distribution of forces and stress of the vessel, and structural stress concentration position had been predicted in this dynamic response analyses. And then the preliminary measurements of displacements and forces, together with more robust vessel supports were progressively implemented and recorded by the triaxial accelerometer and eddy current displacement sensor, providing a satisfactory monitoring and control of the events. It was found that this mid-plane sliding support type of vacuum vessel was stable to bear electromagnetic forces 1MA plasmas operation. The vessel vertical motion and displacements were assumed to scale as Ip, Ip*Bt,the time of plasma current climbing and quenching during about 600 plasma operation shots.

Keywords

Vacuum Vessel；Plasma Disruptions；Dynamic Responses；Vertical Motions.

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PS3-66 ABSTRACT.- c699
E. Vacuum Vessel and Ex-vessel Systems

Electromagnetic Loads on the European DEMO divertor

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The paper deals with the problem of estimating the mechanical loads on the divertor of the European DEMO tokamak associated to electromagnetic transients as those experienced during the plasma disruption events. The analysis is referred to the current version of the divertor design (the so-called *Baseline Design*), based on a dual cooling circuit concept.

The transient analysis of the electromagnetic problem is performed by solving a 3D Magneto-Quasi-Static model, with a numerical Finite Element (FE)-based implementation in Ansys EMAG. Based on the symmetry and periodic boundary conditions, a toroidal sector of the DEMO Tokamak of 22.5 degree is analyzed using a hybrid modelling approach. Specifically, the model contains detailed geometric features of all the divertor subcomponents (cassettes, targets, liner, cooling pipes, etc..) while a coarser mesh model is employed for the major passive external components (vessel, blanket, rails, etc..). The active components (coils and plasma currents) are replaced by suitable equivalent current sources and by a known equilibrium magnetic field. The analysis is carried out with reference to one of the most challenging events for the divertor, namely a downward Vertical Displacement Event (VDE). The Lorentz forces and moments associated to the whole divertor and to its main subsystems (cassettes and cooling) are evaluated, assuming different conducting and magnetic materials.

The computed resultant forces and moments suggest that the maximum loads on the whole divertor assembly and on the cassette body are related to the vertical component of the force and moment. The maximum loads imposed on the cooling system (cooling pipes, vertical targets) are associated to the radial (forces) and vertical components (moments). The results do not significantly change even if EUROFER steel is used for the entire Cooling System components or electrical insulation is applied to the supports between the targets and the cassette body.

Keywords

DEMO, Divertor, Electromagnetic loads, Vertical Displacement Event.

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PS3-67 ABSTRACT.- 71be

E. Vacuum Vessel and Ex-vessel Systems

Progress and challenges of the ECH transmission line design for DTT

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The design of the Transmission Line (TL) as a part of the Electron Cyclotron Heating (ECH) system for Divertor Tokamak Test facility (DTT) is approaching the conceptual design maturity. With an ECH system of 16 MW installed for the first phase and with a total of 32 gyrotrons (170 GHz, ≥ 1 MW, 100s) the TL design is undertaking the challenge of an evacuated Multi-Beam TL (MBTL) concept to deliver the large number of beam lines from the gyrotron hall to the torus hall buildings. The system is organized in 4 clusters, each of them including 8 beamlines. The routing consists of single-beam TL section used to connect the gyrotron output to a beam-combiner mirror unit, a MBTL and a beam- splitter mirror unit to connect to the ex-vessel optics and launchers sections located in the equatorial and upper ports of one sector, for a total of 4 DTT sectors. The TL mirrors will be actively water cooled to cope with the heat load in long pulses due to the high power incident radiation, with the possibility to include advanced concepts for the cooling design compatible with additive manufacturing technology. The characteristics of the system and its components are presented, showing both the progress of the adopted solutions and the current design. Since the main challenge of this TL is to maintain the overall losses below 15%, in this paper we present the expected ohmic losses, spill-over effects, and beam coupling simulations evaluating losses given by high order modes. We describe how the effects have been estimated with electromagnetic simulations and how losses could be mitigated, since TL efficiency could significantly drop due to the presence of non-idealities, like the deformations of mirrors surface ascribed to the microwaves heat loads and possible misalignments and aberrations effects occurring along the line.

Keywords

Electron Cyclotron, Transmission line, optics, electromagnetic simulations.

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PS3-68 ABSTRACT-7370
E. Vacuum Vessel and Ex-vessel Systems

Core plasma thomson scattering diagnostic design

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The Core Plasma Thomson Scattering diagnostic (CPTS) is a diagnostic system aimed to measure the electron temperature (Te) with an accuracy of 10% and electron density (ne) with an accuracy of 5% at the core of the ITER Tokamak plasma with a resolution of 60mm to 20mm. For this purpose, the CPTS system will inject laser light (1064nm Nd:YAG laser) into the vacuum vessel during the plasma pulse, which will be scattered by the electrons in the plasma due to the Thomson Scattering phenomenon. The scattered light will be collected and analysed to determine the temperature and density at the plasma core.

IDOM, under a contract with Fusion for Energy, has developed the full CPTS diagnostic for which a complete, detailed and robust preliminary design is proposed which is compliant with the required level of diagnostic performance and resolution as well as the stringent nuclear regulations.

The in-vessel collection optics is a particularly challenging system due to the compromise between ensuring a broad optical path to collect as many photons as possible to maximize the signal to noise ratio but at the same time ensuring a certain level of shielding to keep the radiation dose levels at the Interspace at reasonable values. Additionally, the harsh environment to which the in-vessel components are subjected (high thermal, EM and seismic loads) is a driver for the optomechanical design. In particular for the plasma facing components, a radiofrequency mirror cleaning technology has also been developed.

Ex-vessel, technical challenges involve among others the need for active alignment provisions under a radiated environment for the injection and collection optics to absorb relative displacements between the in and ex-vessel systems; the generation, relay, alignment and injection of high power lasers along a 40m path or customized Polychromator and Data Acquisition system designs.

Keywords

ITER, Thomson Scattering, diagnostic, in-vessel components, ex-vessel components, design, RF mirror cleaning.

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PS3-69 ABSTRACT-7b4d
E. Vacuum Vessel and Ex-vessel Systems

Shutter development study for the ITER Visible Spectroscopy Reference System

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The ITER Visible Spectroscopy Reference System (55.E6 VSRS) is going to be one of the first optical diagnostic systems in the ITER tokamak chamber, aiming at the measurement of the line-averaged visible continuum emission of the core plasma. Given that the main diagnostic measurement lies within the visible band of the spectrum, the in-vessel diagnostic mirrors are inherently sensitive to contamination originating from the First Wall Conditioning, especially during Glow Discharge Cleaning. In order to protect the mirrors and extend the diagnostic lifetime, an in-vessel shutter is necessary to close the optical path.

The aim of this paper is to introduce the shutter concept presently selected for the VSRS diagnostic system (ex-vessel pneumatic actuation, rotary feedthrough, in-vessel drive shaft, and couplings), from the first decision matrix and optioneering up to the detailed design of the current option. In particular, ITER-dedicated challenges such as Remote Handling compatibility, UHV environment, radiation and disruption loads as well as requirements for fail-open safety mechanisms will be discussed in-depth.

Keywords

VSRS, Visible Spectroscopy Reference System, Spectroscopy, Shutter,.

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PS3-70 ABSTRACT-839e
E. Vacuum Vessel and Ex-vessel Systems

Design of the hybrid joint using low and high-temperature superconductors

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KSTAR has successfully conducted plasma experiments for nuclear fusion energy research. The KDEMO project, which is the next step towards nuclear fusion demonstration, is also planned, requiring a Tokamak device capable of generating a larger magnetic field. As a result, larger-scale superconducting coil magnets are needed, with superconducting CICC (Cable-In-Conduit Conductor) up to several kilometers long being used. However, current technology can only handle lengths of approximately 600-800 meters for CICC, requiring joints to connect multiple segments together. Lap joints have been widely used. For improved performance, the ENEA joint was developed in EDIPO, and ITER has designed splice joints for the central solenoid, inspired by the ENEA joint. This paper presents a hybrid joint design using second-generation high-temperature superconducting tape for joint performance improvement and development of a joint for SUCCEX's CICC. The hybrid joint was benchmarked on the splice joint and designed to connect LTS-HTS sub-cables. To analyze the characteristics according to the design of the hybrid joint, a mockup was fabricated using NbTi and GdBCO conductors.

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Keywords

Hybrid joint, joint, HTS tape.

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PS3-72 ABSTRACT-9643
E. Vacuum Vessel and Ex-vessel Systems

Overcoming reionization issues in the diagnostic neutral beam for the RFX-mod2 experiment

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Consorzio RFX (CNR, ENEA, INFN, Università di Padova, Acciaierie Venete SpA)

Diagnostic Neutral Beam Injectors (DNBI), through the combined use of Charge Exchange Recombination Spectroscopy (CHERS) and Motional Stark effect diagnostics (MSE), are a well-known tool to access important information about magnetically confined plasmas, such as radial profiles of ion temperature, of ion flow, of impurity content and of intensity and direction of the magnetic field. For this purpose, a DNBI was installed and operated in the RFX-mod experiment, which was designed to confine plasma mainly through the Reversed Field Pinch configuration. The diagnostic capability of the installed DNBI was limited by the high recombination of hydrogen or deuterium at the walls: the consequent flux of H2/D2 towards the DNBI led to significant beam loss through reionization in the DNBI duct and at the plasma edge. With the upgrade of the machine to RFX-mod2 and the refurbishment of the DNBI system, the beam loss issue is intended to be overcome through a proper design of the duct geometry and of the related pumping system. The paper will provide a characterization of the past issues, present the design of the improved duct system, estimate beam losses and consequently the preliminarily expected light emission conditions to be exploited by CHERS and MSE.

Keywords

Diagnostic neutral beam, beam reionization, vacuum pumping, beam emission spectroscopy.

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PS3-73 ABSTRACT-a070
E. Vacuum Vessel and Ex-vessel Systems

Technology challenges and integration of the plasma position reflectometer in RFX-mod2

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Consorzio RFX

Plasma position reflectometry (PPR) is a microwave radar technique that aims providing information on the plasma column position and shape by monitoring the edge density profiles at several poloidal positions in Tokamaks. The PPR in-vessel diagnostic equipment is compact, non-invasive and intrinsically resistant to neutron radiation. For this reason, after some initial tests on ASDEX-Upgrade, a partial PPR system is being designed on DTT in view of DEMO, where long pulses and high neutron fluxes are likely to jeopardize the long-term reliability of the internal magnetic sensors.

On RFX-mod2 (R=2.0 m, a=0.49 m), the upgraded version of the previous RFX-mod device, a simplified PPR scheme, consisting of four biostatic reflectometric units, has been conceived to test some of the open issues related to a plasma position control on a fusion reactor. However, its integration in the machine had required different innovative solutions.

This contribution is focused on the technological aspects linked, in particular, to the installation of the antenna pair and the waveguide system in the high field side section of RFX-mod2.

Waveguides, insulated through the application of a ZrO painting, will be routed in between the vacuum vessel and the conductive shell to a vertical port. The severe constraints in terms of physical space available guided the antennas design: a hoghorn antenna model was first numerically modeled and, due to the complex geometry, produced through metal additive manufacturing; then, a post-production surface treatment allowed achieving a surface with characteristic roughness and conductivity comparable to traditional manufactured antennas. Several bench tests are underway to evaluate the overall performance of the system.

Peculiar issues, which are being addressed on larger fusion devices in the design of PPR subsystems in the high field side, are finally briefly reviewed in the light of the experience gained during this specific implementation.

Keywords

Reflectometry, In-vessel diagnostics, Additive Manufacturing.

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PS3-74 ABSTRACT-a907
E. Vacuum Vessel and Ex-vessel Systems

EU-DEMO Vacuum Vessel Port Closure Plate fastening developments

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The EU-DEMO baseline design for the vacuum vessel (VV) currently provides for 48 openings – the so-called VV ports. These are foreseen to accommodate port plugs containing components, such as front ends for diagnostics and heating systems, first wall protection limiters and the first stages of the VV vacuum system. During operation of EU-DEMO all these openings have to be closed vacuum tight by so called port closure plates (PCP) allowing for the required high vacuum environment in the VV. Numerous service lines have to be integrated in the ports, many of which may need to penetrate the PCPs. The VV ports also provide access into the VV for scheduled and unscheduled maintenance of the in-vessel components. Opening and re-closing of the PCPs within a reasonably short time is a requirement for EU-DEMO, that shall demonstrate the viability of electricity production by fusion.

In the field of high vacuum technology, connections using a large number of bolts are an established method for fastening PCPs. From the experience of existing fusion experiments, some disadvantages of bolted flanges are known. In particular, these are the rather long duration of the opening process and the possible fretting.

This paper summarises and describes the current status of the development of possible alternative PCP fastening concepts with improved characteristics. An approach to collaborate with industry to take advantage of state-of-the-art applications and ensure the transfer of knowledge is also presented.

Keywords

EU-DEMO, DEMO, vacuum vessel, port closure plate, port plug, fastening, fixation, remote maintenance, sealing.

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PS3-75 ABSTRACT-f940
E. Vacuum Vessel and Ex-vessel Systems

ITER Visible Spectroscopy Reference System design development status

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The ITER Visible Spectroscopy Reference System (55.E6 VSRS) measures line-averaged visible emission along tangential line of sight through the core of the plasma, and has an intersection with the diagnostic neutral beam (DNB). Signals are acquired using a combination of four optical instruments, each tailored to specific combinations of spectral and temporal resolution. From this data the average ion charge (Z_{eff}) is determined as the primary output. Z_{eff} is an important control parameter in fusion and indicates the level of impurities in the plasma. The VSRS system has a back-up role for electron density (n_e) and supports other diagnostic systems in the measurement of toroidal velocity, ion temperature, fractional content, influx, and relative concentration of intrinsic (H, D, T, He) and extrinsic species (Be, C, N, O, Ne, Ar, Kr, Cu).

The present paper provides an overview and describes the current design status of the VSRS. This comprises the diagnostic system opto-mechanical design, located in and near the ITER tokamak, the measurement instruments located in a remote diagnostic room, and the supporting systems: built-in calibration, active alignment, shutter and plasma cleaning. In particular, ITER-dedicated implementation challenges such as Remote Handling compatibility, magnetic field, UHV environment, radiation and disruption loads will be discussed in-depth. Currently the diagnostic system design is at final design level for the parts and systems outside the tokamak and at preliminary design level for the in-vessel components. This design is supported by various analyses comprising: optical analyses for tolerances, performance under operational load and stray light, performance analysis for the measurements and supporting analyses for the plasma cleaning system.

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

Keywords

VSRS, Visible Spectroscopy Reference System, Spectroscopy, Calibration, Active Alignment, Shutter, Plasma cleaning, First Mirror.

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PS3-77 ABSTRACTeaf9
F. Nuclear System Design

Updates of the Removable Biological Shielding Blocks inside of the Test Cell, part of the Test Systems of IFMIF-DONES

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The International Fusion Materials Irradiation Facility - DEMO Oriented Neutron Source (IFMIF-DONES) is a facility which is designed under the framework of the EU fusion roadmap. It is going to be an essential irradiation facility for testing and qualifying candidate materials under severe irradiation condition of a neutron field having an energy spectrum like the one present in a commercial fusion power reactor. The material specimens are irradiated in a containment structure named Test Cell (TC), which is part of the Test Systems (TS). The first protecting „barrier" against irradiation affecting the surrounding of the TC, are the Removable Biological Shielding Blocks (RBSBs). These 8m+ high, +100 tonnes elements, which are formed by a stainless steel liner filled in by heavy concrete, need to be remotely handled as after the first experiments they will get dose rates above the hands-on limit. Irradiation will also result into a large nuclear heat power deposited in the shielding blocks, which need to be actively cooled by a system of embedded pipes to control the temperature. In this paper the updated design of the RBSBs is described, including the latest achievements and proposals for feasible manufacturing, lifting and positioning possibilities of the blocks inside of the TC respecting the given tolerances.

Keywords

IFMIF-DONES, RBSB, Test Cell, Remote Handling, heavy concrete.

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PS3-78 ABSTRACT-7a61
F. Nuclear System Design

Structural assessment of the contamination protection structure placed on the top of the DEMO bioshield roof

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In the DEMO power plant design, the bioshield roof above the tokamak is a steel structure with radial beams and concrete inserts that can be removed to access the upper ports. On the roof a structure will be placed that subdivides the large volume into 16 containment cells above each upper port and protects other areas against a potential release of any radioactive materials from inside the reactor during maintenance. This structure is called contamination protection structure (CPS). It also provides support to the cask, which is used to transfer the heavy breeding blanket segments to the hot cell building. The structural design of the CPS and the bioshield roof of DEMO has been studied using the Ansys APDL environment. Parametrized FE models have been built to that end.

The results of the linear static analysis show the stress in CPS as well as the bioshield roof is in an acceptable range. The linear buckling analysis shows however, that the CPS may lose stability because of buckling, especially in the region of radial beams. The proposed design modifications yield a structure that is sufficiently strong and rigid to withstand all loading conditions. The observed stress concentrations are local and concern mostly the connections between elements. These issues may be solved by a more precise modelling at further design stage to better understand structure's response to the applied loads. Moreover, the analyses revealed that the actual loads due to the remote handling structures in the maintenance hall acting on the bioshield roof structure are lower than what had been assumed during its preliminary design of the bioshield roof [1]. A substantial reduction of the size of the bioshield roof beams and girders is possible.

[1] Ł. Ciupiński, et al. FED 136, Part B, 2018, pp. 1461-1466

Keywords

Contamination protection structure, bioshield roof, DEMO design.

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PS3-79 ABSTRACT-169e
F. Nuclear System Design

An overview of steady-state electrical loads in the DEMO fusion power plant

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DEMO (DEMOnstration power plant) is ITER successor. With the transition from ITER to DEMO, fusion will go from a science-driven, lab-based experiment to an industry-driven and technology-driven programme. A key aim of DEMO project is the production of electricity, although not at the price and at the quantities of commercial power plants, providing a bridge between the current research-oriented tokamak facilities and the future commercial fusion power plants. DEMO is expected to start operations in the 2050s and is designed to generate around 1 GW of gross electrical power. It will be larger and more complex than current research reactors, incorporating various innovative features and technologies to improve performances and safety.

The electrical system of DEMO is the beating heart of the fusion power plant, and its design is a fundamental step for the project realization. The starting point for the electrical system design and sizing activities is the identification and characterization of the electrical loads within the facility and at service of DEMO sub-systems, such as the plasma heating system, the magnetic field coils, the balance of plant, and the various support systems required for the safe and reliable operation of the plant. Their power requirements, duty cycles, and other relevant features from the electrical and safety point of views must defined to obtain a comprehensive overview of the whole electrical system.

The present work includes the most recent results of long-term research activities linked to the continuous updating of the so-called Electrical Loads List of DEMO. This activity is preliminary to all the sizing and design tasks on steady-state electrical system, and it is strongly linked to the progresses of all the design activities of DEMO subsystems. The paper provides an overview on the principles, the references and the rationales adopted for the identification of the steady-state loads.

Keywords

DEMO Plant Electrical System, steady-state loads, electrical load list, Balance of Plant, power Flow.

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PS3-80 ABSTRACT-8458
F. Nuclear System Design

Approach to Spatial Integration on a Novel & Complex Major Project – STEP Concept Tokamak

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The purpose of this work was to enable the development at pace of a commercially relevant Spherical Tokamak Concept. Taking the design from a table of parameters and idealised geometry to an integrated concept design representation in 16 Months.

STEP is an ambitious programme, tasked with delivering a prototype plant by 2040 as a pathway to commercial fusion energy. Tackling the integration of a system that itself is based on emergent technologies, significant assumptions (e.g. Plasma Physics) and uncertainties is a significant challenge whilst maintaining acceptable levels of commercial relevance and feasibility. This paper shares the developments and approaches to collaboration and communication taken, including the management of a graphical representation (CAD) to overcome this challenge.

Work carried out has resulted in a single point CAD (CATIA) model representing the current integration status including an appropriate approach to change control. Clear channels and methods of communication nurtured a successful collaborative and cross-functional integration approach. Transitioning swiftly from an externally driven approach to integration, to an inherent, self-orchestrating one across all product areas. This paper presents results as a single point Concept Design for the Integrated Tokamak, a fully consolidated management approach, technological integration challenges faced, and considerations for the future (post concept design).

Introducing a centralised design process, appropriate control and clear communication utilising CAD has been the key to maturing the STEP Tokamak Concept. Initial focus on facilitating collaboration and utilising this environment as a 'sandbox', enabled early establishment of a 'space-claim' concept design in which to iterate. Staging the integration approach and CAD model fidelity developments through 'Identification, Realisation and Accommodation' built a resilient foundation to progress through the full design process. This approach has enabled early identification of not only where the integration challenges lay but also their significance, resulting in strategic deployment of valuable resource

Keywords

Spatial Integration, Fusion, Tokamak, STEP, Spherical, CAD.

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PS3-81 ABSTRACT-8517
F. Nuclear System Design

Thermal-hydraulic analysis and last process design update of the ITER Component Cooling Water System 1

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The purpose of this work was to enable the development at pace of a commercially relevant Spherical Tokamak Concept. Taking the design from a table of parameters and idealised geometry to an integrated concept design representation in 16 Months.

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Keywords

ITER, CCWS-1, BoP, Thermal hydraulics.

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PS3-83 ABSTRACT-97ee
F. Nuclear System Design

Nuclear analyses for the integration of ITER Equatorial Port 2

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The present work is devoted to nuclear analyses in support of the ITER diagnostic Equatorial Port 2 (EP#2) integration.

ITER EP #2 is a diagnostic port based on the long-modular Diagnostic Shielding Module (DSM) housing the following systems: Disruption Mitigation System (DMS) in DSM#1 and #3, core imaging X-Ray Crystal Spectrometer (XRCS) in DSM#2. Ensuring adequate radiation shielding is a major challenge since the diagnostic systems require several apertures from the Vacuum Vessel (VV) through the Port Interspace (PI) and up to the Port Cell (PC).

In the present study, a three-dimensional MCNP model of EP#2 has been developed, starting from the latest design available, and successively integrated into the reference 40° ITER C-Model. Comprehensive nuclear analyses have been carried out employing the D1SUNED v3.1.4 code based on the MCNP Monte Carlo transport code. Relevant nuclear quantities during and at the end of plasma operations have been evaluated: i.e., neutron and gamma fluxes and energy spectra along the port from the Diagnostic First Wall (DFW) up to the Bio-Shield Plug (BP), nuclear heating, neutron damage, helium and tritium production, and shutdown dose rate. This analysis allowed the identification of potentially critical areas, and therefore, the implementation of additional shielding options aimed at reducing the neutron streaming and mitigating the radiation field in the PI region. In this paper, the results of the analyses are presented and discussed. Some solutions to mitigate nuclear loads and to improve the shielding in PI area are proposed and their impact has been assessed. Finally, some recommendations for the optimization of the design of EP#2 are provided as well.

PURPOSE: The purpose of this study is to investigate the application of equipment qualification (EQ) to the Protection Important Components (PICs) of the ITER, based on the EQ technologies and practices experienced from the commercial nuclear power plants.

Keywords

ITER, Diagnostics, Neutronics, MCNP, Activation, Port, Integration.

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PS3-87 ABSTRACT-a049
F. Nuclear System Design

Neutronic Analyses for the IFMIF-DONES Test Cell and Adjacent Rooms

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This work presents neutronic analyses of the most irradiated rooms in the IFMIF-DONES facility. The rooms surround the IFMIF-DONES Test Cell (TC), where the radiation is produced due to the impinging of the accelerated deuterons on the liquid lithium target and delivered to the materials to be used in the DEMO fusion reactor. The TC is isolated at all sides by a very thick (3-6 m) radiation shield composed of removable concrete blocks and ceiling, as well as fixed walls and floor. However, these concrete blocks and walls have technical channels to operate the IFMIF-DONES facility. Radiation streaming from the TC source through these channels and the radiation environment created in the adjacent rooms have been analyzed. The primary Through Wall Beam Duct (TWBD) channel provides the entry of the deuteron beam to TC. Diagnostics of the deuterium-lithium (d-Li) target is performed by the In-Vessel Viewing System (IVVS) using the secondary TWBD channel. The IVVS head is placed inside Target Interface Room (TIR), adjacent to TC behind the 3-m wall upstream to the deuteron beam. On the opposite side of TC, the Neutron Beam (NB) tube and shutter have been designed to guide collimated neutrons from TC to Complementary Experiments Room (CER) through the 6.4-m thick wall. This paper includes the results of the neutronics modeling and analyses of the mentioned rooms (TC, TIR, CER) and corresponding components (TWBD, IVVS, NB-shutter). The CAD-based radiation transport is performed by the McDeLicious code – the MCNP code modification, and the SuperMC code used for CAD-to-MCNP model conversion. Neutron and photon fluxes, nuclear heating, neutron damage, and biological and absorbed dose rates have been calculated. In conclusion, it is demonstrated that the planned apertures of TWBD and NB-tube do not compromise the TC shielding performance, maintaining the nuclear safety of the IFMIF-DONES operation.

Keywords

Fusion neutronics, Nuclear heating, IFMIF-DONES, Test Cell (TC), Target Interface Room (TIR), Complementary Experiments Room (CER).

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PS3-89 ABSTRACT-a8ed
F. Nuclear System Design

Neutronics analysis and workflow for First Light Fusion reactor design.

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Initial design studies of the First Light Fusion reactor are being carrying out. This article reports on the development of an automated neutronics workflow which is a core part of the wider reactor design. The neutronics workflow utilises parametric CAD model creation using CadQuery, auto-mated conversion to DAGMC geometry with CAD-to-h5m and transport simulations in OpenMC. The neutronics model creation and simulation is driven via a Python API and this interface allows for scoping studies to be carried out with ease. Neutronics outputs include Tritium Breeding Ratio (TBR), heat deposited in the blanket and vacuum vessel heating, neutron spectra at a variety of locations and dose maps. It was possible to identify design spaces that could satisfy neutronics requirements. The automated workflow is the main discussion point of the article, to demonstrate the work- flow indicative neutronics outputs of a representative point design are also presented. However, further analysis in other disciplines would be neces- sary to assess the design candidates comprehensively. The workflow aims to provide an API that will be utilised to ultimatly provide surrogate models for the First Light Fusion systems code. A systems code will allow linking the neutronics outputs together with additional disciplines (e.g. engineering analysis) could allow for further design space refinement and is planned for future research. These initial design studies of the First Light Fusion reactor have thus paved the way for efficient and automated neutronics analysis ofdesign candidates.

Keywords

Neutronics,reactor,workflow,automatic,parametric.

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PS3-90 ABSTRACT-a8ed
F. Nuclear System Design

Design status of the neutrons and gamma-rays diagnostics for the Divertor Tokamak Test (DTT) facility

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The Divertor Tokamak Test (DTT) facility is a fully superconducting nuclear fusion device under design and construction in Italy, with the primary mission of testing power exhaust strategies in view of DEMO.

DTT will work with deuterium fuel and a maximum 2.45 MeV neutrons yield rate of ~ 1.5×10^{17} n/s is expected. The emission of 14.1 MeV neutrons due to triton burn-up will be up to few per cent of that of 2.45 MeV neutrons.

The paper presents an overview of the design status of the neutrons and gamma-rays diagnostics for DTT, which include:

- Time-resolved neutron yield monitors for the measurement of the instantaneous neutron emission strength: fission chambers, BF3 detectors, scintillators, single crystal diamond detectors distributed around the machine.
- Neutron activation system for the measurement of the absolute neutron yield: activated foils moved between a counting station and an irradiation location close to the plasma boundary by means of a pneumatic system.
- Neutron/gamma camera for the spatially-resolved measurement of neutron emission, runaway-induced bremsstrahlung radiation and gamma-ray emission from reactions between fast ions and plasma impurities: massive shielding block, located in front of an equatorial port, hosting a set of collimators equipped with neutrons/gamma-ray detectors (He-4 detectors, scintillators, single crystal diamond detectors).
- Time of flight spectrometer for high resolution measurements of the neutron emission spectrum along a collimated line of sight: plastic scintillator detectors, with a first detector S1 placed in front of the collimated neutrons beam and a second set of detectors placed at a known distance from S1 recording a fraction of the neutrons scattered in S1.

The discussion, with different level of detail depending on the specific diagnostic, focuses on establishing a reference design through the definition of components, interfaces, CAD drawings, detectors types, performances and calibration procedures.

Keywords

DTT, diagnostics, neutrons, gamma-rays.

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PS3-91 ABSTRACT-ae98
F. Nuclear System Design

Design of a THz interferometer system for Korea Superconducting Tokamak Advanced Research

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A single channel THz interferometer (THI) system has been developed to replace the 280 GHz millimeter-wave interferometer (MMWI) system [1] for Korea Superconducting Tokamak Advanced Research (KSTAR). The overall system configuration such as introduction of ~1 THz millimeter-wave probe beam, optics and signal processing electronics has been upgraded to improve the system robustness against the equipment aging and fringe jump issues. The new THz interferometer system consists of a heterodyne system with 1002.24 GHz probe beam and 10 MHz measurement intermediate frequency (IF). One of the challenges in the optics design is that the front optics, consisting of a plasma facing lens and beam splitter, had to be shared with a different diagnostic, multi-channel (8 radial and 24 poloidal channels) electron cyclotron emission imaging (ECEI) system [2] since the two diagnostics equipped with large aperture optics should be installed on a single diagnostic port. Here, a large aperture wire-grid beam splitter [3] is employed to separate the ~1 THz O-mode beam for the THI and 75-110 GHz X-mode beam for ECEI. The details of system design, initial commissioning results and laboratory test of the optical system with the wire-grid beam splitter will be presented in this paper.

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Keywords

THz, Interferometer, Density measurement.

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PS3-91b ABSTRACT-cbb2
F. Nuclear System Design

Thermal and structural analyses on different mirrors of the Multi Beam Transmission Line of DTT ECRH system

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Inside the research field of Nuclear Fusion, the Divertor Tokamak Test facility (DTT) [1] is an Italian project, aimed at investigating alternative power exhaust solutions under integrated physics and technical conditions that can reliably be extrapolated to the future nuclear fusion DEMOnstration power plant. In DTT, to achieve a large ratio value of the power crosses the separatrix, it is necessary to provide to the plasma supplementary heating power of 45 MW. One of the three heating systems is the Electron Cyclotron Resonance Heating. Its architecture is based on 4 clusters, each one constituted of 8 gyrotrons (Radio-Frequency source), 1 quasi-optical multi-beam transmission line (MBTL) and 8 independent antennae. The power, generated by the RF sources (170 GHz, 1 MW, 100 s pulse duration), is transmitted by a series of oversized flat and focusing mirrors and launched into the plasma by an antenna.

In the MBTL, the reflection of the 8 separate microwave beams on the same mirror surface heats the structure, being a fraction of beam power absorbed. So, during the RF operation, the mirror temperature increases, thus generating deformations due to thermal expansion that would result in lower transmission line efficiency. For this reason, the mirrors need to be cooled and carefully designed in terms of thermal and structural properties, to guarantee the required optical performances. The refrigerant fluid is the water and the maximum power to be disposed of is 13 kW with heat flux peaks in the range of 0.1-0.2 MW/m^2. In this work the conceptual design of different MBTL flat and parabolic mirrors and their cooling layouts is presented, resorting to thermo-structural finite element simulations. The presented analyses aim to individuate the deformed structures and reconstruct the deformed reflective surfaces necessary to include this effect in the evaluation of the transmission line losses.

Keywords

Thermo-structural analysis, Mirror, ECRH.

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PS3-92 ABSTRACT-5dc0
G. Safety Issues and Waste Management

Investigation on MELCOR code capabilities for the simulation of Lithium-Lead chemical interactions for fusion safety applications

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In the WCLL-BB liquid blanket concept, the tritium breeder is used in the form of liquid metal. A dedicated ancillary system provides the recirculation of the liquid metal enabling tritium extraction from the blanket and purification of the liquid breeder. Among the postulated accidental scenarios to be investigated regarding the PbLi system, a loss of liquid metal is one of the most critical for safety. As a consequence of this loss, chemical reactions might occur between the lithium contained inside the alloy, the air and the water. These reactions may lead to significant increases in temperature and pressure and to the formation of hydrogen inside the building; the evaluation of the impact of these events is essential to define future safety guidelines. Using MELCOR for fusion v.1.8.6., a first approach to investigate an out-vessel loss of PbLi and chemical reactions occurring between pure lithium and air is presented in this work. The release time and the amount of PbLi leaked in the assembly were analysed as basis of this work. The temperature and pressure trends were then investigated through the default MELCOR package for Lithium air chemical reactions; however, this approach does not permit to use PbLi as working fluid, since it works only with pure Lithium. Therefore, a new model of the reaction was developed in this work by means of MELCOR Control Functions, which allows the simulation of the reaction between lithium and air using PbLi as working fluid, in order to investigate a more realistic scenario also considering the thermal capacity of the PbLi. To identify the best approach, results from different approaches are compared and the limits of the code are commented in this work. Due to the assumptions and simplifications adopted, results are very conservative in terms of temperature and pressure reached in the system.

Keywords

MELCOR for fusion, WCLL-BB safety, Lead-Lithium chemical interactions, Multiphysics.

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PS3-93 ABSTRACT-6bf1
G. Safety Issues and Waste Management

Preliminary risk analysis for a CECE-CD system at ICSI Rm. Valcea

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From the ecological and workers safety point of view, both CANDU nuclear fission reactors and Tokamak-type fusion reactors, such as JET (Joint European Torus) or ITER (International Thermonuclear Experimental Reactor), require the purification of water with a low tritium content by separating it through combined processes of electrolysis, catalytic isotopes exchange and cryogenic distillation.

This paper provides a detailed overview of the method used by the authors to analyse the maximum characteristic risk of a pilot scale CECE (Combined Electrolytic Catalytic Exchange) process similar to those required for water detritiation of ITER, intended to be coupled to the cryogenic (CD) system on ICSI site.

The approach begins by carring out a PHA (Process Hazard Analysis) qualitative analysis of What If/Checklist type followed by the accident scenario analysis based on the multi-causality principle and uses the FT (Fault Tree) method to identify in a logical diagram the causes that can generate unwanted Top Events. Finally, the consequence of the worst credible accident is quantitatively evaluated and the package of preventive and protective measures necessary to ensure the safe operation of the installation is compiled.

Keywords

Electrolysis, cryogenic distillation, characteristic risk, tritium, process hazard analysis, fault tree.

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PS3-94 ABSTRACT-6c4d
G. Safety Issues and Waste Management

Radiation Data Analysis Tool (RaDAT)

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This abstract introduces the Radiation Data Analysis Tool (RaDAT). This software, developed under a specific task order of ITER project, was designed using Matlab and it is a GUI (Graphical User Interface)-based compact application intended to support expert and non-expert users in: (1) visualization analysis of 2D radiation maps, (2) radiation qualifications of electronic equipment and (3) Occupational Radiation Exposure (ORE) analysis.

It was certified for Protection Important Activities (PIA) and authorized to be used for equipment radiation qualification activities. Its main functionalities allow: (1) to visualize quickly 2D radiation maps (.vtr files) on different planes and positions, investigating radiation conditions in different spots of the Tokamak Complex Building (TCB); (2) to analyse automatically radiation conditions of lists of millions of equipment volumes (.stl files), providing the worst and best radiation conditions on Excel tables; (3) to calculate the total biological dose cumulated by a worker along a specific route, passing from a workstation to another, in the TCB. Another feature implemented in RaDAT is the Flexibility Radius (FR) option, which could be applied to the volumes, extending them by this FR, in order to take into account any possible relocation in proximity of the nominal installation spot.

RaDAT allows reducing the analysis time considerably. It is capable to analyse one radiation map, finding the worst and best radiation conditions within a single volume (e.g.: an electrical enclosure) in ~ 50 ms, using a standard office laptop, reducing the analysis time by a factor ~1000, compared to a manual analysis approach on standard visualization tools.

Keywords

Fusion energy, radiation analysis, equipment qualification, occupational radiation exposure, software development, matlab.

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PS3-95 ABSTRACT-131d
G. Safety Issues and Waste Management

Development of RELAP5/MOD3.3 for dynamic tritium transport analysis and its application to the COOL blanket

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Achieving tritium self-sufficiency is a critical requirement of fusion reactors. Meanwhile, tritium is a main radioactive source term and may cause safety hazards in case of accident. It is necessary to understand the tritium transport process and estimate the relevant quantities in components and systems. The tritium transport analysis module is integrated into RELAP5/MOD3.3 to obtain a multiphysics system-level code. In this code, the temperature, flow and material fields are coupled to solve inefficiency owing to the manual definition of thermal conditions in the previous system tritium analysis code, which enables dynamic tritium transport analysis in time and space. The capability of generic modeling is extended to provide various tritium transport processes to fit the different geometric and material configurations of the components, which allows refined node division with the consideration of inhomogeneity. The supercritical CO_2 cOoled Lithium-Lead (COOL) blanket, an advanced blanket candidate for the Chinese Fusion Engineering Testing Reactor (CFETR), is selected as an application object of the code. The COOL blanket and its system are modeled with the modified RELAP5 to study dynamic tritium transport behavior at the system level during steady states and plasma pulse operation. The temperature and pressure of PbLi and S-CO_2 are obtained. Detailed dynamic tritium performance is assessed, including concentration, permeation, inventory and extraction. This work provides a useful simulation tool for the design and safety analysis of fusion reactors. Its application on safety assessments will be carried out in the next step.

Keywords

Ritium transport, Modified RELAP5, COOL blanket.

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PS3-96 ABSTRACT-81f5
G. Safety Issues and Waste Management

Contamination assessments of demo wcll breeding blanket primary heat transport system

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In water-cooled fusion reactors, the assessment of the primary system contamination is essential for waste management, machine availability, occupational radiation exposure, and radiological hazard determination. The primary cooling water is not only directly activated by the intense neutron field but is a contamination vector for a significant variety of gamma emitters with short to long decay half-lives. Corrosion products can be activated in the regions of the circuit under neutron flux and then released in the cooling water.

In the EU-DEMO fusion power plant equipped with the Water-Cooled Lithium Lead Breeding Blanket (WCLL-BB) concept, the primary coolant undergoes intense neutron fields in the first wall and the breeding zone regions of the blanket. Activated Corrosion Products (ACPs) are then formed, released into the water, transported in the cooling loop and finally deposited onto the ex -vessel surfaces of the Primary Heat Transport System (PHTS), where working personnel are susceptible of being radiologically exposed.

This work addresses the complete assessment of ACPs in the WCLL-BB-PHTS of EU-DEMO. The simultaneous and multi-physical processes behind the ACP formation are tackled using OSCAR-Fusion, a comprehensive tool developed by CEA to assess contamination in fusion nuclear reactors. The whole system is modelled with zero-dimensional nodes with assigned geometrical, thermal-hydraulics, material and chemical parameters. Mono-energetic homogenized activation reaction rates calculated with MCNP are given to the regions exposed to the neutron flux. Results are provided in terms of mass and activity inventories of ACPs as deposit and inner oxide layers of components (pipes, heat exchangers, pumps…), ions in solution, particles in suspension, and filters trapping.

Mobilizable inventories such as ions, particles and deposits are important source terms in accidental scenario evolutions, while the whole activity inventory constitutes the main long-term gamma emitting source for dose rate determination in the tokamak building rooms housing the PHTS equipment.

Keywords

EU-DEMO, Activated Corrosion Products, Breeding Blanket, PHTS.

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PS3-97 ABSTRACT-8a0f
G. Safety Issues and Waste Management

Ex-vessel LOCA analysis for the EU-WCLL TBS using a fusion-adapted MELCOR model

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Fusion for Energy (F4E) are developing the European Water-Cooled Lithium Lead (EU-WCLL) Test Blanket System (TBS) to demonstrate tritium breeding technology within the ITER fusion tokamak. ITER is specified as French Nuclear Facility INB-174 and demonstrating the safety of the EU-WCLL TBS is necessary for the development and licensing of this system. Deterministic accident analysis is an essential part of the safety demonstration.

A fusion-adapted MELCOR model of the EU-WCLL TBS was created by the University of Sapienza, Rome using the same methodology previously reported for helium-cooled Test Blanket Module (TBM) concepts: Helium-Cooled Lithium Lead (HCLL) and Helium-Cooled Pebble Bed (HCPB). Jacobs have further developed the model to address the design evolution of the EU-WCLL TBS. Validation of the MELCOR model is not currently possible, due to insufficient suitably representative integral effects tests and experiments. However, activities have been undertaken to build confidence in the modelling and the accident analyses performed. These include steady state and transient analyses, comparison with finite element analyses and sensitivity studies for accident scenarios.

The updated MELCOR model of the EU-WCLL TBS is being used to analyse a number of accident scenarios, including Loss Of Coolant Accidents (LOCAs) due to the rupture of a cooling pipe, either inside or outside the Vacuum Vessel. The analysis is required to assess the thermal-hydraulic response of the systems, decay heat removal from the TBS, and any radiological releases. The development and application of the MELCOR model to analyse an ex-vessel LOCA originating in the high-pressure coolant pipework within the vertical shaft is reported together with preliminary results of the accident analysis.

The views and opinions expressed in this paper do not necessarily represent those of the ITER Organisation.

Keywords

Water-Cooled Lithium Lead, Test Blanket System, MELCOR, LOCA, accident analysis.

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PS3-98 ABSTRACT-9dbf
G. Safety Issues and Waste Management

Neutron Activation Analysis of CNS-I Corrosion Products in Fusion Reactor

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Neutron activation of corrosion products (CPs) is the dominant radiation hazard of water-cooled loops of fusion reactor, and contributes most of the Occupational Radiation Exposure (ORE) during normal operation and maintenance. For meeting the demand of low activation, recently a new kind of steel CNS-I (China Nuclear Steel-I) has been developed in China. In this paper, the performance of CNS-I on ACPs (activated corrosion products) source term was analyzed for the first time. The corrosion rate of CNS-I under the operation condition of International Thermonuclear Experimental Reactor (ITER) was measured using a high temperature flowing water corrosion experiment loop. Then based on the model of ITER LIM-OBB water-cooled loop, some evaluation and comparison between the above low-activation steel CNS-I and the traditional steel SS316 were made from the point of view of ORE produced by ACPs. The results showed that CNS-I produced higher dose rate of ACPs than SS316 during the operation phase of the reactor due to its bad corrosion-resistance, which is related to the CPs mass of the steel. But during the shutdown phase of the reactor, CNS-I presented advantage on ORE decrease due to its good activation-resistance, which is related to Co element content in the steel.

Keywords

Neutron activation, CNS-I, ACPs, Fusion reactor.

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PS3-99 ABSTRACT.- ab2e
G. Safety Issues and Waste Management

Preparation of activated carbon from biofibers for capture of tritium containing compounds

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Until year 2009 carbon fibre composite materials have been used as plasma facing materials in the vacuum vessel of the Joint European Torus and are considered as divertor materials for other fusion devices, like stellarators. However, it was determined, that retention and accumulation of the fusion fuel, deuterium and tritium takes place. Among the proposed solutions, an introduction of the metallic coating on the divertor materials has been introduced [1].

However, it is expected to have tritium containing materials in the fusion reactors, therefore detritiation methods are developed [2]. In order to capture the released gaseous compounds, an activated carbon material could be used.

In the present research a method of preparation of activated carbon for capturing [3] tritium is developed and tested. As source of carbon is biofibers from sheep wool, while the absorption effects are investigated with infrared spectrometry.

A gas transfer line with incorporated activated carbon filter and bubbler of tritiated water is connected to Bruker Vertex 70v infrared spectrometer. The infrared spectra registered in the range of 600-4000cm^{-1}, resolution ±4 cm^{-1}, 32 spectra per measurement. After sorption, the carbon samples oxidized and the amount of accumulated tritium measured with liquid scintillation method. Total amount of accumulated tritium gives information about application of activated cardon filters for tritium capture during detritiation of plasma exposed materials.

Research performed within the European Regional Development Fund 1.1.1.1 Practical Orientation Research 4th round project No. 1.1.1.1/20/A/155 "Development of novel and innovative composite materials with enhanced sorption properties from renewable biological natural resources available in the Republic of Latvia for commercial air purification filtration systems".

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Keywords

Tritium, activated carbon, infrared spectrometry.

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PS3-100 ABSTRACT-ae68
G. Safety Issues and Waste Management

Investigation on dust resuspension under in-vessel loss of coolant accident of fusion reactor

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Large amounts of radioactive dust will be produced by energetic plasma–surface interactions (Plasma-Material Interactions, PMIs) in the fusion reactor. During an in-vessel loss of coolant accident (LOCA) of the helium-cooled blanket, the high-pressure helium gas ingress into the vacuum vessel (VV) can cause resuspension and migration of dust deposited in VV. The DustSAFER facility was set up to perform experiments related to dust resuspension under in-vessel LOCA of the fusion reactor. The present work deals with the effect of dust size, break location, break size and initial pressure, and other scenarios on dust resuspension in VV. In addition, a three-dimensional (3D) modelling of DustSAFER was made with the CFD commercial code ANSYS FLUENT, in order to get an overview of the fluid dynamics behavior during an in-vessel LOCA event. The results showed that the resuspension amount of tungsten dust of 3.511 μm was greater than that of 1.66 μm, and the dust with larger break size had a higher resuspension rate. Moreover, the pressure of VV could rapidly reach a steady state in tens of seconds for all break sizes. Besides, the CFD found that the flow field changed dramatically near the break and captured the Mach disk. This work provides references for further research on the resuspension and migration of dust in the fusion reactor.

Keywords

In-vessel LOCA; dust resuspension; DustSAFER; CFD.

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PS3-101 ABSTRACT-af83
G. Safety Issues and Waste Management

Solid waste products of EU DEMO – focus on tungsten dust reprocessing

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Operation of EU DEMO or a fusion power plant will produce various kind of waste and result in the creation of radioactive material, through neutron activation and/or contamination with tritium.

For the future EU DEMO nuclear fusion reactor, the preliminary construction designs of the individual main sections are known. These designs and the operation of test facilities testing sub-operations leading to nuclear fusion are the basis of considerations which materials are likely become waste. Continuously generated waste materials are solid dusts. Their origin is mostly escape from the reactor. It will also appear during internal periodic maintenance of a decommissioned fusion device, most often from the armour surfaces. Another group of similar wastes is generated during the treatment of replaced parts.

The source of waste is the dust abrading after the overhaul of the equipment and other small residues. Tungsten dust, which contains tritium, is produced in the largest quantity.

Revision and overview of expected solid waste masses by distinguishing what could be coming from operational process and what could come from decommissioning according to their activation classification levels and with regards to the storage potential capacities. Proposals for processing methods (in DEMO site or outside DEMO site) allowing to reduce and minimize the solid waste shall be proposed. The lessons learnt from ITER are to be considered as well.

Main methods of reprocessing waste dust using high-temperature technologies, such as induction heating or MSO (Molten Salt Oxidation) technology, are proposed.

The main goal of this article is to update the related studies developed in previous years within WPSAE in EUROfusion project and potential updates in the design that could lead to types of waste not yet considered.

Keywords

Solid waste, tungsten, dust particle, MSO, Induction heating.

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PS3-103 ABSTRACT-cb76
G. Safety Issues and Waste Management

Neutron Irradiation-Induced Shut-Down Dose Rate Estimation In EU DEMO Divertor

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The EU DEMO (EUropean DEMOnstration power plant) fusion power reactor is being developed in the framework of the EUROfusion Power Plant and Technology (PPPT) programme and, in future, will become ITER's successor. With DEMO construction, fusion will go from laboratory and science-driven concepts to the industry and technology-driven programs. Even though EU DEMO should become the first fusion power plant with its fundamental goal to produce more energy than consumes.

The divertor is the critical element for power exhaustion, ash and helium removal in DEMO. In 2014 the first divertor conceptual design was introduced. Although the ITERs divertor design was successfully developed, it needs some improvement for EU DEMO; neutron irradiation dose is expected to be ten times higher than ITER. The main divertor components are Plasma Facing Components (PFC), shielding liner and reflector plates, divertor cassette, and locking system on a vacuum vessel. Estimating shutdown dose rates is important for the safety of reactor operators and the public and for protecting other critical reactor systems.

The paper presents shutdown dose rate analysis using MCNP coupled with R2Smesh by employing a semi-heterogeneous and up-to-date model of EU DEMO divertor, which appeared to be a challenging problem in the field of neutron transport calculations.

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Keywords

EU DEMO, fusion, R2Smesh, MCNP, neutronics.

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PS3-104 ABSTRACT-399b
H. Models and Experiments for FNT

Computational scheme for fast production of parametric stellarator models suitable for neutronic analysis

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The stellarator is an alternative to the tokamak for achieving fusion as an energy source. Considering the success of the Wendelstein 7-X, this option has gained renewed attention. In particular, the pre-conceptual development of HELIAS has started in the framework of the Prospective Research & Development EUROfusion work package.

During this phase, the feasibility of several design options affecting the general geometric shape of HELIAS, such as the number of coils, and the blanket concepts, will be studied, including neutronic aspects. The workflow of the neutronic analysis required in this phase consists in: i. actualizing the CAD model, ii. simplifying and converting to the MCNP neutronic model, and iii. simulating and analysing the results. The whole process is iteratively repeated for optimizing the HELIAS design. The CAD models, representing HELIAS homogeneous models, are easily produced from the last close flux surface calculated in plasma analysis. However, these CAD only contain sp-lines surfaces, which are not directly suitable for MCNP calculations. Furthermore, when they are simplified and converted to MCNP format, the input typically contains several small geometric errors due to the precision during the conversion, which produces lost particles during transport. The CAD model simplification and the solution of the associated geometrical errors require months of human resources delaying the iterative Helias design.

In this work, we presented a computational scheme for producing parametric MCNP input models of stellarators to allow fast changes in the Helias design, with minimal geometrical errors. This scheme, whose application requires around a day, is based on three steps: i. The CAD model suitable for MCNP simulation is produced by the *ad-hoc* tool Helias-Geom, ii. This CAD model is traduced by GEOUNED, and iii. The small geometrical overlaps of the cells are solved by an iterative but automatic process based on D1SUNED new features.

Keywords

MCNP, Geometry, Helias, Stellarator.

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PS3-105 ABSTRACT-4c72
H. Models and Experiments for FNT

Neutron Yield Semi-Monte Carlo Simulating Method for DT Fusion Neutron Source

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[5]FDS Consortium

The gaseous target fusion neutron source is an important supplement and extension of the large fission/spallation neutron facilities in wide applications in fusion neutronics, material irradiation test, isotopes production and so on. The exact calculation of neutron yield and spatial distribution with a large size geometry of gaseous target neutron source is very important for design and subsequent application studies. It is a more efficient and reliable approach to calculate the parameters coupled analytical method with Monte-Carlo method than using them alone.

The fusion neutron yield and spatial distributions examining relates to the spatial characteristics (number density, energy) of deuterons and the fusion cross section. Firstly, the consideration on deuteron transport is Monte-Carlo method with CSDA and the multiple scattering respectively for the interactions with the extranuclear electrons and the scattering with the nuclear of the target atoms. The analytical energy losing of deuteron and neutron yield are calculated using the datasheets of stopping power and fusion cross section adapted from the result of SRIM-2015 and DROSG2000. The calculating process is rearranged with finite element technology, and designed incorporating CFD for the subsequent extensive study on the effect of target gas circulation and beam heating.

The method has been verified using a DT fusion neutron source model with windowless gas target and compared with experiment results and the GEANT4 simulations. Compared with the experiment results, the simulated neutron yield error is 9.79% and the relative error of total neutron yield is better than -3.34%. The spatial neutron yield distributions are in good agreement with the results of GEANT4. In particular, the calculation time is reduced from ~200 core-hours with GEANT4 to 50 core-hours with the developed semi-Monte Carlo method, greatly improving the calculation efficiency.

Keywords

Neutron Yield, Fusion Neutron Source, Gaseous Target, Semi-Monte Coral Method.

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PS3-106 ABSTRACT-5345
H. Models and Experiments for FNT

Design of an Energy Transfer Storage System Based on EAST

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EAST, fusion facility institute of plasma physics, which is a full superconducting tokamak with shaped poloidal cross-section. When the Tokamak device is discharged, a phenomenon of plasma rupture often occurs. The occurrence of plasma rupture has brought great harm to the Tokamak device. Although the current method of protection against rupture has achieved a relatively good degree, It relieves the cracking effect, However, it can not reduce the total amount of electromagnetic energy dissipated inside the vacuum chamber.Based on this, the energy transfer storage system about EAST is designed in this paper. The energy transfer storage system includes an overvoltage protection unit and an energy storage unit.Through circuit analysis and modeling, an over voltage protection unit is designed, and the over voltage generated by plasma rupture is suppressed below 1200 V. Through the simulation of PSIM, the pulse capacitance value and the energy transfer coil inductance value are adjusted, and the corresponding transfer electromagnetic energy and capacitance storage energy are obtained. The data processing and analysis are carried out. Finally, the appropriate energy transfer coil inductance value and pulse capacitance value are selected,and the corresponding energy storage unit was designed.

Keywords

Plasma rupture, Overvoltage, Energy transfer storage.

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PS3-108 ABSTRACT-d28e
H. Models and Experiments for FNT

Calculation of Neutralization Efficiency and Beam Transport for the Two Region Arc Plasma Negative Ion Source

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EAST, fusion facility institute of plasma physics, which is a full superconducting tokamak with shaped poloidal cross-section. When the Tokamak device is discharged, a phenomenon of plasma rupture often occurs. The occurrence of plasma rupture has brought great harm to the Tokamak device. Although the current method of protection against rupture has achieved a relatively good degree, It relieves the cracking effect, However, it can not reduce the total amount of electromagnetic energy dissipated inside the vacuum chamber.Based on this, the energy transfer storage system about EAST is designed in this paper. The energy transfer storage system includes an overvoltage protection unit and an energy storage unit.Through circuit analysis and modeling, an over voltage protection unit is designed, and the over voltage generated by plasma rupture is suppressed below 1200 V. Through the simulation of PSIM, the pulse capacitance value and the energy transfer coil inductance value are adjusted, and the corresponding transfer electromagnetic energy and capacitance storage energy are obtained. The data processing and analysis are carried out. Finally, the appropriate energy transfer coil inductance value and pulse capacitance value are selected,and the corresponding energy storage unit was designed.

Keywords

Negative ion, ion source, Neutral beam, NBI, Arc plasma, Neutralization Efficiency, Beam transport.

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PS3-109 ABSTRACT-c63f
H. Models and Experiments for FNT

HCPB TBS Tritium transport model at system level with EcosimPro®

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Tritium transport modelling plays an important role in the design activities of the European TBMs (Test Blanket Module). Considering the difficulty in handling tritium due its radioactivity and high permeability, one of the main goals of this simulation model is to determine the amount of tritium per unit time that can be collected, accounted for and routed out of the TBS (Test Blanket System) auxiliary systems to the tritium systems of the ITER Tritium Plant.

Different approaches can be followed to develop a tritium transport modelling tool in the TBS area depending on the problem dimensionality (0D/1D/3D) the time evolution strategy (steady state or transient) or the permeation regime (diffusion or surface limited) This paper presents the simulation work using a 1-D dynamic multi-isotopic transport simulation tool, called TRITIUM, developed within the EcosimPro® platform as result of a collaborative effort between *Empresarios Agrupados Internacional* (EAI)/CIEMAT

TRITIUM comprises a set of libraries containing first-principles models of the various terms involved in tritium transport such as: generation, advection, chemical reactions, permeation, diffusion, trapping, recombination and dissociation. TRITIUM has been developed for tritium transfer modelling and has robustly shown its strong performances for this purpose.

This work describes the HCPB (Helium Cooled Pebble Bed) TBS simulation model at system level. Together with the TBM concept, the most relevant ancillary systems are considered. The possibility to simulate NOS (Normal Operation State) and STM (Short Term Maintenance) conditions are included in the subsystem formulation.

Some parametric studies are run, and the simulation results are used to predict the tritium concentration, permeation to coolant and possible leaks in the different TBS sub-systems and components as a function of time, in its different chemical forms, with enough compromise between model accuracy/complexity/reliability.

The outcomes of this work provides valuable information for the design of the HCPB TBS.

Keywords

HCPB, TBM, transport model, EcosimPro, tritium accountacy, permeation.

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PS3-110 ABSTRACT-6bff
H. Models and Experiments for FNT

Design of modular Hydrogen trap for the lithium purification system of IFMIF-DONES

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This paper presents the design of a Hydrogen Trap (H-TRAP) developed for the lithium circuit LITEC, which is representative of the lithium purification system of IFMIF-DONES. The H-TRAP is a collaborative effort between IDOM and CIEMAT under the EUROfusion framework project. The purpose of this research is to investigate the influence of the main working parameters of the H-TRAP: working temperature, amount of getter (Yttrium), residence time, and aspect ratio of the tank. The efficiency of yttrium as a hydrogen getter will be studied.

The LITEC system aims to purify lithium for use in the IFMIF-DONES fusion facility, and the H-TRAP plays a crucial role in this process by capturing hydrogen. To fulfill the research objectives of this prototype and be compatible with LITEC interfaces, the H-TRAP has been designed with a modular structure that allows for easy replacement and handling. This design takes into account the different working modes and addresses the potential issue of lithium stagnation by incorporating features that facilitate drainage and disassembly.

The development of the H-TRAP focused on performing thermomechanical numerical analysis and hydrodynamic calculations to ensure that the H-TRAP can accommodate the experiments mentioned. The main results of the research show that the H-TRAP meets the design requirements and allows for experimentation with the desired range of functional parameters. The structural requirements were successfully checked against RCC-MRx(2012).

This study provides a detailed understanding of the H-TRAP design and its potential for advancing research in the context of lithium purification for fusion facilities such as IFMIF-DONES.

Keywords

Hydrogen-TRAP, lithium loop, ifmif-dones.

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PS3-111 ABSTRACT-75d4
H. Models and Experiments for FNT

Sensitivity and uncertainty analyses of SAD and SED in the FNG HCPB Tritium Breeder Module benchmark

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The Secondary Energy Distribution (SED) and Secondary Angular Distribution (SAD) play a crucial role in determining the behavior of fusion products in the shielding blankets of fusion reactors. This paper proposes a precise method to analyze the sensitivity and uncertainty of SAD and SED. Sensitivity analysis adopts the generalized first-order perturbation theory, which consider not only the direct effect caused by cross-sections changed, but also the indirect effect caused by flux spectrum changed due to cross-sections changed. Uncertainty analysis is based on sensitivity analysis, combined with covariance matrix to quantify uncertainty. Multigroup cross-sections and related covariance matrix are obtained from Fusion Evaluated Nuclear Data Library (FENDL-3.2) using NJOY2016. The method was used to calculate sensitivity and uncertainty of SAD and SED towards tritium production ratio in FNG HCPB Tritium Breeder Module benchmark. The results show that SAD and SED can cause large uncertainties in fusion shielding analysis.

Keywords

Sensitivity, Uncertainty, SAD, SED, Tritium Breeder Module.

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PS3-112 ABSTRACT-7af8
H. Models and Experiments for FNT

The Water cooled lithium lead thermal-HYDRAulic experimental platform

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The Water cooled lithium lead thermal-HYDRAulic (**W-HYDRA**) experimental platform is constituted of three different experimental loops, aimed at supporting relevant aspects connected with the design, the technology and the safety analysis of the Water Cooled Lithium Lead Breeding Blanket and Test Blanket Module. These three loops are: Water Loop, STEAM and LIFUS5/Mod4. The Water Loop is a facility that can be operated at high pressure and temperature (design pressure and temperature are 18.5 MPa and 350°C, respectively) with sufficient mass flow rate and power (i.e. up to 5kg/s and 1 MW) for testing full-scale TBM and scaled down BB mock-ups, under relevant loading conditions. The Water Loop is integrated with a vacuum chamber where water cooled mock-ups can be tested under relevant heat flux conditions, reproduced by an Electron Beam gun device. The LIFUS5/Mod4 loop is an integral test facility representative of the PbLi loop of ITER. When, connected with the Water Loop, it is possible to study the thermal-hydraulic performances of test sections (e.g. featuring portions of the WCLL breeding blanket) and the response at system level of multifluid multiphase trasients, such as the postulated accident "in-box Loss Of Coolant Accident" at full TBM scale. The STEAM loop is designed to address the steam generator operation challenges connected with the pulsed operation of DEMO. It is featured by an high pressure water/steam secondary side loop, with the capability to face the fast load changes occurring in a nuclear fusion tokamak. Its primary system is the "high power" branch of the Water Loop characterized by a larger heating power (up to 3.1 MW) and mass flow rate (up to 20kg/s). The paper describes the design features and capabilities of the W-HYDRA experimental infrastructure, providing details about the status and the planning of the construction, and the already scheduled experimental programs.

Keywords

DEMO, Water Loop, STEAM, LIFUS5/Mod4, experimental facility.

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PS3-113 ABSTRACT-7dda
H. Models and Experiments for FNT

'Effects of multi-toroidal neon injections on the asymmetry of heat flux distribution with EMC3-EIRENE modelling

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The control of heat flux deposition is one of the major concerns for the long-pulse high-performance operation of fusion facilities [1-2]. Extrinsic impurity seeding is considered as an effective solution for scrape-off layer (SOL) power dissipation to alleviate damage to divertor by spreading excessive heat loads over a large wall area [1]. However, the toroidal asymmetric heat load distributions have been captured with spatial-localized impurity injection on tokamak and stellarator, such as Alcator C-Mod and LHD. The previous studies on EAST show that neon impurity injections near the in- and out-board strike points lead to the toroidally asymmetric distributions of heat load on the in- and out-board divertor targets, respectively [3-5].

In order to achieve the relatively uniform mitigation of heat loads along the whole torus, investigations on heat flux distribution under multi-toroidal locations of neon impurity seeding on EAST have been conducted by the three-dimensional (3D) edge transport code EMC3-EIRENE. The non-uniformity of heat flux distribution has been analysed by means of the heat load fall-off width and the peak heat load on divertor targets. The impacts of neon injection flux on the heat load distributions are studied, which indicates that high neon injection flux for a single toroidal neon injection cannot lead to the relatively uniform mitigation of heat loads along the whole torus. On this basis, multi-toroidal neon injections near the in- and out-board strike points have been investigated in order to attain relatively uniform heat load distributions in the toroidal direction.

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Keywords

Neon injection, heat flux, EMC3-EIRENE, EAST.

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PS3-114 ABSTRACT-858e
H. Models and Experiments for FNT

Numerical and analytic solutions for liquid metal flows under non-uniform magnetic field gradients in a circular pipe

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Magnetohydrodynamic (MHD) effects are of great importance to liquid metal flows in the cooling and breeding systems of fusion reactors. MHD has the potential to significantly affect the flow of the liquid metal, impacting on the pressure drop across the pipe network, as well as the cooling and breeding properties of the liquid metal. Accounting for these effects within the design process is essential to better estimate pumping power requirements and to optimise the layout of the pipe network.

The MHD flow of an electrically conducting fluid changes in response to a varying magnetic field. The exact nature of the flow will depend on various factors such as the geometry of the ducts, the fluid and wall material properties, the magnetic field distribution, and the balance of hydrodynamic and MHD forces. The distribution of the non-uniform magnetic field, as present in fusion reactors, has the capacity to lead to an azimuthal variation in velocity and pressure, as well as a modification of the flow regime (e.g., laminar to turbulent) owing to the axial gradients of electric potential resulting in 3D current loops.

In extension to existing knowledge and mathematical models, as well as numerical investigations, the MHD flow in electrically conducting and insulating straight circular pipes is investigated, under an externally applied non-uniform magnetic field. A parametric study of the effects of the magnetic field gradient is performed comparing the results of numeric simulations and asymptotic analysis for high magnetic fields.

Keywords

Magnetohydrodynamics, MHD, non-uniform magnetic field.

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PS3-115 ABSTRACT-88d1
H. Models and Experiments for FNT

Visible cameras as a tool to study electron-beam shape

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The characterization of the spatial distribution of particle trajectories, or the respective current distribution, in powerful electron beams, is an important scientific and practical task necessary to improve the quality of electron beam technologies. For various applications, a small divergence is required, to transport the electron beam over long distances and to focus it onto a small spot. To characterize an electron-beam, the simultaneous knowledge of accelerating voltage, beam current, focus coil current and vacuum level provides little insight into the properties of the beam itself. Visible cameras are used to study the shape of the electron beam when these parameters are varied. A series of pictures are collected at different orientations of the camera around the beam. The light emission comes from the Ar background gas excited by the energetic electrons composing the beam. The information is used to reconstruct the 2-dimensional shape of the beam by means of tomographic inversion.

Keywords

Beam diagnostics, tomography.

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PS3-116 ABSTRACT-1ddd
H. Models and Experiments for FNT

Preparing for CHIMERA commissioning by virtual testing via systems simulations

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CHIMERA (Combined Heating and Magnetic Research Apparatus) is a unique, multi-physics fusion component loading facility under construction at the UKAEA Yorkshire site. Once complete, CHIMERA will be used to test large in-vessel component modules under reactor-like conditions. Digital capability is being developed in parallel to the facility construction and will be utilised to influence and plan the facility commissioning and first test campaigns, the developed simulations will also be validated by the experiments.

A systems simulation of the commissioning sample under test (SUT) and relevant sections of the CHIMERA rig has been developed. The systems simulation is a series of linked fast running surrogate models covering hydraulic, thermal, electromagnetic, and mechanical domains. The systems simulation has recently been upgraded to include transient thermal and electromagnetic loading, now covering all commissioning load cases. This involves scaling load vectors for a component mode synthesis mechanical reduction, using a novel modal coupling method.

The test sensors, such as thermocouples and strain gauges, have been integrated into the systems simulation. Rigorous sensitivity analysis is applied to understand the system input-response sensitivities and to optimise the placement of such sensors. A fatigue assessment has also been integrated, allowing tracking of progressive damage of SUTs during an experiment. Using these capabilities, a suggested commissioning test plan will be presented, which aims to bring the facility online with mitigated risk of facility or SUT damage and to maximise the use of the data to efficiently validate the simulation.

Keywords

Digital Twin, Thermal, Mechanical, Electromagnetism, FEA.

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PS3-117 ABSTRACT-8b32
H. Models and Experiments for FNT

Verification and validation procedure for the tritium processing modelling libraries of EcosimPro

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The first version of the tritium transport modeling libraries for EcosimPro was co-developed by CIEMAT and EAI in 2010. Initially, the code was focused on addressing safety and fuel self-sufficiency issues for both breeding blanket systems and the fuel cycle of a nuclear fusion reactor. EcosimPro libraries have become a reference in European programs for evaluating tritium inventories and losses due to permeation to the environment in Test Blanket Module and Breeding Blanket systems. During these years, verification, validation, and benchmarking processes were carried out in parallel with these works. Likewise, the inherited error in the results of the code due to the uncertainty in the measurement of the properties of fusion functional materials used as input was evaluated.

The compilation of this process is presented in this work to establish a degree of confidence in the future models results.

Keywords

Tritium, Modelling, Breeding Blanket, Fuel Cycle.

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PS3-120 ABSTRACT-4982

I. Repair and Maintenance

A Conceptual Design of Automatic In-vessel Inspection Systems

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In-vessel components of a fusion device can be damaged by various plasma faults event during the operation. The damaged components and fragments of PFC can affect the stability of the device, leading to accidents. To prevent accidents, it requires to check the conditions of the components regularly. However, human workers cannot inspect the components directly because the in-vessel environment is high radiation. This paper presents a conceptual design of an automatic in-vessel inspection system (AIIS), including system configuration, hardware design, and structural analysis as part of a stage-wise approach to developing the AIIS. The AIIS comprises a robot arm, a gripper, visual inspection sensors, operating systems, data processing systems, control systems, digital twins, and user interfaces. In AIIS, the gripper and the sensors are attached to the robot arm and operate in the in-vessel environment. The operating system establishes an automatic inspection plan and manages the inspection according to the inspection plan based on the control system and the digital twin. During the operation, the data processing system identifies damaged components based on sensing data. The inspection results can be used in three parts 1) Generating a minimum repair time plan, 2) Raw data of the component life prediction system, and 3) The basis for an emergency stop decision.

We designed the robot arm and gripper with access to a small port (~250mm). The designed robot arm is around a 9m articulated robot arm that can operate under high vacuum (about 10-5Pa) and high temperature (about 300°C radiant heat). The reachability of the robotic arm will be tested to verify that all in-vessel components can be inspectable. The results can utilize as the basis of the path planning for in-vessel component inspection.

Keywords: AIIS, Remote Control, Inspection, Repair, Maintenance, Automation

Keywords

AIIS, Remote Control, Inspection, Repair, Maintenance, Automation.

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PS3-121 ABSTRACT-4b00
I. Repair and Maintenance

Assessment of different lifting devices for the shipping bays for DONES main building

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During the installation and maintenance phase of IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO Oriented NEutron Source), several components need to be transported vertically between different floors inside the DONES main building. There are totally four shipping bays inside the main building, two of which are mainly responsible for components transportation. The lifting devices in the shipping bays should be carefully selected based on the requirements of components, including their sizes, weights, shapes, activation status, shielding requirements, human escorts and transportation routes. Considering these requirements and the building information, five possible lifting devices are proposed, namely parallel ropes lift, heavy load storage and retrieval machine, rack and pinion lift, upside-down scissor lift as well as rigid chain lift. The optioneering method is applied to assess the different lifting devices with the help of weighted criteria. As a result, parallel ropes lift, heavy load storage and retrieval machine and rigid chain lift are assessed as most appropriate lifting devices in the shipping bays for DONES main building. A market survey is carried out on these three kinds of lifting devices. The 3D model of commercial products will be integrated in the main building including the transport platform.

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Keywords

DONES, maintenance, lift devices, optioneering.

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PS3-122 ABSTRACT-4dce
I. Repair and Maintenance

Manufacturing and Repair of ITER Blanket Panels: Exploring Joining Techniques for Beryllium Tiles and CuCrZr/Cu Joints

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The ITER Blanket, a layer of 440 panels, will form the first wall to protect the chamber housing from the fusion reaction. Each panel measures 1 meter in height and 1.5 meters in length and weighs up to 1.5 tons. They will be joined onto a stainless steel-copper alloy bi-metallic structure to cover an area of 600 m^2. Essentially, the panels are thick metallic blocks made up of beryllium tiles, with stainless steel shield blocks installed at the back of each panel to complete the ITER Blanket

Beryllium tiles can be de-bonded or damaged during manufacturing or handling. A viable manufacturing process requires a method for repairing beryllium tiles. The repair involves removing the damaged block and attaching a new block to the structure. To facilitate the attachment process, a thin layer of copper is initially applied onto the beryllium material, thereby allowing to perform the joining with the copper.

High-temperature brazing is a viable method for creating strong joints between Cu and CuCrZr. However, the main challenge lies in the temperature limitations of the process. The properties of CuCrZr tend to deteriorate as the temperature and exposure time increases.

Solid state sintering of silver particles has been identified as an alternative because of its excellent thermal and electrical properties. This low temperature joining technique is based on the principle of diffusion welding, where a material is subjected to a temperature below its melting point and assisted by an external pressure to consolidate it into a solid.

This work presents the results of an experimental development in which CuCrZr/Cu joints were achieved using nano-Ag sheets. The maximum processing temperature at the joint surface was below 300°C.

Keywords

ITER, repair, beryllium tile, joining, nano-Ag.

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PS3-123 ABSTRACT-8d81
I. Repair and Maintenance

Design and development of a flexible manipulator as test rig platform for robot flexible behaviors in remote handling operations

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Long-reach flexible manipulators used for inspection and maintenance in fusion power plants exhibit elastic link behaviors that need to be predicted in detail to plan and perform safe operations. One of the great challenges in planning safe remote maintenance operations is taking into account the real flexible behaviors into the control systems of such robots, aspect which is currently almost neglected. Testing of such control algorithms can be done in simulations, even if the final validation should be done ideally on real systems; however only few hardware facilities exist all over the word for this goal.

In this work we present the development of a scaled robotic test rig that can be used to validate modeling and control strategies for flexible manipulators used for Remote Handling tasks in nuclear fusion plants. The test rig is a 5-DoF (degrees-of-freedom) flexible manipulator, composed of 4 revolute joints and 1 prismatic joint. The kinematic structure and the typology of joints has been selected in order to be representative for longer flexible manipulators used for nuclear fusion research as the TARM (Telescopic Articulated Remote Manipulator) of UKAEA. The test rig is equipped with onboard strain sensors placed along the last three links and one IMU (inertial measurement unit) sensor placed at the tip of the robot. In particular, here we report its mechanical and electric design as well as its software architecture. Furthermore, we report the usage of the test rig in a control strategy to suppress undesired vibrations.

Acknowledgement

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Keywords

Flexible manipulators, remote maintenance.

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PS3-126 ABSTRACT-8017
I. Repair and Maintenance

Assessment of architectural changes that benefit remote maintenance structural feasibility for Large Port-Based Tokamaks focusing on the study of the aspect ratio and the inboard blanket segmentation

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Large port-based tokamaks will have Tritium breeding blankets inside their vacuum vessel. Inboard blanket handling has been identified as the most significant issue with remote handling of the blankets for maintaining large port-based tokamaks. This is due to the offset loads creating large moments and complex kinematic movements required to extract them from the vessel.

This study investigates architectural changes to a large port-based tokamak that aim to improve the remote handling of the breeder blankets. Two approaches have been undertaken. The first focuses on improving the inboard blanket handling by revising the blanket segmentation and developing a novel overlapping interface design. The second approach increasing the upper port size and the useable radial size of the tokamak by altering the major radius of the Tokamak – maintaining other key parameters.

Both approaches have been assessed by establishing the key remote handling loads and moments and comparing to a baseline model. Additional FEA analysis has been undertaken on the overlapping interface design to indicate its structural feasibility.

Both have demonstrated benefits by making the remote maintenance procedure more feasible. The asymmetrical split of the inboard breeding blankets reduces the radial moment loads transferred to the blanket transporter. The overlapping interface greatly benefits remote maintenance, lowering the bending moment loads of the inboard blankets by up to 46%.

In addition, the extra space generated in the upper port by increasing the Tokamak major radius compensates for the additional blanket mass generated, resulting in lower overall remote maintenance handling moments.

Keywords

RM, DEMO, Blanket handling.

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PS3-127 ABSTRACT-86ce
I. Repair and Maintenance

RAMI analysis of ITER diagnostic Radial Neutron Camera

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RAMI (Reliability, Availability, Maintainability and Inspectability) assessments are mandatory part of the design process for all ITER systems to anticipate possible risks in terms of reliability and availability and support reliability growth program. A RAMI assessment performed on the ITER Radial Neutron Camera (RNC) diagnostic system is presented. The assessment is aimed at evaluating the RNC design capability to provide the neutron emissivity radial profile measurement with required reliability and availability. The RNC is composed by two collimating structures equipped with neutron flux detectors, the In-port RNC sub-system and the Ex-port RNC sub-system respectively. Such systems radially view different plasma locations thus enabling a tomographic approach in the emissivity profile reconstruction. Both in-vessel and ex-vessel detection systems (sensors, collimators, shielding) and full acquisition system chain (front-end and back-end electronics) are considered in the analysis. The RAMI performance was assessed by means of reliability block diagrams (RBDs) to assess system capability of achieving required mean inherent availability fixed at 99.5 % and 88.3% for the ex-port and in-port system respectively for 2 years of operations. In particular, a set of failure events for each RNC component was initially defined by means of a failure mode and effect analysis. The resulting unavailability conditions of the systems were then identified. Hence identified groups of events were used to feed the RBDs model definition according to reliability-wise integration of the considered components. The integrated RAMI performance of RNC systems was finally estimated and considering the current level of design development, they appear able to meet stated requirement thanks to design redundancy.

Keywords

Reliability, Availability, Radial Neutron Camera, ITER, failure, maintenace.

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PS3-128 ABSTRACT-796a
J. Burning Plasma Control and Operation

Development of a real time spectroscopic neutron camera for feedback plasma control of a DT burning plasma

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In a fusion plasma, neutrons are generated in the reactions D + D → He3 + n (2.5MeV) and D + T → He4 + n (14.1 MeV). Neutron radiation is of great importance because it carries the energy produced by the fusion reactions, in addition to meaningful information on the ions fuel distribution functions. This makes Neutron Emission Spectroscopy (NES) diagnostics one of the best candidates for feedback plasma control of the fuel ions.The design of the proposed neutron camera diagnostic system for Deuterium - Tritium (DT) reactor is based on the Chemical Vapor Deposition (CVD) single crystal diamond technology. This type of detectors combine high resolution NES measurements (<1% at 14 MeV) with high counting rate operation (up to 2 MHz) which are the base for achieving real time (10-100 ms time resolution). The neutron camera consists of a vertical array of diamond detectors, thus allowing NES on each line of sight, to obtain radial profile information of the neutron emission and plasma parameters, in particular of the plasma ion temperature (Ti) which on a burning plasma dominated by thermal emission can be directly inferred from the measured neutron spectrum broadening. This work comprehends the first steps of the diagnostic development process. A real time algorithm, implemented on a Field Programmable Gate Array (FPGA) is selected, developed and tested to calculate the Ti from the diagnostic data of a CVD detector, where the major requirements considered are the algorithm stability, precision and time complexity. Subsequently a prototype diagnostic system based on a CVD diamond matrix is tested at the ISIS neutron facility on the 14 MeV DT NILE beam-line.

Keywords

Plasma control, Feedback control, Neutron DIagnostic, Real Time feedback, Neutron Spectroscopy.

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PS3-129 ABSTRACT-79ee

J. Burning Plasma Control and Operation

Visible imaging system with changeable field of view on the HL-2A tokamak

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A new visible imaging system characterizing a flexible optical design and delivering high resolution photos is established on the HL-2A tokamak. It features a modular configuration, consisting of a front-end imaging lens, a set of bilateral telecentric relay lenses and a camera. To avoid the effects of plasma radiation (X- and gamma-rays) and magnetic field variation on the camera, it should be away from the tokamak machine. Therefore, the length of relay lenses determines the total size of the imaging system. The merit of this imaging system is to realize the variation of field of view (FOV) by interchanging the front-end prime lenses, or by using a zoom lens directly, rather than designing the optical system afresh, which lowers the cost drastically. The primary purpose of varying FOV is to enrich the versatility of this system, i.e. focusing on a narrow FOV such as gas puff imaging (GPI) or a wide FOV such as the plasma cross sections. This visible imaging system is applied during the HL-2A experiments, providing the high quality pictures about the plasma-wall interaction (PWI), divertor detachment, and pellet injections, etc. That a strong radiation close to the X point is correlated to the completely detached inner target is confirmed by the photos. This X-point radiation is correlated to the maximum pressure loss along the separatrix from upstream to the target.

Keywords

Visible imaging diagnostic, divertor detachment.

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PS3-130 ABSTRACT-8508
J. Burning Plasma Control and Operation

Integrated Physics Analysis of the 100 MWe-class Compact Helical Fusion Reactor

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The LHD-type helical reactor has an advantage on its intrinsic steady-state operation capability and its high design certainty based on the construction and operation of the LHD. To realize a compact fusion power plant, simultaneous achievement of the reduction of the reactor size and the increase in the fusion power, is required. However, the reduction of the reactor size is limited due to the lack of the space for the neutron shielding. The increase in the fusion power is also limited by the restriction of the plasma performance due to the trade-off between the MHD stability and the energy confinement. Recently, it was found that a slight change in the shape of the helical coils enables the improvement of the plasma performance as well as the enlargement of the space for the neutron shielding by the coil-shaping-based stellarator optimization code, OPTHECS [1]. The improvement of the efficiency of the various plant facilities is also expected by the recent progress in the design studies and engineering R&Ds. In response to these progresses, a new design of helical fusion reactor which enables 100 MWe net electric power with twice the reactor size of the LHD, has been proposed by Helical Fusion Co., Ltd. To confirm the design feasibility of this new reactor, integrated physics analysis including MHD stability, neoclassical transport and alpha particle behavior is being conducted while taking into account engineering considerations such as the structural analysis of helical coils. In the presentation, the results of this integrated physics analysis and the operation scenario will be discussed.

[1] H. Yamaguchi, 28th International Toki Conference, O1-4, Toki, Gifu, Japan, Nov. 5-8, 2019, https://www.nifs.ac.jp/itc/itc28/

Keywords

Helical Reactor, Heliotron, Operation Scenario, Integrated Physics Analysis.

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PS3-132 ABSTRACT-c651
J. Burning Plasma Control and Operation

Special Plasma Control technologies with novel Results on EXL-50

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EXL-50[1], a solenoid-free spherical torus [2-3] has been built in less than one year with a simple PF system. Three magnetrons and five gyrotrons have been put into operation step by step during two years for driving and heating the plasma. In this solenoid-free machine, several special control technologies, quite different to the conventional ones have been suggested and put into operation, novel experimental results have been obtained. In this paper, the detailed information of plasma control on EXL-50 has been introduced and the novel and exotic results have been described. With these control technologies, 1) the TF current is operated in two flattops with a short transition period in order to be compatible with different ECRH frequencies or different ECRH harmonic waves; 2) The current in outer PF coils close to equatorial plane often change its direction for pulling plasma a little outer for VDE stabilization, or quickly control plasma radial position when plasma current suffers a uncontrollably rise due to the ECRH power jump; 3）The plasma current which is mainly carrying by energetic electrons[4] can change its direction only by changing the total vertical field. Some novel phenomenona have also been introduced. We observed, 1) Two plasma rings can exist together; 2) Apparent plasma current spike appears due to TF current ramping down. These technologies may apply to other low aspect ratio torus machines, and these phenomenona may appear in other torus machines in the future.

Keywords

Solenoid-free, spherical torus, plasma control, energetic electron.

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PS3-133 ABSTRACT-7b03
K. Inertial Confinement Fusion Studies and Technologies

Phase imaging of irradiated foils at the OMEGA EP facility using phase-stepping X-ray Talbot-Lau deflectometry

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Diagnosing the evolution of laser-generated High Energy Density (HED) systems is fundamental to develop a correct understanding of the behavior of matter under extreme conditions. To this end, Talbot-Lau interferometry constitutes a promising tool, since it permits simultaneous single-shot X-ray radiography and phase-contrast imaging of dense plasmas. In this talk, we present the results of an experiment at OMEGA EP that aims to probe the ablation front of a laser-irradiated foil using a newly established diagnostic platform, the Talbot-Lau X-ray Interferometer. A CH foil was irradiated by a laser of 150 J, 1 ns and probed with 8 keV laser-produced backlighter radiation from Cu foils driven by a short-pulse laser (150 J, 10 ps). The ablation front interferograms were processed in combination with a set of reference images obtained ex-situ using the phase-stepping method. For the first time, we managed to obtain simultaneously attenuation and phase-shift images of a laser-irradiated foil for electron densities above 1e22cm-3. The results presented showcase the capabilities of Talbot-Lau x-ray diagnostic methods and their potential to diagnose the hydrodynamic evolution of HED laser-generated plasmas through high-resolution imaging

Keywords

Talbot-Lau, X-ray interferometry, OMEGA EP, phase contrast imaging, deflectometry.

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PS3-135 ABSTRACT-1e18
K. Inertial Confinement Fusion Studies and Technologies

Spectrally resolved emittance monitor for Laser-Plasma Ion Accelerators

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Since the demonstration of transport, focusing and energy selection of laser-driven multi-MeV ion beams [1,2], their potential applications, such as fast ignition approach to ICF [3], isochoric heating of dense plasmas [4] or ultrafast proton probing of plasmas [5], have attained plenty of attention. The extraordinary laminarity that the accelerated ions driven by laser-plasma interaction offer is the fundamental property for effective post-acceleration transport and focalization onto small focal volumes, required for most of the foreseen applied research.

We report on the evaluation of spectrally resolved beam laminarity and emittance values for multi-MeV ion beams [6], by means of a combined methodology between pepper-pot [7] and Thomson Parabola techniques. This approach shows operational improvements with respect to previous works [8,9], as well as substantially better spectral and spatial resolution but still agreeing with past measurements. The first statistics of energy-resolved emittance and trace-space of the ion beam driven by the Petawatt class laser VEGA 3 @ the CLPU facility in Spain are presented.

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Keywords

Laser-driven ion accelerator, plasma, emittance.

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PS3-136 ABSTRACT-9d47
K. Inertial Confinement Fusion Studies and Technologies

Commissioning experiment on proton acceleration and characterization with the petawatt-femtosecond class Laser VEGA-3 at CLPU

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Here we report on the first proton acceleration experiment with the PW-class Laser VEGA @ CLPU[a]. The experiment has been carried out by focusing VEGA-3 laser onto micrometric Aluminum solid foils with different thickness and scanning a wide range of laser pulse durations and peak intensities. Protons with a broadband energy spectrum up to a maximum energy value of 22 MeV have been accelerated via Target Normal Sheath Acceleration (TNSA[b,c]). Several high-repetition-rate (HRR) diagnostics have been used in parallel to measure and characterize the proton beams. The results are in consistence with the predictions reported by the theoretical models[b,d,e].

[a]https://www.clpu.es/en
[b]M. Passoni *et al.*, *New J. Phys.* **12** 045012 (2010).
[c]P. L. Pole *et al. New J. Phys.* **20** 013019 (2018).
[d]M. Salvadori *et al. Journal of Instrumentation* **17** (04), C04005 (2022).
[e]J. Schreiber *et al., Phys. Rev. Lett.* ***97*** *045005 (2006).*

Keywords

Láser-plasma interaction, Laser particle acceleration.

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PS3-139 ABSTRACT-e645
K. Inertial Confinement Fusion Studies and Technologies

Preliminary nuclear analysis of HYLIFE-3: a thick-liquid-wall chamber for inertial fusion energy.

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The HYLIFE-II fusion chamber design was produced in the 80's at Lawrence Livermore National Laboratory (LLNL), for an Inertial Fusion Energy (IFE) power plant based on a heavy ion driver. The HYLIFE-II used an array of FLiBe (mixture of lithium and beryllium fluorides) jets to surround the point of target ignition providing shielding against neutrons, x-rays, and target debris. This neutronically thick region (~50 cm effective thickness) also served as the breeding fluid to generate enough tritium for a self-sufficient operation. The design was extensively studied, and showed the feasibility of ICF for energy production, but its development was discontinued when work on heavy-ion-driven ICF was discontinued at LLNL.

Recent developments on laser amplification based on Raman and Brillouin scattering in gases aim to produce laser energy pulses well over 10 MJ at a fraction of the cost of previous laser technologies. Such high-energy laser enables the use of high-gain and more robust targets, which are ignited with laser illumination through only two ports. These developments have restored the interest on HYLIFE design, leading to a new HYLIFE-3 proposal.

This work contains a description and basic nuclear analysis of the new HYLIFE-3, using state-of-the-art computation technology. First, a simple analysis is shown with a simple spherical model, showing the required FLiBe flow thickness to efficiently protect the first wall and the vacuum chamber from neutron radiation damage, as well as producing an estimate of the ideal tritium breeding capability. The work is followed by an analysis on a more detailed chamber design, stressing the weak points of the system, where FLiBe flow does to provide full protection, calculating the expected durability of laser optics, as well as a more realistic estimate of the tritium breeding capability.

Keywords

Neutronics, Hylife-II.

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PS3-140 ABSTRACT-f734
K. Inertial Confinement Fusion Studies and Technologies

Characterization of hot spots generated in DT plasmas by fast quasi-monoenergetic ion beams: self-heating/igniton curves and burning gains

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The HYLIFE-II fusion chamber design was produced in the 80's at Lawrence Livermore National Laboratory (LLNL), for an Inertial Fusion Energy (IFE) power plant based on a heavy ion driver. The HYLIFE-II used an array of FLiBe (mixture of lithium and beryllium fluorides) jets to surround the point of target ignition providing shielding against neutrons, x-rays, and target debris. This neutronically thick region (~50 cm effective thickness) also served as the breeding fluid to generate enough tritium for a self-sufficient operation. The design was extensively studied, and showed the feasibility of ICF for energy production, but its development was discontinued when work on heavy-ion-driven ICF was discontinued at LLNL.

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This work contains a description and basic nuclear analysis of the new HYLIFE-3, using state-of-the-art computation technology. First, a simple analysis is shown with a simple spherical model, showing the required FLiBe flow thickness to efficiently protect the first wall and the vacuum chamber from neutron radiation damage, as well as producing an estimate of the ideal tritium breeding capability. The work is followed by an analysis on a more detailed chamber design, stressing the weak points of the system, where FLiBe flow does to provide full protection, calculating the expected durability of laser optics, as well as a more realistic estimate of the tritium breeding capability.

Keywords

Ion Fast ignition, quasi-monoenergetic gaussian ion beams, hot spot, burning gains.

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PS3-143 ABSTRACT-4f73

L. Fission-Fusion Synergy and Cross Cutting Technologies

TRISO fuel processing using a spherical tokamak based fusion neutron source

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Fuel availability and the generation of long-lived radioactive byproducts are often quoted as important sustainability barriers for energy generation via the nuclear fission process. Nuclear fusion, although being much cleaner, is still in a technology maturation stage, with the nuclear aspects still full of uncertainties. The design, construction and operation of volumetric, fusion-based fast neutron sources would accomplish two things at the same time: open the door to new fuel cycles based in untapped resources such as thorium and accelerate the development of fusion technology on the nuclear side, where the operational experience is scarce. We have explored the use of fusion neutron sources for enrichment of fresh fertile material and destruction of minor actinides in partially spent fuel under the current fission fuel paradigm, but feedback from industry suggests that current enrichment and reprocessing technologies will be difficult to compete with. Based on this information from industrial actors, the use of fast neutron irradiators built around a low aspect ratio tokamak for processing TRISO fuel is explored. Unlike current fuel, TRISO requires much higher levels of fissile material enrichment (up to 20%) and is notoriously difficult to reprocess using current technology. The OpenMC and ORIGEN codes are used to study fuel breeding and reprocessing of prismatic high temperature gas-cooled reactor (HTGR) fuel assemblies irradiation and assess the viability of a fusion-based fast neutron source for processing TRISO fuel around the figures of merit of tritium breeding ratio and fuel enrichment speed.

Keywords

TRISO, reprocessing, hybrid systems, spherical tokamaks.

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PS3-144 ABSTRACT-6121
L. Fission-Fusion Synergy and Cross Cutting Technologies

Iaea activities on synergies in technology development between nuclear fission and fusion for energy production

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IAEA

The IAEA is working to facilitate sharing of relevant experience in fission technology to benefit the fusion community in technology development for electricity production. Considering the worldwide acceleration towards the early deployment of nuclear fusion for energy production, the IAEA launched a new initiative aimed at addressing the great engineering challenge of fusion, by promoting transfer of technology and know-how from fission to fusion. The main objective is to identify and analyse all the possible synergies on technology development and deployment between nuclear fission and fusion.

Fusion based power plants for energy production will face many challenges already well-known and addressed for the deployment of nuclear fission power technology, from design to construction to decommissioning through operations, infrastructure needs and economic competitiveness. While the fusion technology is rapidly maturing through the realization of large experimental facilities like ITER and other innovative fusion machines, it is important to develop a thorough techno-economic understanding of the subject matter for an optimized and well-informed development path of fusion power plants.

The IAEA is undertaking the development of a framework on synergies between nuclear fission and fusion. The objective is to support the fusion community in its effort to accelerate the technology development and deployment of nuclear fusion systems for energy production, Work is progressing through a series of Technical and Consultancy Meetings[1], with participation from a number of the IAEA member states. Final output consists in the production of an IAEA Nuclear Energy Series publication on Synergies in Technology Development between Nuclear Fission and Fusion for Energy Production.

Keywords

Synergy, Synergies.

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PS3-145 ABSTRACT-cc65
L. Fission-Fusion Synergy and Cross Cutting Technologies

Fusion for high-value heat production

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Electrification of the energy system underpins most net zero strategies. As it stands, renewables could provide a low-cost way of decarbonising electricity demand. But how marginal generation costs of renewables are increased by storage, conversion and demand-side management requirements at high renewable fractions is unclear. This is especially pertinent as electrification increases grid electricity demand - particularly for energy-intensive industrial processes, which mostly require heat.

This paper analyses the proposal of supplying heat to a number of industrial processes using fusion energy - for example, the production of hydrogen and ammonia. By supplying medium-temperature heat as a grid-scale heat pump, the energy produced by fusion is of higher value compared with electricity. This enables fusion to better compete in the future, where electricity will form more of the primary energy supply - an environment more relevant for fusion than today's energy system.

Keywords

Heat, economics, value, energy transmisión.

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PS3-146 ABSTRACT-bddb
L. Fission-Fusion Synergy and Cross Cutting Technologies

Applicability of neutron measurement techniques in fission reactors to breeding blankets of DEMO

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Online neutron measurements are routinely performed in fission reactors. The functions are: improving the reactor dependability and safety, power monitoring, characterizing of irradiation conditions in material testing reactors, assessing the performances in demonstration reactors. The conception strategy of a measurement system involves: definition of the expected needs and performances, based on the physics of the device to be measured; evaluation of the operational constraints; identification of the best available sensor technology.

The breeding blankets are a key component of a fusion reactor such as DEMO. The needs their neutron instrumentation should fulfil are: tritium accountancy, through neutron flux measurements in the breeding zones; inventory of activated structures, beneficial for handling procedures and life extension of the blankets; plasma monitoring thanks to a space distribution of neutron sensors behind the front wall, in complement to other plasma diagnostic systems.

The expected performances and operational constraints are very demanding, nevertheless some of those, as the neutron flux range and high temperature, are close to what is encountered in sodium-cooled fast neutron reactors. Besides, adding sensors directly within the blankets is not possible because accommodating for detector size and cable management would result in new sources of failures. The diagnostic slim cassette, primarily designed for microwave diagnostics, is an attractive location, however the representability of the measurements has still to be investigated.

The strengths and drawbacks of neutron sensors are discussed with respect to the performances and constraints. Self-powered neutron detectors and fission chambers benefit from a vast return of experiment in fission reactors and received recent developments in the framework of studies for sodium-cooled fast neutron reactors. The innovative optical fission chamber is discussed for its electromagnetic immunity and high dynamics demonstrated up to seven decades. Dosimetry is also considered for its ability to provide an absolute calibration and spectral unfolding.

Keywords

Neutron sensors, fission reactors, breeding blankets.

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PS4-1 ABSTRACT-a573
A. Plasma-Facing High Heat Flux Components

Thermal and structural analysis of JT-60SA Actively Cooled Divertor target submitted to high heat flux

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JT-60SA is a joint international fusion experiment being built and operated in Japan under the framework of the Broader Approach Agreement and Japanese National Fusion Programme, aiming at an early realization of fusion energy by conducting supportive and complementary work for the ITER project, all towards supporting the basis for DEMO. In this context, a collaboration has been established via EUROfusion between Fusion for Energy and CEA to develop actively cooled targets of the JT-60SA divertor.

The first Actively Cooled Divertor (ACD) target, planned to be operated in 2029, has graphite as armour material and TZM (a Titanium-Zirconium-Molybdenum alloy) as heat sink material. To be DEMO relevant, ACD target with metallic armour material is planned to be operated in 2033. For this target, tungsten or tungsten alloys are considered as potential candidates as armour material, while CuCrZr and TZM are considered as options as heat sink materials.

This paper focusses on the choice of the design (material, dimensions...) of these targets. Results of thermal and structural analysis of these targets, under heat plasma loading as well as electromagnetic (EM) loads, are presented. The calculations, carried out in the environment of Ansys 2021R2 code, have taken into account the heat loading under plasma operation (with a peak heat flux of 10 MW/m^2) and the EM loads from a representative plasma disruptions (VDE 30 ms). Design criteria from RCC-MRx are used in the structural analysis in order to comply with nuclear requirements. RCC-MRx 2015 is used for the mechanical assessment of metallic components, while for the graphite the comparison between the maximum stress and the yield stress has been considered. The analyses performed show that the thermal load on target during the plasma heat loading drives target designs due to differential expansion at the interface between the heat sink and the tiles.

Keywords

JT-60SA, HHF, target, plasma heat load, electromagnetics, thermal, transient structural.

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PS4-2 ABSTRACT-a6b8
A. Plasma-Facing High Heat Flux Components

Pre-conceptual design of the steering mirror for the DEMO Electron Cyclotron Heating system

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An Electron Cyclotron (EC) Heating and Current Drive (HCD) system will be installed in the EU DEMO Tokamak to provide plasma heating and stabilisation by coupling up to 130 MW of mm-wave power into specific regions of the plasma. The mm-waves will be generated by gyrotrons and propagated through a quasi-optical multi-beam mirror system (MBMS), and subsequently through corrugated waveguides, to launchers located at dedicated Equatorial Ports (EPs). The present design of the launcher features two mid steering antennas (MSAs) for plasma stabilisation each consisting of three-beam layouts and steering mirrors positioned at the upper and lower locations of the EP.

The proposed concept for the steering mirrors operates on the pantograph principle, a mechanical assembly of parallel linkages, which translates the movement produced by a pneumatic actuator into an exact angular rotation of the mirror around an offset virtual axis. This design uses flexures with specific structural compliance as mechanical joints resulting in the desired movement of the linkage system, while avoiding the tribological difficulties associated with mechanisms operating under vacuum. This paper describes the present status of the DEMO EC HCD steering mirror as well as the analyses performed to validate this design.

This work has been carried out within the framework of the EUROfusion Consortium, via the Euratom Research and Training Programme (Grant Agreement No 101052200 - EUROfusion) and funded by the Swiss State Secretariat for Education, Research and Innovation (SERI). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union, the European Commission, or SERI. Neither the European Union nor the European Commission nor SERI can be held responsible for them.

Keywords

DEMO, EC HCD, steering mirrors, hydro-dynamic analysis, thermo-mechanical analysis.

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PS4-3 ABSTRACT-a9b2
A. Plasma-Facing High Heat Flux Components

Brazing alloys characterization for EU-DEMO Divertor Target

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In the framework of the EUROfusion Work Package of the DEMO Divertor, relevant efforts have been addressed to the technological development of Plasma Facing Components. A research activity has been undertaken at ENEA to support the development of technological solutions for monoblock-pipe joining in order to reduce the use of materials having high activation and/or degradation under neutron irradiation. For this purpose, a dedicated brazing alloy screening activity has been carried out. A total of seven brazing alloys have been identified and tested: GemCo, Nicuman23, TiCuNi, CuTiZrNi and three alloys based on different percentages of Copper and Germanium. For each brazing alloy, a wettability test on W and Cu specimens has been performed. Then, three mock-ups have been realized using the three different brazing alloys GemCo, Nicuman23 and TiCuNi, respectively. Each mock-up is composed of two W monoblocks without Cu interlayer joined with Wfiber-Cu composite pipes. Nondestructive examinations with Ultrasonic Testing (UT) performed on each mock-up showed that monoblocks surface was not fully attached. Other two mock-ups have been realized using GemCo and Nicuman23, respectively, joining the monoblocks on CuCrZr pipes. Each mock-up has two monoblocks, one with Cu interlayer and one without interlayer. The UT results on these latest samples have shown excellent results both in the case of the presence of interlayer and without. From the results, GemCo seems to be the most promising among the tested commercial alloys, thanks to its low amount of Nickel (i.e., low neutronic activation) and the good joining capabilities.

Keywords

DEMO Plasma Facing Components, Divertor target mock-ups, Divertor materials, brazing alloys characterization, nondestructive analysis.

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PS4-4 ABSTRACT-ae29
A. Plasma-Facing High Heat Flux Components

STEP Limiters – Manufacturing Trials of PFCs with Low Thermal Conductivity Features

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During transient plasma events, the Plasma Facing Components (PFC) of the Limiters will be subjected to Ultra High Heat Flux (UHHF) thermal loads of up to 300 GW/m^2 for millisecond durations. These PFCs must also withstand high steady-state thermal loads in the range of 1 – 5 MW/m^2, so require active cooling of the armour. These UHHF loads threaten the integrity of the coolant pipe and one design solution to this is to introduce a low thermal conductivity layer or thermal break, which acts to slow down the incoming thermal wave as it conducts through the material and reduces the peak temperatures seen by the coolant pipe. The introduction of these layers or barriers adds manufacturing complexity and introduce material joints which may lower the thermal limits of the component.

The objective of this paper is to introduce the low conductivity monoblock options being explored by STEP, covering design investigation and initial manufacturing trials. Three concepts under investigation are: the Thermal Barrier Limiter concept with a steel interlayer covering the coolant pipe, and two Layered Armour Limiter concepts with low-conductivity layers. The investigation demonstrated the potential viability of different routes to manufacture low-conductivity layers, whilst preserving the high temperature limits PFC. Further design and process refinements are proposed.

Keywords

Plasma Facing Components, High Heat Flux, Manufacturing, Limiters.

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PS4-7 ABSTRACT-c6da
A. Plasma-Facing High Heat Flux Components

Design, manufacturing and commissioning of endoscopes for monitoring water-cooled divertor in W7-X

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The modular stellarator Wendelstein 7-X (W7-X) in Greifswald (Germany) started operation in 2015 with short pulse limiter plasmas and continued with pulsed divertor plasmas in 2017-2018. In 2022, operation phase (OP) OP2 started after installation of 10 water-cooled CFC armored divertors, allowing for steady state operation.

Since divertor heat loads are very sensitive to plasma parameters, each water-cooled divertor needs to be monitored to interrupt or adapt plasma operation once overload is detected. For that purpose ten endoscopes are planned: two in the so-called AEA ports in module 3 and eight more in the AEF ports in the other modules. The infrared (IR) radiation from the plasma facing surface as well as the plasma edge radiation in the visible (VIS) range is captured through a pinhole in a water-cooled plasma facing head and transmitted to the rear side outside the vacuum where the light is split and captured by an IR and VIS camera. The challenge is to reach a high-resolution image of the entire target while capturing a large field of view (FOV) of 120 degrees.

In a previous paper [1], the design of the AEA endoscopes was presented. In the current paper, the manufacturing, commissioning and first operation results of the AEA endoscopes is shown as well as the design of the AEF endoscopes. This design includes improvements resulting from the experience of manufacturing and commissioning the AEA endoscopes.

[1] J. Fellinger et al.: Design of endoscopes for monitoring water-cooled divertor in Wendelstein 7-x, Fus. Eng. & Des., Vol. 158, Sep. 2020

Keywords

IR Thermography, Wendelstein 7-X, Optics, Additive Manufacturing.

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PS4-8 ABSTRACT-cd7c
A. Plasma-Facing High Heat Flux Components

Reflection Discrimination of Tungsten PFC with Full Ray Information

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Tungsten tends to be adopted for plasma facing component (PFC) due to its high energy threshold and its chemical stability [1]. However, tungsten was reported to have more than 90 % reflectivity at 3 μm of wavelength [2]. This reflective environment will cause diagnostic systems to lose their functions especially in infrared wavelength regions. This effect was already reported to cause overestimation up to 85 % in an infrared imaging system [3], and most diagnostic systems will have signal distortions. For this reason, the reflection of tungsten PFCs needs to be concerned. For its solution, the idea discriminating signals from reflections was proposed in PSI-25 [4]. It is the numerical method using full ray information, and the ray information was derived through the lab-built ray tracing program. The result showed the nice result of the reflection elimination through simulation, and the multiple cameras are suggested to get rid of dead zones (out of camera view angle). As a series of the topic, the study of the multiple camera will be presented here.

Reference

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Keywords

Tungsten reflection, tungsten PFC, tile temperature measurement, reflection discrimination, PFC ray tracing.

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PS4-9 ABSTRACT-cfa4
A. Plasma-Facing High Heat Flux Components

Electromagnetic analysis in support to the EU-DEMO Upper Limiter design

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The protective approach chosen for the European DEMOnstration (EU-DEMO) fusion power reactor involves the use of protruding limiters as sacrificial components to safeguard the first wall (FW) of the Breeding Blanket (BB). The limiters serve to protect the FW during the discharge flat-top and, more crucially, during planned and unplanned plasma transients. In the latter, limiters withstand not only the thermal energy generated by the plasma but also increased electromagnetic (EM) loads, which must be accounted for in the design and optimization of the limiter system. Four types of limiters are considered for positioning at specific poloidal and toroidal locations in the FW region, where direct contact with the boundary of the collapsing or moving plasma is expected. They include the upper limiter (UL), outer mid-plane limiter (OML), outer lower limiter (OLL), and inner mid-plane limiter (IML). This study focuses on the UL and its design evaluation through electromagnetic simulations using Finite Element Method (FEM) numerical techniques. The simulations involved modeling a 22.5-degree DEMO2017 sector, taking into account the main sources of the magnetic field and in-vessel components while approximating their complex geometry and material properties. Two designs for the upper limiter have been considered: a box-like configuration and a plate-like configuration. Simulations were performed to analyze both designs' response to an upper vertical disruption event (UVDE), identified as one of the most critical scenarios for such a component. The simulations evaluated the EM behavior of the UL, including the magnetic field distribution, induced currents, and ferromagnetic loads due to the presence of EUROFER. Impact of halo currents has also been assessed against the uncertainties in their magnitude and distribution, which depend on the in-vessel component design, electrical connectivity, plasma temperature, and halo width. The simulation results provided valuable insights into the performance and optimization of the upper limiter system.

Keywords

EU-DEMO, Upper Limiter, Electromagnetic loads, plasma disruption, UVDE.

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PS4-10 ABSTRACT-9285
A. Plasma-Facing High Heat Flux Components

Investigation of plasma plumes created during the exposition of a liquid tin filled-tungsten Capillary Porous System target at the OLMAT High Heat Flux facility

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Power exhaust remains as one of the main challenges on the pathway to magnetic fusion energy realization. Tungsten (W) is the material of choice for the ITER divertor. For its extrapolation to a DEMO-like device, the baseline plasma scenario will require a cold plasma edge (~ 5eV) and a very high and challenging (~95%) radiated power fraction [1] in order to assure the survival of the W elements (heat flux limits ~ 5-10 MW/m^2 at steady state [2]) during an economically viable lifetime. Liquid Metal (LM) Plasma Facing Components (PFCs), in particular tin, may potentially increase the power handling limits beyond the W capabilities while offering an option that can be more resilient to potentially catastrophic transient events such as disruptions [3]. The OLMAT (Optimization of Liquid Metal Advanced Targets) High Heat Flux (HHF) facility [4] is used to pursue the experimental development of alternative LM PFCs and enables the exposition of prototypes to heat fluxes from 5 to 58 MW/m^2 in pulsed operation (30-150 ms duration) with repetition rates up to 2 min^{-1}. This work presents the results of the experiments with a target composed of a tungsten felt structure filled with liquid tin in which a single Langmuir probe was embedded and successfully operated. It allowed diagnosing the plasma plume (electron temperature and density evolution) created in front of the target during the HHF exposition within a temporal scale in which the vaporization of tin dominated the plasma build-up, being high enough to induce shielding effects on the incoming heat fluxes. The global results of this characterization are addressed, being completed with line spectroscopy, fast-frame/infrared camera imaging and pyrometry (target thermal response) measurements, attempting the pioneering, in-situ diagnosis of these local, tin-enriched plasmas.

Keywords

Plasma facing Components, Liquid Metals, High Heat Flux Facility, Vapor shielding, Plasma characterization.

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PS4-11 ABSTRACT-e717
A. Plasma-Facing High Heat Flux Components

Thermal and structural analysis of the European Water-Cooled Lithium Lead breeding blanket concept in the L-H mode transition

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The design of a breeding blanket system (BB) which reliably fulfils the prescribed requirements is pivotal for the construction of the EU DEMO fusion reactor. For this reason, the BB conceptual design activities require a tighter interaction with plasma physics modelling, so to employ relevant boundary conditions to compare, and then down-select, the BB design alternatives. The preliminary thermo-mechanical study herein reported highlights the importance to assess in deep the BB performance during plasma operational transients. In particular, analyses of the BB thermal and mechanical transients during plasma ramp-up have been carried out, with focus on the L-H transition. Attention has been paid to the Water-Cooled Lithium Lead (WCLL) BB concept, one of the two driver blanket candidates for the EU DEMO project. Adopting DEMO-relevant plasma configurations, the fusion power and the radiative power time evolutions during the plasma ramp-up phase have been modelled under different assumptions using the code ASTRA. The obtained power profiles have been used to calculate the time-dependent nuclear power density and radiative heat flux the WCLL BB is subjected to, in order to perform the corresponding transient thermal analysis. Then, the structural analysis has been carried out in order to verify the fulfilment of the RCC-MRx structural design criteria in the most critical time steps. The performed thermo-mechanical analysis have been carried out adopting the Abaqus FEM code. Results obtained are presented and critically discussed, highlighting their importance for the EU DEMO integrated design.

Keywords

DEMO, WCLL BB, L-H mode transition, plasma ramp-up, thermo-mechanics, transient analysis.

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PS4-12 ABSTRACT-ea5c
A. Plasma-Facing High Heat Flux Components

Uncertainty quantification of a multi-physics model for in-vessel components design in a fusion power plant

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Future fusion reactor designs will heavily rely on numerical simulations for improvements in efficiency, safety, and reliability. Plasma-facing components (PFCs) in particular, must withstand harsh complex loading scenarios such as extreme thermal loads combined with repeated intense thermal shocks. While ITER and CHIMERA will act as first test devices for PFCs in the next years, the historical data currently available is limited and experiments focus on the most essential material properties. Typical simulations of subsets of relevant multiphysics phenomena are not acceptable anymore and efficient multiphysics coupled simulation approaches that are flexible and easy-to-adopt (hence using verified and validated reliable tools) are needed instead.

In this study, to the best of the authors' knowledge the first example of a workflow to effectively study PFCs coupling the same models analysts developed in their single physics analyses is presented that includes neutronics, thermal and hydraulics physics. The workflow allows sensitivity analyses to condense the range of parameters of interest, calibrations of parameters to experimental data, and uncertainty quantifications to provide best estimate predictions with confidence bounds as increasingly demanded from nuclear regulations. The pros and cons of the approach are presented with the current achievements and challenges to overcome in the coupling of multi-physics models. The use of pre-existing models allows the reduction of workflow development times and faster design explorations with fusion reactors entering the manufacturing stage.

Keywords

Numerical simulations, multi-physics simulations, workflow integration, sensitivity analysis, experimental calibration, uncertainty quantification.

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PS4-13 ABSTRACT-ebec
A. Plasma-Facing High Heat Flux Components

Concept Design of the STEP Outboard Build

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The development of the STEP outboard build is a fundamental challenge for the successful realisation of the STEP tokamak. The outboard build includes the outboard limiters, first wall, divertors and blankets. The performance of these components is closely linked such that the design and integration of these components must be done as a combined effort to ensure the performance of each component is balanced so as a whole, they can meet their requirements. The outboard build design must consider requirements of service connection routing, remote maintenance, structural load paths, thermal management and coolant operating parameters for power generation. All these requirements must be balanced to produce a successful integrated design. Here, we will present the current concept design of the STEP outboard build, it's predicted performance based on the integrated analysis and discuss how the competing design requirements were balanced.

Keywords

Divertor, First Wall, Limiters, Blanket, Plasma Facing Component.

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PS4-14 ABSTRACT-ec33
A. Plasma-Facing High Heat Flux Components

Design and Modeling of a Dissipative Divertor that Isolates the Target from the X-point in DIII-D

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The design of a dissipative divertor with the objective to enable experiments that quantify the mechanisms behind divertor detachment in regimes close to that of a tokamak reactor is presented using the boundary simulation code SOLPS-ITER [1]. Such a divertor is slated for installation in the DIII-D tokamak in a modular manner during the upcoming 5-year plan. Divertor designs are simulated in deuterium plasmas with intrinsic carbon impurities from graphite walls with total injected power of 25 MW, as future DIII-D auxiliary heating is expected to be comprised of approximately 15 MW NBI and ECH upgraded to 10 MW. Baffling in the SOL and private flux region is varied to investigate the impact of closure, with an outer poloidal leg length of ~50 cm consistent with a stable T_e gradient of 200 eV/m and maintaining ~1 eV at the targets and >80 eV at the X-point. Profiles of cross-field particle and thermal transport coefficients are developed based on constraining the ratio of the separatrix density to the pedestal density in the range of 0.5 to 0.7, a core density in attached cases of ~60% of the Greenwald density, and a SOL heat flux width of 2.0 mm given by the ITPA scaling. Divertor performance is evaluated based on metrics including the peak heat flux at the divertor target, T_e near the strike point, the stability and location of the radiation front, and the ability to reach deep detachment with impurity seeding and pumping. These results provide insights into divertor design, the physics of detachment, and power exhaust solutions.

Work supported by US DOE under DE-FC02-04ER54698.

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Keywords

Tokamak Divertor, SOLPS-ITER, Detachment, PFC, SOL, Power Exhaust, Boundary Plasma Physics, Scrape Off Layer.

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PS4-17 ABSTRACT-1c5c
A. Plasma-Facing High Heat Flux Components

Pre-conceptual design of the fixed mirrors for the DEMO Electron Cyclotron Heating antenna

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The Electron Cyclotron Heating system (ECH) of the DEMO nuclear fusion reactor is foreseen to be used as the primary source for heating, current drive and instability control of the plasma. The required mm-wave power is ensured by the installation of a sufficient number of gyrotron sources. The mm-wave beams generated in the gyrotrons are guided to the dedicated Equatorial Ports (EP) through quasi-optical transmission lines between the facility buildings and corrugated waveguides.

Each launcher unit comprises twenty-two beam-lines, in both fixed frequency (FF) and tuneable frequency (TF) variants. In the FF variant of the mid-steering antenna (MSA), two groups of three beams provide plasma instability control of Neoclassical Tearing Modes (NTMs), with two sets of launching mirrors (one fixed plane mirror and one focusing steering mirror). For the TF variant, all mirrors are fixed and the frequency of the launched beam is tuned.

Additionally, two groups of eight beams are available for plasma bulk heating (BH), current drive (CD) and thermal instability (TI) control, with one fixed plane mirror and one fixed focusing mirror. These two Multi Beam Mirror (MBM) antennas, operating with both the FF and TF gyrotron configurations, are positioned symmetrically relative to the equatorial mid-plane and aim at the plasma centre.

The mirrors are designed to withstand the thermal loads generated during operation, including neutron heating, plasma radiation, ohmic losses and stray radiation. To fulfill the design requirements, efficient heat removal strategies based on water cooling principles minimising pressure drop are implemented.

This paper presents the pre-conceptual design of the first MBM and MSA mirrors, validating the structural and geometric boundary conditions. Using FEA tools, thermo-hydraulic and thermo-mechanical simulations were performed with the available thermal loads to optimize the cooling parameters and perform a first stress assessment on the two mirrors.

Keywords

DEMO, Multi Beam Modules, Mid-steering Modules, thermal-hydraulic simulations, thermo-mechanical simulations.

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PS4-19 ABSTRACT-C2b7
B. Blanket Technology

Blanket test facility for mockup testing on water cooled ceramic breeder blanket system

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Intensive research and development activities have been conducted for design of a water-cooled ceramic breeder (WCCB) blanket system. In this system, 598K/15.5MPa of pressurized water removes heat injected and accumulated in blanket containers made of a reduced activation ferritic/martensitic steel, F82H. The container has lithium ceramics and beryllium pebbles as tritium breeder and neutron multiplier, respectively. The blanket system will be operated under conditions unique to fusion environment and WCCB blanket system: coolant that correspond to those of pressurized water as mentioned above, high heat flux (HHF) due to plasma bombardment, risk of coolant leakage and beryllium-water reactions in the accident. To contribute to further development of blanket systems, experimental apparatus have been installed in Rokkasho Fusion Institute, National Institutes for Quantum Science and Technology. The apparatus enables to conduct the following experiment on full-scale mockups: 1) HHF test to demonstrate heat removal of surface heat flux by electron beam gun simulating plasma, 2) flow assisted corrosion, and 3) coolant ingress into dummy pebble beds. A desk-top apparatus for beryllium alloy-steam reaction rate measurement with sample holder with 100 mm height and 32 mm inner diameter have been installed. The electron beam gun bombards container mockup and available to a range of 1600 × 500 mm^2. The HHF apparatus is jointly operated with high temperature and high pressure water cooling system which provides Max. 0.9 kg/s of flow rate. The specification of mockups and overview of experimental result will be stated in the presentation.

Keywords

Pressurized water, water-cooled ceramic breeder, full-scale mockup, high heat flux, flow assisted corrosion, coolant leakage.

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PS4-20 ABSTRACT-c304
B. Blanket Technology

Development and characterization of electrochemical hydrogen sensors using different fabrication techniques

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Hydrogen isotopes measurements will be a key point for future fusion reactors to be able to operate properly. Electrochemical sensors are a good option because they are practical and economical and they have low power requirements, remarkable repeatability and accuracy, among others. However, finding suitable materials that can operate under harsh environments is one of the numerous challenges for hydrogen detection. Perovskite-type materials have interesting properties for this type of application such as high proton conductivity, good chemical stability and mechanical strength, among others. Because of these characteristics, perovskite materials are excellent candidates for the development of high-temperature hydrogen sensors. Usually, these types of materials are typically pellet-shaped using uniaxial pressure. Other geometries, such as crucibles, can enhance the design and performance of the sensors. Ceramics with complex geometries can be prototyped using either conventional methods, such as cold isostatic pressure (CIP), or novel technologies, such as ceramic 3D printing, which offers the advantage of quick prototyping.

This work describes the development of electrochemical hydrogen sensors based on perovskite-type ceramic BaCe0.6Zr0.3Y0.1O3-α (BCZY). Two different technologies were used: CIP for obtaining crucible-shaped samples and 3D printing for bullet-shaped elements. The sintered samples were characterized by XRD and SEM. Then, electrochemical sensors were assembled. To do so, both sides of the electrolyte were coated with platinum ink in order to act as the working (the external part of the sensor) and the reference electrodes (the inner part of the sensor). Finally, the response of the sensors was evaluated at 400, 500 and 600 ºC using hydrogen calibration mixtures in argon in a potentiometric mode. These results suggest that both, CIP and 3D-printed BCZY sensors have the ability to detect hydrogen in these environments, enabling a game-changing solution for monitoring fusion processes which requires the quantification of hydrogen isotopes

Keywords

Hydrogen sensors, solid electrolytes, 3D printing.

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PS4-21 ABSTRACT-0331
B. Blanket Technology

Low pressure hydrogen isotope transport parameters in materials for tritium recycling systems.

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Metals from group V such as vanadium and niobium have been proposed for the construction of tritium recycling systems and related technologies for liquid metal breeder blankets, especially because of their nominally high tritium permeability values. This selection has been made according to the traditionally accepted values of hydrogen/deuterium permeability in these materials, which were calculated by multiplying solubility and diffusivity data. However, it has not been possible to reproduce these numbers in a laboratory so far, most probably due to both the extrapolation of the calculated values to an inadequate range of temperatures and the difficulty for avoiding surface effects. The experimental determination of H isotope permeability in these materials has become a priority for the development of the Permeation Against Vacuum (PAV) and other tritium extraction concepts, and new hydrogen and deuterium permeability coefficients have been this way provided at least for vanadium [1].

Given the potential importance of the surface effects and the expected low tritium pressures and high temperatures, the characterization of the surface limited permeation regime is necessary in order to fully describe the H isotope transport properties of these materials.

The determination of the adsorption and recombination constants for hydrogen and deuterium in vanadium and niobium, for which only very scarce and scattered data can be found in the literature, is proposed. For this, permeation experiments at low driving pressures are carried out at two different and complementary facilities, what not only provides a *cross check* validation of the results, but also permits to extend the versatility of the experiments. Surface composition and microstructural analyses are also performed in order to determine the influence of grain boundaries, surface state, etc. in the permeation values.

[1] **M. Malo** et al., Membranes 12 (2022) 579. https://doi.org/10.3390/ membranes12060579

Keywords

H isotope transport parameters, niobium, vanadium, surface limited regime.

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PS4-22 ABSTRACT-0653
B. Blanket Technology

Recent Progress of Fusion Nuclear Safety and Nuclear Materials at INEST, CAS

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Tritium safety is one of the key issues for fusion reactor, and tritium should be well controlled to prevent it from excess release into the atmosphere and exposure to workers. Numerical simulation of tritium transport behavior under multiple conditions were conducted including normal operation, incident and accident, to support the detailed tritium confinement system design of tritium plant for China Fusion Engineering Test Reactor (CFETR). A series of experiments were carried out to study the explosion characteristics and mechanism of the pure tungsten dust and hydrogen/dust mixtures with a 20 L spherical vessel and a 5 L closed combustion tube, providing a reference for the accident prevention of the future fusion reactor. In the area of fusion structure materials, laser powder bed fusion additive manufacturing of ODS-RAFM steel by prefabricated multi-component nano-oxides are developing to fulfill the requirement of TBM' complex structure, and improved types of RAFM steels and high entropy alloys have been developed based on new design strategies in order to fulfill the requirements for higher service temperature and irradiation dose of commercial fusion reactor in future. As for functional materials, two kinds of two-dimensional nanomaterial composite proton exchange membranes, Graphene-Nafion and hBN-Nafion, were prepared by the solution casting method to simplify the preparation process of protium composite membrane used in protium separation two-dimensional material for engineering application. Moreover, an international big science plan cultivation project called ALIANCE (Axisymmetric LInear Advanced Neutron source) was launched and is under active research.

Keywords

Tritium safety, Dust explosion, Additive manufacturing of ODS-RAFM, Improved types of RAFM steels and high entropy alloys, Proton exchange membranes, Neutron source.

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PS4-24 ABSTRACT-d089
B. Blanket Technology

The high chemical compatibility with static PbLi of oxide ceramic candidates for the advance DCLL

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In the framework of FP8 and FP9 EUROfusion activities, alumina and zirconia ceramics have been characterized at CIEMAT for fusion applications as insulating PbLi channel components for the advanced Dual Coolant Lead Lithium Breeding Blanket (DCLL BB). The electrical and thermomechanical performance were tested at close to operational conditions (temperature, thermal gradient). Nevertheless, the long-time testing in contact with hot PbLi had not yet been addressed to assure their chemical compatibility with the flowing liquid metal. Therefore, the oxide ceramic samples were exposed during 1000 hs to PbLi at 600ºC in the COrrosion Experiment at Static conditions (COES) located in the Liquid Metal Lab at LNF-CIEMAT. Changes in the microstructure of the solid treated material were identified by means of scanning (transmission) electron microscopy (SEM and STEM) on cross sectional preparations. To conclude on the surface chemical compatibility with PbLi, the elemental composition within the surface region in contact with the liquid metal was analyzed by the microscope-coupled EDX detector, and through elemental depth profiles using secondary ion mass spectrometry (SIMS) technique. As expected for highly resistant ceramics, the materials slightly undergo dissolution of the grain boundary component in the first microns from surface, likely due to reaction with the diffused lithium.

Keywords

Lead-Litium, Pb-Li, Breeding Blanket, secondary ion mass spectrometry (SIMS), Oxide ceramic, Liquid Metal Lab (LML), COrrosion Experiment at Static conditions (COES).

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PS4-25 ABSTRACT-d0e7
B. Blanket Technology

Procurement Status of ITER Blanket Shield Block in China

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Shield blocks (SB) are important components in ITER Blanket system, providing thermal and neutron shielding to the vacuum vessel and external vessel components. There are a total of 440 shield blocks in ITER Blanket system, 220 of which are procured by China.

The first step in the procurement activities is the process qualification through the manufacturing and testing of a full-scale prototype (FSP). In June 2015, China DA started manufacturing a 9-ton 316L(N) ITER grade stainless steel forging for the FSP, testing results including chemical content, mechanical testing result and non-destructive testing are all accepted by IO. In August of 2016, the FSP manufacture started; steps included machining datum, drilling deep holes, side machining, welding cover plates, and final machining. In the beginning of 2018, China DA completed all test including the most important the hot helium leak test (HHLT) of the FSP, after HHLT, final dimensional inspection was done in March of 2018.

Since mid of 2018, China DA started the series production of SBs, starting from SB10 products. By now (end of July 2021), China DA has done with more than 190 pieces of forgings, whose production and testing results has been accepted by IO. China DA also has completed all manufacturing and testing on 50+ products. By 2022, China DA has completed 49% of the procurement.

R&D work carried out during the procurement activities. China DA developed two facilities for HHLT, whose sensitivity can reach $10*10^{-12}$ Pa*m^3/s level at 250 ºC, performance better than ITER requirements ($4*10^{-11}$ Pa*m^3/s), also could be utilized for inspecting the vacuum performance of large components. China DA also completed R&D work on low friction coating. R&D result shows that if roughness of metal surface is less than 0.2um, on which the LFC friction is less than 0.1 within testing distance of 2m.

Keywords

ITER, Shield Block, Procurement Status, R&D work.

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PS4-27 ABSTRACT-d5c9
B. Blanket Technology

Introduction on Tritium Transport Analysis model for HCCP Breeding Blanket System

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In implementing the He-cooled Ceramic Pebble breeding blanket system, many design elements and experiments must be supported, and a development of tritium transport analysis model is a key element for safety and design of the breeding blanket system. Because a tritium release into environment from the breeding blanket system can be a radioactive risk, the accurate calculation and prediction shall be preceded to prevent a fusion reactor incident.

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

THETA-FR is modelled from the integration of Matlab Simulink/COMSOL multiphysics. It calculates a dynamic hydrogen isotopes (H/D/T) transport phenomena in the HCCP breeding blanket system and predicts the transient tritium retention, permeation and release amount into the environment from the breeding blanket system.

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

Keywords

Breeding Blanket System, Tritium Transport Analysis Model.

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PS4-29 ABSTRACT-e269
B. Blanket Technology

Validation of V, Si and Ni cross sections from trial versions of JEFF-4 ND library on fusion-relevant shielding benchmarks

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The aim of this contribution to the validation of the JEFF-4 test libraries on fusion-relevant neutron shielding experiments was to complement the established validation procedures, therefore the effort was focused on materials of secondary importance for fusion, i.e. V, Si and Ni. The following benchmark cases were analysed: IPPE V, FNS V, FNG SiC, TUD SiC, Oktavian Si, Oktavian Ni. MCNP inputs, provided with the latest release of the SINBAD database were taken as a basis. Some of them were modified, e.g., for the IPPE V experiment, the simulation was performed in time rather than energy domain in order to more directly recreate the measurement. Using these modified inputs, the results obtained with JEFF-4 trial versions 0 and 1 were compared with results obtained with existing (i.e. already officially released) ND libraries JEFF-3.3, JENDL-4.0 and ENDF/B-VIII.0. JEFF-3.3 was taken as a reference and base library, i.e. for materials other than the studied material. In addition, a comparison of evaluated and experimental data for important reaction cross sections for V-51, Si-28, Ni-58, and Ni-60 was performed. The results indicate that in some cases JEFF-4T1 does not perform as well as other ND libraries. In such cases, it is recommended to revert back to JEFF-4T0 (V-51(n,tot), Ni-58(n,tot)) or JEFF-3.3 (Si-28(n,tot), Ni-60(n,tot)) or to perform a new evaluation or energy-dependent measurements (V-51(n,n'), Si-28(n,α)). For the future, continuation and extension of this work is foreseen. Especially the computational model of the TUD SiC experiment requires additional improvements.

Keywords

Fusion neutron shielding, nuclear data, benchmark experiments.

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PS4-30 ABSTRACT-e622
B. Blanket Technology

Solubility of helium in liquid Pb-16Li

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Research Centre Rez

Tritium self-sufficiency is a key aspect for fusion reactors such as European DEMO relying on the D-T fusion reaction . The tritium production in breeding blankets is based on neutron reaction with lithium which however produces almost equal amount of helium as a by-product. In the breeding blankets concepts, which count on the use of liquid Pb-16Li alloy as breeding material, the produced helium may exceed its solubility limit. Helium bubble formation in a supersaturated solution within the breeding zone would have potential impact on blanket structure cooling as well as on tritium desorption.

An apparatus for determining helium solubility in Pb-16Li is currently being built at Research Centre Řež. The estimated detection limit for the optimized apparatus design is approximately $1 \cdot 10^{-14}$ mol He/mol Pb-16Li. The aim of this contribution is to report on the commissioning of the apparatus. The tests include saturating the liquid Pb-16Li by helium at various temperatures and pressures and finding optimal conditions for helium desorption in the detection chamber to allow evaluation of the helium solubility in the Pb-16Li alloy.

Keywords

Breeding Blanket, Pb-16Li, Helium, Solubility.

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PS4-32 ABSTRACT-e9c6
B. Blanket Technology

Deuterium detection using electrochemical sensors

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The measurement of hydrogen isotopes will be of great interest for future fusion reactors to ensure their proper operation. For this reason, electrochemical sensors will be suitable tools for tritium quantification, as they can perform on-line and in situ measurements and thus prove the self-sufficiency of the breeding systems. One of the many challenges in hydrogen sensing is finding materials suitable for use at high temperatures and in aggressive environments. In this regard, perovskite-type ceramics, such as $BaCe_{0.6}Zr_{0.3}Y_{0.1}O_{3-\alpha}$ (BCZY), exhibit high proton conductivity and excellent physical and chemical stabilities. These properties make perovskite materials ideal candidates for the development of high-temperature hydrogen sensors. In the present work, BCZY electrolyte was used to construct amperometric sensors for hydrogen isotopes monitoring.

Firstly, BCZY powder was synthesized by the solid-state method and characterized using SEM and XRD. Secondly, it was sintered as a pellet and bound to an alumina tube to construct the electrochemical sensors. Both sides of the electrolyte were platinized and connected to platinum wires to act as working (the external part of the sensor) and counter electrodes (the inner part of the sensor). Finally, amperometric measurements were performed inside a stainless-steel reactor at 350, 400 and 500 °C and applying 0.15 V between electrodes. The response of the sensors was evaluated using hydrogen and deuterium calibration mixtures in a concentration range from 150 to 300 ppm H_2 or D_2 in Ar. Finally, measurements with hydrogen and deuterium were compared to determine the isotope effect on the response.

Keywords

Hydrogen isotopes, electrochemical sensors, deuterium, solid state electrolytes.

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PS4-33 ABSTRACT-ea55
B. Blanket Technology

Study on Hot Isostatic Pressing conditions of ARAA for fabrication of the Korean breeding blanket first wall

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The helium cooled concept with solid breeding materials is one of the candidates for Demonstration fusion power reactor (DEMO) breeding blanket in Korea. The first wall (FW) is a key component of the breeding blanket. The FW should be operated while keeping the structural integrity under the severe condition which is subject to high surface heat flux, neutron wall load and high coolant pressure simultaneously. Various fabrication methods have been studied for the development of the FW manufacturing technologies. For one, a FW fabrication method with combination of the drilling and bending is very simply way to make only circular cooling channels using existing technologies with low cost. But rectangular cooling channels which have larger heat transfer area are more preferable. Hot isostatic pressing (HIP) bonding is considered as the most promising method for achieving such complex shape. The HIP bonding was applied to fabricate the FW with rectangular cooling channels in this study. The structural material of the FW is ARAA which is a reduced activation ferritic/martensitic steel under development in Korea. Prior to fabrication of the FW mock-ups we investigated the HIP bonding conditions of ARAA. Temperature, holding time and pressure were considered as the parameters to optimize the HIP bonding conditions. High temperature over 1000℃ during the HIP process led to changes of microstructures and hardness. Some local bonding lines were observed at HIP joints. The microstructures and hardness were recovered and the local bonding lines disappeared after post heat treatment. Mechanical properties of HIP joints were assessed by performing the tensile tests. The avg. tensile strength of the HIP joint was about 3% lower than that of the base metal. The Charpy impact tests will be also conducted. Full scale mock-ups of the FW will be fabricated with the optimized HIP bonding conditions.

Keywords

HIP joint, Mechanical properties, Post heat treatment.

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PS4-34 ABSTRACT-f2dc
B. Blanket Technology

Impact of Nuclear Heating Distribution on the structural behavior of ITER FW Panel 11

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The ITER blanket system is the innermost part of the Tokamak, and it is devoted to protecting the vacuum vessel and the ITER components against thermal and nuclear loads from the plasma. Each blanket consists of two parts: (a) the First Wall (FW) panel, which faces the plasma, and (b) the Shield Block (SB), which provides the bulk of the shielding. The FW panels are made of Beryllium tiles welded on a CuCrZr layer and supported by a stainless-steel structure. They are cooled by means of a pressurized water circuit at 40 bar and an inlet temperature of 70°C.

In view of the upcoming serial manufacturing of the ITER FW panels, Fusion for Energy (F4E) is updating the thermal and mechanical analyses of the baseline model for FW panel 11. This FW is subject both to surface heat flux up to 2 MW/m2 due to plasma radiation and nuclear heating (NH), as well as to electromagnetic loads. Regarding the NH, the baseline load corresponds to a distribution computed previously with a 2D simplified and homogenized model with a limited number of details. Recently, a new set of nuclear heating data have been computed with a 3D fully heterogeneous MCNP E-lite model which includes a fully detailed representation of the geometry and materials. The integral heating in the FW with this last model constitutes a 25% reduction compared to the 2D counterpart, i.e., 470.2 kW (3D distribution) and 623.1 kW (2D distribution).

Therefore, several thermal and mechanical analyses have been performed with the commercial Finite Element software ANSYS considering both 2D homogenous and 3D heterogenous nuclear heating data. Responses are then compared and thus the possible impact of the two distributions on the thermal and mechanical behavior of the FW panel 11 is assessed.

Keywords

Finite Element, First Wall Panel, Nuclear Heating.

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PS4-35 ABSTRACT-f7a5
B. Blanket Technology

Electrical properties of ceramic coatings after heavy-ion irradiation and lithium implantation

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Tritium permeation barrier has been investigated using ceramic coatings for nearly a half century to ensure fuel efficiency and radiological safety in fusion reactor blankets. Our recent studies elucidated irradiation and corrosion effects on microstructure of the coating, electrical conductivity, and hydrogen isotope permeation toward the practical application to the blankets. In a liquid lithium-lead blanket concept, recoil atoms of lithium flowing in the blanket channels and iron from the structural materials by collision with high-energy neutrons will be implanted in the coating, resulting in change in the coating properties. However, lithium implantation effects on the properties of the ceramic coatings have not reported yet. Therefore, this study focuses on electrical conductivity for the ceramic coatings after heavy-ion irradiation and lithium implantation.

Yttrium oxide and zirconium oxide coatings with approximately 700 nm in thickness were fabricated on reduced activation ferritic/martensitic steel F82H plates by magnetron sputtering and metal organic decomposition, respectively. After deposition of 4-mm^2 platinum electrodes, 250-keV lithium ions were implanted into the coatings with an averaged concentration of 1200–12000 appm. Some of the coatings were damaged by 6-MeV nickel ions with a damage concentration of 1 dpa before lithium implantation. Electrical conductivity before and after the irradiation and implantation were measured at room temperature by electrical impedance spectroscopy.

The as-coated samples showed the electrical conductivities of 10^{-8}–10^{-6} S m^{-1} for the yttrium oxide coating and 10^{-9}–10^{-7} S m^{-1} for the zirconium oxide coating. After lithium implantation, the samples showed an increase in the conductivity by 1–2 orders of magnitude and no dependence on lithium concentration. On the other hand, the lithium-implanted samples after heavy-ion irradiation showed similar conductivities of 10^{-4}–10^{-2} S m^{-1}, indicating that the conductivity reached a saturation. In the presentation, the conductivities at elevated temperatures and other coating properties will be discussed.

Keywords

Coating, Irradiation, Implantation, Electrical conductivity.

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PS4-36 ABSTRACT-f826
B. Blanket Technology

Lumped Parameter Porous Flow Modeling of Solid Tritium Breeding Blanket Materials

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Closing the fusion fuel cycle is an integral part of transitioning fusion science towards fusion energy production. Due to tritium scarcity, tritium breeding blankets are necessary for fuel production. Among several different tritium breeding blanket concepts, solid breeder materials are considered for use in configurations such as cellular ceramic breeders or pebble beds. Transport of tritium out of the ceramics is one of the primary design considerations, and in this work we explore approaches to tritium transport modeling in porous media using various tools within the Multiphysics Object-Oriented Simulation Environment (MOOSE) framework to produce a "hybrid fidelity" approach to design and analysis. The Navier-Stokes Module within the framework is used for Computational Fluid Dynamic (CFD) simulations to generate boundary and flow conditions. Then the Thermal Hydraulics Module (THM) is used to generate 1D, 2D, and 3D lumped parameter models for each the cellular breeder block and pebble bed designs, which are compared to literature-based correlations and more detailed models using the Porous Flow Module. The performance of the lumped parameter models for each concept is evaluated by studying the overall mass transport and pressure variations in the range of tritium breeding blanket operating conditions. The success of the modeling approach is discussed as well as the performance of each design relative to each other both in the simulated analysis and comparisons to literature.

Keywords

Porous flow modeling, tritium breeding blanket, lumped parameter.

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PS4-37 ABSTRACT-f8d8
B. Blanket Technology

A second-order partial diffusion equation model to compute the diffusion of hydrogen deuterium and tritium assisted by stress and strain fields

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In the first wall of fusion nuclear reactors, hydrogen and the hydrogen isotopes, deuterium and tritium, are present in different ways. On one hand, tritium and deuterium are injected from plasma and tritium is also generated in the breeding blanket. On the other hand, hydrogen is generated by reactions n-p, in addition, corrosion causes the appearance of hydrogen at the steel-water interface and, even, the radiolytic decomposition of water coolant is prevented by adding hydrogen. Thus, the presence of hydrogen in the blanket (first wall) of a fusion nuclear reactor causes hydrogen accumulation that affects the structural integrity of the material in the form of hydrogen assisted failure related phenomena (hydrogen embrittlement, hydrogen assisted cracking). In such damage processes, hydrogen transport by diffusion is a key stage which is governed by the gradient of hydrostatic stress and the gradient of material hydrogen solubility that is one-to-one dependent on plastic strain undergone by the blanket material. This way, the existence of residual stress produced by cumulative plastic strains caused during metal conforming process or mechanical pre-damage are taken into account. Within this framework, in this paper a second-order partial equation model is developed in order to quantify the hydrogen diffusion assisted by stress and strain in applications in fusion nuclear technology with the aim of improving the structural integrity assessment of these key structural components by evaluating the hydrogen transport by diffusion towards potential hydrogen-damage places in the blanket.

Keywords

Fusion nuclear technology, blanket, first wall, diffusion of hydrogen deuterium and tritium, stress and strain fields, residual stress.

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PS4-38 ABSTRACT-fa16
B. Blanket Technology

Electrical conductivity of candidate ceramic materials for flow channel inserts

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Magnetohydrodynamic pressure losses are one of the key issues in development of Dual Coolant Lithium Lead (DCLL) breeding blanket (BB) concept for European demonstration fusion power plant EU DEMO. Flow channel inserts (FCI) are functional components embedded along the Pb-16Li channels inside the DCLL BB modules. Their purpose is to minimize the magnetohydrodynamic pressure drop by breaking the current loop between the BB module structure and the liquid metal. A simplified design of FCI component consist of two layers - steel and ceramic. In this arrangement, the ceramic is directly exposed to the Pb-16Li liquid metal at the operating temperature of DCLL BB. Therefore, the study of electrical conductivity performance of the proposed ceramic materials at elevated temperatures is crucial for the DCLL BB design.

A series of direct electrical current measurements were performed on various ceramic materials by means of three electrode arrangement, with two working electrodes on the opposite sides of a flat sample and one guard electrode. The electrical conductivity was determined during thermal cycling in the temperature range from 50 to 550 °C in inert, oxygen free atmosphere. The results show that the obtained values are well below the electrical conductivity limit required for FCI components.

Keywords

Electrical conductivity, Flow channel inserts, Ceramics, Dual Coolant Lithium Lead, breeding blanket.

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PS4-40 ABSTRACT-fe7c
B. Blanket Technology

Simulation study of tritium diffusion in the liquid lithium lead eutectic alloy.

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We have studied through large-scale molecular dynamics simulations the temperature dependence of the diffusion coefficient of atomic and molecular tritium in the liquid Li15.7Pb84.3 alloy. The interaction between the components of the system (tritium, lithium and lead) have been described using a neural network potential, which was previously trained so as to reproduce the results of smaller scale simulation samples comprising the same components studied by first principles simulations.

Keywords

Tritium diffusion, liquid Li-Pb eutectic alloy, molecular dynamics simulation, neural network potentials, first principles simulations.

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PS4-41 ABSTRACT-b997
C. Fuel Cycle and Tritium Processing

Design and development of a hydrogen pellet centrifuge accelerator for the JT-60SA

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SENER Aeroespacial is developing a state-of-the-art hydrogen pellet centrifuge for the JT-60SA Tokamak under contract from Fusion for Energy (F4E), with support from Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT). This collaboration leverages the valuable expertise of the Max Planck Institute for Plasma Physics in the creation of previous generations of centrifuges in order to modernize and advance the concept towards a tool with multi-actuation capabilities. The project is currently undergoing a final design review, with the imaging system's qualification models being developed.

JT-60SA will be the world's largest experimental device for the study of thermonuclear reaction in a fully ionised hydrogen/deuterium plasma confined in a Tokamak configuration. The JT-60SA Pellet Launching System (PLS) serves the purpose of delivering various hydrogen pellets -mm-sized solid bodies - to the plasma.

The centrifuge accelerator within the PLS is designed to receive pellets from up to three different sources at a total rate up to 80 Hz and accelerate them to achieve speeds between 100 and 600 m/s using a centrifugal arm rotating at a frequency of 20 to 120 Hz. Since it is designated for varying tasks, arriving pellets will be different in size and potentially as well in their consistence. In order to facilitate monitoring of the acceleration process, the system is equipped with three high speed imaging diagnostic systems. These systems are positioned after the pellet sources, in the entrance funnel following pellet acceleration, and in the inspection chamber just prior to the pellet leaving the centrifuge skid.

This communication summarizes the driving requirements of the JT-60SA centrifuge accelerator, the design solutions being developed by SENER and the current development status.

Keywords

Pellet, Centrifuge, Accelerator, Hydrogen, Deuterium, Fuel.

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PS4-42 ABSTRACT-be4e
C. Fuel Cycle and Tritium Processing

Continuous tritium extraction from falling LiPb droplets in a vacuum

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Continuous tritium extraction from falling Lithium-Lead (LiPb) droplets in a vacuum was experimentally measured. Purpose was to confirm the feasibility to the continuous tritium extraction system for fusion fuel cycle. The extraction efficiency, stability, and trouble-free running were verified.

Campaigns were performed on the liquid metal test loop (Oroshhi-2) at National Institute for Fusion Science (NIFS). Tritium breeding in a liquid LiPb blanket was assumed by dissolving a deuterium through membrane in a circulating LiPb. The extraction efficiency was measured by the deuterium flux released from LiPb droplets while falling in a vacuum chamber. It is predicted that the deuterium release rate is maintained same level without degradation throughout the operation. Because new droplets are incessantly generated from circulating liquid LiPb through nozzles. The droplet surface is fresh, scarcely contaminated by incident molecules. Experimental conditions were as follows, the LiPb flow rate of 0.3 L min^{-1}, the nozzle diameter of 1.0 mm by 4 nozzles, the falling height of 0.5m, the temperature of 350 degree C, and 10 hours continuous operation.

Obtained extraction efficiencies were between 0.6 and 0.7 throughout the operation. Corresponding deuterium dispersion coefficient in LiPb droplet was approximately $2.0\times10^{-7} m^2 s^{-1}$. These outcomes agreed with results of the short period proof-of-principle experiment in 2013. It was performed in a laboratory with a nozzle, and 1 minute single shot operation. It is considered that the extraction efficiency of this method is depend on the release rate of each single droplet while falling. It is independent of the flow rate. This suggests the scalability to design a larger size extraction device.

Next challenge is the compatibility with DEMO reactor requirements, particularly the extraction efficiency and the LiPb flow rate. Potential degradation by multi droplets interference effects must be clarified.

Keywords

Tritiuim, extraction, efficiency, Lithium Lead, LiPb, droplets,.

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PS4-43 ABSTRACT-c5b8
C. Fuel Cycle and Tritium Processing

Tritium Compatible Vacuum Pump Train for Plasma Exhaust

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Plasma chambers for magnetic confinement requires pumping divertor region where magnetic field, radiation and heat flux should be considered. For pulsed inertial type confinements, wet wall and possibly mists and dusts are anticipated. In both cases non-mechanical non-organic pumps are favored. Kyoto Fusioneering is developing tritium compatible vacuum pumps for various types of plasma exhaust for fusion reactors. Metal Diffusion Pumps (MDPs) using alkali metals as the primary fuel cycle pump for DT burning plasma devices, both for near term experiments and Fusion Power Plants (FPP). The vacuum pumps for DT machines should be capable of continuously exhausting hydrogen isotope gases including tritium, at a flow rate of the order of 100 Pam3/s from the chamber at around 1 Pa range and compresses and transfer for fuel cycle with minimal tritium inventory. Following MDPs, Proton Conductor Pump (PCP) is developed by the authors to separates hydrogen isotopes for purification and recycling with small delay and inventory. Experiments are performed for MDPs driven with Na vapor, hydrogen, deuterium, helium, and their mixtures to evaluate the pumping speed and compression characteristics. For PCP, electrochemical transfer of hydrogen isotopes from gas mixture is measured. Basic pumping function of both MDPs and PCP was measured, and the results suggest possible application and its parameter space of operation. In this presentation, we report the pumping characteristics of MDPs and PCP, feasible, and realistic design of evacuation system for plasma exhaust. We also report the next step pump train test of MDPs and PCP for relevant exhaust gas environment and magnetic field in a closed cycle demonstration campaign.

Keywords

PCP, DIR, Toritium, Roughing pump, Metal diffuser pump, Fuel cycle.

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PS4-44 ABSTRACT-ce22
C. Fuel Cycle and Tritium Processing

Ammonia Palladium Membrane Reactor Development

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Ammonia is likely to be a common impurity in the fusion fuel cycle as tritiated ammonia readily forms when tritium and nitrogen are in contact. While sorbents can effectively trap ammonia, the adsorbed ammonia poses significant accountability challenges, as tritiated ammonia is traditionally difficult to remove from sorbents. Therefore, it is preferred to catalytically decompose the ammonia before it can adsorb onto the sorbent beds. The historical route to decompose the ammonia into tritium and nitrogen is through a fixed bed reactor, but this is limited to the equilibrium concentrations for the reactor's temperature and pressure. By simultaneously removing one of the decomposition products, the equilibrium can be shifted to near 100% conversion. The use of palladium silver membranes works to remove the hydrogen isotopes, enabling extremely high conversions. However, standard catalysts are only active at temperatures much higher than palladium silver membranes can withstand. Savannah River National Laboratory, in collaboration with the University of South Carolina, has recently developed new catalysts that are highly active at palladium silver membrane operating temperatures. In order to test the effectiveness of the catalyst inside palladium silver tubes, a palladium membrane reactor was needed in order to couple the ammonia decomposition with hydrogen removal. This required developing a method to braze palladium silver tubes to stainless steel plenums, as the palladium silver tubes cannot be welded due to the extremely thin tube walls. A bench-scale palladium membrane reactor was fabricated and used to evaluate different catalysts to develop an optimal catalyst metal composition and catalyst support for ammonia decomposition at the palladium silver operation temperatures. The brazing development methodology and PMR catalyst development will be presented.

Keywords

Tritium, ammonia, catalyst, diffuser, Palladium.

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PS4-46 ABSTRACT-1c1e
C. Fuel Cycle and Tritium Processing

Gas Raman: Detecting Gases within the Fuel Cycle

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IS-Instruments (ISI) is a micro-SME based in Kent, England. ISI specialises in the development of remote sensing instrumentation for industrial applications, specifically Raman. Over the past four years, ISI, in collaboration with the Optoelectronics Research Centre (ORC) and Jacobs Clean Energy, have been developing a compact and easy-to-use gas-phase Raman spectrometer. This instrument can successfully observe numerous gaseous species using only one channel, including hydrogen and it's isotopologues, an essential requirement for the fusion industry.

As the molecules in a gas sample are more diffuse, a Raman spectrum can be extremely difficult to obtain. To achieve a result, either a high-power laser or an increased pathlength are required. ISI's instrument combines their high-throughput spectrometer technology with a Hollow Core-Microstructure Optical Fibre (HC-MOF) designed by the ORC. The benefits of this include not only an increased pathlength, but also a small sample volume and a closed system capable of online/offline measurements. By increasing the pathlength, the number of excitation photons from analyte molecule interactions also increases, resulting in improved sensitivity. This result can be achieved using microliters of sample distributed within the void of the HC-MOF, beneficial for hazardous or rare gases.

During phase one of the Fusion Industries Programme, (SBRI funded) ISI, ORC and Jacobs continued to improve the functionality of the spectrometer. Under the guidance of UKAEA, the team confirmed the instrument's ability to accurately detect and quantify H_2, D_2 and HD with notable selectivity [Appendix 1]. Strong concentration linearity was observed for all three compounds, this data, along with literature values was used to extrapolate for tritium. Throughout 2023 the team aim to continue this development, and test the instrument's ability to analyse for tritium.

Additionally, ISI have been working on a multichannel gas-phase Raman spectrometer, to analyse samples from separate locations simultaneously using only one detector.

Keywords

Spectroscopy, Measurement, tritium, gas analysis, Raman spectroscopy, research..

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PS4-47 ABSTRACT-dc6e
C. Fuel Cycle and Tritium Processing

Assessment of Metal Foil Pump Configurations for EU-DEMO

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It is a challenging but key task to reduce the tritium inventory in EU-DEMO to levels that are acceptable for a nuclear regulator. As solution to this issue, a smart fuel cycle architecture is proposed based on the concept of Direct Internal Recycling (DIR), in which the Metal Foil Pump (MFP) will play an important role to separate the unburnt hydrogen isotopes coming from the divertor by exploiting the super-permeation phenomenon. The MFP is foreseen to be installed inside the DEMO lower port, which is of very complex geometry and will be used as port for remote maintenance activities. Because the MFP is part of the pumping channel not only in burn phase but also in dwell, additional requirements on the performance of the whole system in both operational phases must be fulfilled.

In this study, we will present the assessment of three different MFP configurations (parallel long tubes, sandwich and halo) by means of the Test Particle Monte Carle (TPMC) simulation. For the first time, the whole integrated vacuum system has been modelled with the TPMC code ProVac3D. Because the code has been parallelized with high efficiency, we can systematically simulate this very complex system in super computer Marconi-Fusion, and the system conductance for helium molecules, the pumping speed and the separation factor for deuterium molecules under different physical and geometric parameters are finally obtained. These results are essential for the development of a suitable MFP design in the vacuum pumping train of EU-DEMO.

Keywords

Direct Internal Recycling, Metal Foil Pump, Super Permeation, Monte Carlo Simulation.

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PS4-48 ABSTRACT-df34
C. Fuel Cycle and Tritium Processing

Concept of the HCPB TER using non-evaporable getters for tritium recovery

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The reference technology for BEMO Helium Cooled Pebble Bed breeding blanket for the tritium extraction is based on tritium release in a He purge gas followed by a 2-stage tritium recovery process from the purge gas in the Tritium Extraction and Recovery system. The recovery process is based on adsorption of Q_2O on reactive molecular sieve beds, while Q_2 is trapped using cryogenic molecular sieve beds. Several risks have been identified with the reference technology for the system, like low reliability of BB system due to large pressure difference between the cooling system and the purge gas in TER, or large amount of LN2 consumption during the operation of the cryogenic beds operation. The development of the TER system architecture proposes replacement of the cryogenic beds with non-evaporable getter beds and the operation of the system at the same pressure as the BB cooling system, 8 MPa. By this development, the risks identified with the reference technology are mitigated or excluded.

Keywords

NEG technology, tritium recovery from helium.

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PS4-49 ABSTRACT-e9a9
C. Fuel Cycle and Tritium Processing

The interaction between some candidate blanket materials with tritium and associated detritiation technologies for a magnetic confined fusion reactor

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Several candidate blanket materials are chosen for a magnetic confined fusion reactor like CFETR. The permeation and acumination of deuterium and tritium in these materials were investigated experimentally. Results showed that the plasma state of hydrogen isotopes lead to a high permeation and inventory in tungsten and the irradiation generated defects contribute even more. The hydrogen isotopes in and on the plasma facing materials can be measured on site with the laser induced breakdown spectrometer (LIBS) or laser induced desorption spectrum (LIDS) approaches with an acceptable precision. The tritium permeation rate in the plasma facing components during a full power day was assessed to be around 1 gram and the saturated tritium inventory was 213 grams after 10^5 times of the 1000s-span shots. More than 90% of tritium in the tungsten block and tungsten dust can be removed through heat desorption at 600~800℃ while hydrogen isotopic exchange contributes little to the efficiency. Tritium generated in beryllium and beryllium titanium alloy by neutron spallation need to be removed at high temperatures more than 1000 ℃. Tritium in the RAFM steels can be easily removed at 400 ℃ due to the 1~2 magnitudes of higher diffusivities than the austenitic stainless steels. RAFM steels posses a poor tritium compatibility yet under a normal running condition, the mechanical degradation is acceptable. Tritium produced in the lithium ceramic pebbles can be extracted and recovered with an efficiency higher than 99% through the He-0.1%H_2 gas purging and subsequent He-H_2 separation by Pd-Y membranes. Tritium in the liquid Pb-17Li alloy can be extracted and measured with the permeation membranes made of Niobium foil also with a high recovery efficiency of 90%. Pure tritium can be obtained with a hydrogen isotopic seperation by thermal cycling adsorption process (TCAP) and the approach was developed.

Keywords

Fusion reactor, blanket, tritium retention, tritium recovery.

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PS4-51 ABSTRACT-f9c7
C. Fuel Cycle and Tritium Processing

Organic vacuum pump fluids for the vacuum pumping of fusion power plants

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Vacuum pumps are the heart of the fusion fuel cycle, but most proposed pumping technologies are not capable handling the required flow rates and vacuum pressures. Oil-containing vacuum pumps can readily meet the flow requirements, but vacuum fluids will degrade in fusion-relevant environments due to contact with tritium and exposure to high energy radiation. Here, we describe a methodology to determine suitable vacuum fluids candidates, purify these candidates post-oxidation, post-exposure to deuterium, and post-exposure to gamma radiation (<7.5 MGy), to simulate a process in which vacuum fluids can be recovered and regenerated during the fusion fuel cycle. A series of oils, including a highly purified mineral oil (MO), phenyl silicone oil (PSO), and a polyphenyl ether (PPE), are shown to be suitable candidates for vacuum pumping. Additionally, we describe a simple purification methodology to remove oxidized functionalities and the associated isotopologues induced by contact with deuterium from the candidate vacuum fluids This purification methodology can also be applied to radiological damage with moderate effect. Finally, we demonstrate that the sorbents can be regenerated through electromagnetic microwave digestion

Keywords

Vacuum fluid, vacuum oil, purification, viscosity, oxidation.

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PS4-52 ABSTRACT-faec
C. Fuel Cycle and Tritium Processing

Design Study on Fuel Buffer System for Continuous Processing of the Tritium Plant Considering Burn and Dwell Operation Scenarios

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The most important purpose of the Tritium Plant in the fusion fuel cycle is to stably supply tritium and deuterium of a constant composition. To achieve this, it is desirable for the Tritium Plant to be designed as a continuous process. In this study, we aim to propose a fuel cycle concept and a design for the interface system, the Fuel Buffer System (FBS), between the Tritium Plant and Tokamak with continuous operation of the Tritium Plant. The buffer vessel is considered as a key interface device in the FBS. We evaluate the tritium inventory and the design impact of the FBS in a tokamak burn-and-dwell operating environment. By applying known ITER operating scenarios, we perform sensitivity analysis on the tritium storage limit, the operating pressure of the buffer vessel, and fuel cycle operation time. Based on the results, we suggest the design strategy for fusion fuel cycle in the K-DEMO.

Keywords

Tritium plant, fuel buffering, conceptual design, continuous processing.

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PS4-53 ABSTRACT-fcce
C. Fuel Cycle and Tritium Processing

Alkali metal vapor diffusion pump for fusion reactor evacuation

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Vacuum pump for the fusion system is quite important regardless of the confinement style, which must have high efficiency and resistance against tritium contamination and radiation. Additionally, it should work in strong magnetic field, without the release of high-Z element depending on the fusion system.

The diffusion pump which usually utilizes oil or mercury vapor to transfer the momentum, can be a candidate when the vapor source is replaced to the light alkali metal (such as Li or Na) which is free from radiation degradation. In this work, we tried to modify a commercial diffusion pump and the pumping speed is investigated. As the modification, the oil was replaced by sodium of the same volume, a voltage controller was added to the heater, and a cooling fan for the radiator-fin was replaced to a speed controllable one. Thermocouples to monitor the surface temperature of the pump were also attached.

Evacuation by Na vapor was obviously observed when the boiler temperature was close to or above 950K. The heating of the boiler and the cooling of the radiator-fin showed a positive effect on the pumping speed. So far, evacuation against He gas achieved 30L/s with Na vapor which is a bit slower than the one with oil vapor (~120 L/s) in the as received pumping system.

Considering these results and our past works on the purification of liquid metal, liquid alkali metal will be applicable for the vapor diffusion pump for fusion system.

Keywords

Vacuum pump, tritium pump, diffusion pump, alkali metal.

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PS4-56 ABSTRACT-a347
D. Material Engineering for FNT

Research on the Low Cycle Fatigue Behaviors of Gradient Nano-Structured CLAM Steel

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Reduced activation ferritic/martensitic (RAFM) steels have been selected as candidate structural materials for future advanced nuclear power systems. Pulsed plasma operation mode is used in most of the design schemes of fusion reactors, which puts forward higher requirements on the fatigue performance of structural materials. To improve the fatigue performance of China low activation martensitic (CLAM) steel, the gradient nano-structured (GNS)-CLAM steel was developed.

A GNS surface layer with a thickness of ~85μm was prepared on CLAM steel utilizing surface mechanical rolling treatment (SMRT). The mean grain size was approximately 43 nm at the topmost surface and increased gradually with depth. The stress-controlled tension-compression fatigue experiments showed that the fatigue life at room temperature enhanced more than 6 times in the SMRT samples compared to the corresponding base metal counterparts. The relationship between the applied stress amplitude and the fatigue lifetime, and the fracture morphology showed that the surface strengthening and strain delocalization were caused by GNS layer, which suppressed the surface crack initiation process. Hence, the fatigue properties of CLAM steels improved. The deformation compatibility in GNS layer and coarse-grained boundaries leading to more dislocation interactions and accumulation during the cyclic process, also plays a crucial role in enhancing the fatigue properties of CLAM steel. In addition, the GNS-CLAM steel showed high stability of microstructures at a high temperature of 550℃/60 min due to the pinning effect of $M_{23}C_6$ and MX precipitates, which contributed to an improvement of fatigue life at 550℃/260 MPa by 5.5 times compared to the original one. Moreover, the high-temperature fatigue resistance enhancement in the SMRT samples was supposed to be the synergistic contribution of compressive residual stress arising, grain nanocrystallization/refinement, and the strain delocalization introduced in the gradient nanostructure.

Keywords

CLAM; Gradient Nano-Structured; Low Cycle Fatigue.

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PS4-57 ABSTRACT-a3ce
D. Material Engineering for FNT

Weld Technologies for Group V materials (V, Nb)

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Current tritium extraction systems considered within the fusion field rely on the use of Group V materials due to their high permeability to tritium and good compatibility with operating temperatures. However, the assembly of these materials with the supporting structure (typically stainless steel) is a difficulty. Different welding technologies have been used for assuring the joint and a good leak tightness.

Niobium is a refractory material with a fusion point of 2477ºC. First, it has been welded to SS304L sleeves by electron beam welding (EBW) that act like a transition part to the rest of the component. Sometimes, this type of weld makes microcracks inside the weld that are not compatible with vacuum tightness. If the welding is repeated in order to seal the microcracks the joint becomes very brittle. Due to this, brazing technology has been used trying to get a good leak-tightness weld.

In a similar way, vanadium has a high melting point (1910ºC). The attachment of vanadium and steel has been performed firstly by EBW of small windows with martensitic stainless steel 410. The welding of large systems (1 m app.) produces some leaks which are impossible to be repaired. With small dimensions, EBW and brazing were made with satisfactory results in both technologies. At higher scale, with brazing, there is not leak tightness due the deformation of the window during the thermal cycle. Results obtained with the EBW process were positive with a strong and leak-tightness weld.

In both cases, ESS Bilbao has performed tests using a procedure mixing EBW and brazing technologies having some good results for weld strength and leak tightness. They are promising but more tests are needed repeating the same procedure to assure the repeatability of the process.

Keywords

Weld Technologies, Electron Beam Welding, Brazing.

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PS4-58 ABSTRACT-aad9
D. Material Engineering for FNT

The effects of heat treatment process on mechanical properties of Bi-2212 Round Wires

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Magnetic confinement is considered the most promising way to achieve controlled nuclear fusion. The magnet system in a fusion reactor consists of toroidal field (TF) coils, poloidal field (PF) coils, central solenoid (CS) coils and correction coils(CCs). Low-temperature superconducting materials represented by NbTi and Nb_3Sn will not be able to meet the requirements of ultra-large currents and ultra-strong magnetic fields in future fusion reactor magnet systems. (Bi-2212), with the excellent critical current density () and the ultru-high upper critical magnetic field (), has become one of the most potential candidates among the high-temperature superconducting (HTS) materials for the manufacture of the next generation of superconducting magnets. As a kind of brittle phase, Bi-2212 is extremely sensitive to strain. Therefore, the mechanical property is a key factor for its application. In order to investigate the effect of different heat treatment processes on mechanical properties of Bi-2212 round wires (RWs), axial tensile measurements on Bi-2212 RWs were performed at room temperature, 77K and 4.2K with different heat treatments, including with overpressure heat treatment (OPHT), pre-overpressure heat treatment (pre-OPHT) and pre-overpressure heat treatment followed by overpressure (pre-OP+OP) heat treatment. It was found that the wires with pre-overpressure heat treatment had the highest mechanical strength. Besides, the wires that were overpressure heat-treated had paralleled performance compared to the ones with pre-OP+OP heat treatment. This provides a new theoretical basis and technical direction for the research and manufacture of Bi-2212 Cable-in-Conduit Conductors (CICCs) used for nuclear fusion.

Keywords

Nuclear fusion,Bi-2212,heat treatment,mechanical properties,CICCs.

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PS4-59 ABSTRACT-add9
D. Material Engineering for FNT

Effects of Size and Static Temperature Exposure on the Shear Strength of Tungsten Bonded CFC Blocks for Divertor in JT-60SA

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A satellite tokamak, JT-60SA, is being built up at Naka in Japan. JT-60SA research plans metal wall experiment to contribute the research of ITER and developments for DEMO. JT-60SA employs carbon fiber strengthened carbon (CFC) composites divertors. One of ideas to form the metal wall for the experiment is to use clad divertor blocks which a tungsten plate set on CFC for the convenience of operation at JT-60SA. Previous researches revealed that the small size tungsten and CFC clad plates were able to be fabricated using sinter bonding methods. The full scale of a divertor block is 33mm x 31mm, therefore, it is necessary to evaluate the size effects and uniformity of the bonding. A full scaled W/CFC block were cut and evaluate strength by a four-point bending test to reveal the correlation the strength and the position on the cross-section to evaluate the uniformity of the joint. Thermal stability of the various sintered joint interface also was evaluated. The joint specimens are statistically heat treated at high temperature for over 1000 h in vacuum and investigated by a cantilever share test. Normal SiC powders are used for the sinter bonding, and the other candidate materials are also tested in this investigation.

Keywords

JT-60SA, Divertor, tungsten, CFC, joining method.

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PS4-60 ABSTRACT-bca6
D. Material Engineering for FNT

Radiation safety study for the high-duty operation of the LIPAc RFQ in Rokkasho

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The International Fusion Materials Irradiation Facility (IFMIF) project aiming at material tests for a future fusion power plant is now in its Engineering Validation and Engineering Design Activities (EVEDA) phase under the Broader Approach Agreement between Japan and EU. As part of the activities the construction of the Linear IFMIF Prototype Accelerator (LIPAc) is in progress at Rokkasho, Japan. For the validation of the accelerator design to provide 40 MeV and 125 mA continuous wave (CW) deuteron beam on the liquid lithium target, the commissioning of LIPAc is conducted in a stepwise approach to achieve the 125-mA, CW beam acceleration through the low energy section of the IFMIF accelerator up to 9 MeV. After having achieved an important project milestone to accelerate the world highest current deuteron beam of 125 mA at 5 MeV with 0.1% duty cycle (1 ms pulse) through the Radio Frequency Quadrupole linac (RFQ) in July 2019, the next objective is set to validate the acceleration of the deuteron beam with the RFQ at higher duty cycle targeting CW. To this end, the MEBT extension line with quadrupole magnets and steering magnets was newly developed and installed instead of the Superconducting RF accelerator. The High Energy Beam Transport (HEBT) section and the high-power beam dump that can dissipate 125-mA, CW deuteron beam were also installed to complete the beam line for the high duty operation. The present paper describes the details of the study devoted for the radiation safety analysis for obtaining the license from Nuclear Regulation Authority of Japan and to operate the RFQ at high duty cycles up to CW. In particular, the details of the radiation source definition for neutrons and photons, the d-D reaction contribution analysis and the shielding analysis will be discussed.

Keywords

LIPAc, IFMIF, radiation, neutron, shielding.

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PS4-61 ABSTRACT.-31a4
D. Material Engineering for FNT

Comparative study of preparation methods and properties of lead-lithium eutectic for fusion technology

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A key component in the development and operation of future commercial fusion reactors is the breeding blanket (BB). This component is used as source of tritium, neutron irradiation protector (shielding function), and coolant. Lead-lithium alloy (PbLi) has been selected as the most appropriate candidate for liquid BB since breeder and neutron multiplier are combined in the same compound. A critical aspect in fusion technology is that PbLi will be produced according to nuclear materials QA (Quality Assurance) standards, because any variation in the physical and chemical properties of the alloy will have severe consequences in its regenerative function inside the blanket. For this reason, it is necessary to perform an exhaustive control of the eutectic composition and the impurity content of the PbLi alloy. In addition, an evaluation of thermal, chemical (Li segregation) and structural homogeneity should be accomplished.

In this work, multiple samples collected from PbLi ingots prepared by different procedures have been thermally (melting point), chemically (lithium content and metallic impurities) and microstructurally (crystalline phases) characterized by different techniques such as TG-DTA (Thermogravimetry - Differential Thermal Analysis), FAES (Flame Atomic Emission Spectrophotometry), XRF (X-ray Fluorescence), ICP-OES (Inductively Coupled Plasma Optical Emission Spectroscopy), ICP-MS (Inductively Coupled Plasma Mass Spectrometry) and XRD (X-Ray diffraction). The obtained results are used to determine the influence of the preparation method in the physical-chemical properties of PbLi and in their homogeneity.

Keywords

Breeding blanket, lead-lithium eutectic, nuclear material quality assurance.

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PS4-62 ABSTRACT-c1b9
D. Material Engineering for FNT

The Impact of Nitrogen Pre-Implantation, the Crystal Orientation and the Recovery on the helium blisters growth mechanism in copper

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During the fusion nuclear reaction, the reactor first wall is damaged by energetic helium particles. The helium accumulates, displacing the matter atoms, forming bubbles in the bulk and blisters on the surfaces.

The intent of this research is to understand the influence of nitrogen pre-implantation, crystal orientation, and the thermomechanical history of copper on helium implantation damage.

The samples were implanted with nitrogen and helium at the Helmholtz-Zentrum Dresden-Rossendorf facility. Before and after implantation/s, the samples were analyzed using SEM and STEM. The crystal lattice orientation at the sample's surfaces, was obtained by the electron backscatter diffraction method.

The results showed that the as received copper developed larger blisters than compared with recovered one. Moreover, pre-implantation recovering inhibited the blisters formation on (100) oriented grains but induced the blisters formation on (111) oriented grains.

In addition, copper samples pre-implanted with high doses of nitrogen show superficial blistering, regardless of the post helium implantation. Furthermore, it was found that nitrogen implantation induced the copper nitrate formation. Besides, the uniform distribution of these nitrates observed before helium implantation, becomes bimodal after helium implantation.

We concluded that: 1) The crystallographic orientation and the thermomechanical history of copper influence the number and size of surface blisters induced by helium implantation. 2) Under the specific conditions and parameters examined in this investigation, preimplantation of nitrogen prior to helium implantation does not prevent bubbles and blister formation on copper

Keywords

Nitrogen, Implantation, Helium, Blisters , Copper.

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PS4-63 ABSTRACT-c2c9
D. Material Engineering for FNT

Influence of Thermal Aging on the Deuterium Retention and Permeation Behavior in the RAFM Steel

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Reduced Activation Ferritic/Martensitic (RAFM) steels have been chosen as the primary candidate structural material for the future fusion reactors due to their better thermo-physical, thermo-mechanical properties and post-irradiation performance compared with austenitic stainless steels. As one of the RAFM steels, China low activation martensitic (CLAM) steel has been chosen as the primary structural material of the FDS series PbLi blankets for fusion reactors, CN helium cooled ceramic breeder (HCCB) test blanket module (TBM) for ITER and breeder blanket of China fusion engineering test reactor (CFETR). Hydrogen isotopes, generated from neutron transmutation, tritium breeders, and D-T plasma, will diffuse into material and lead to an abominable embrittlement effect. The permeation of hydrogen isotopes through structural walls even affect isotopes mixture balance in plasma and cause the radioactive contamination. Permeation and retention behavior of hydrogen isotopes, deuterium and tritium in CLAM steel is vital for the economy and safety of fusion reactors.

As one of candidate structural materials for future fusion reactor, China low activation martensitic (CLAM) steel would degenerate gradually during service at high temperature. However, the deuterium retention and permeation behavior in CLAM steel after thermal aging is not well understood. In this study, the CLAM steels were subjected to exposure at 923 K for 8000 h. Subsequent gas driven permeation (GDP) and thermal desorption spectroscopy (TDS) experiments were carried out to investigate the deuterium permeation and retention behavior, respectively. The results indicated that the deuterium retention of aged steel decreased after thermal aging for 8000 h. By contrast, the deuterium permeability of aged CLAM steel increased. The difference of the retention and permeation behaviors of deuterium between various aged CLAM steels are attributed to the change of grain/lath boundaries.

Keywords

Deuterium; Retention and Permeation; RAFM Steel.

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PS4-64 ABSTRACT-c4ac
D. Material Engineering for FNT

Development of Advanced Ternary Beryllium Intermetallic Compounds

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Beryllides, beryllium intermetallic compounds, have recently emerged as promising candidates for advanced neutron multipliers in the DEMO reactor due to their high stability and low reactivity with water vapor. $Be_{12}Ti$ and $Be_{12}V$ are representative beryllides that have been studied recently. In addition, beryllides have been considered for loading as a neutron multiplier material in the form of pebbles. However, nowadays, beryllides are expected to be used as block-type advanced neutron multipliers, because of their lower swelling characteristics for the advanced blanket for a JA DEMO reactor.

To design alternative advanced blankets, it is essential to establish a mechanical property database for block-type beryllides. Additionally, it is also important to develop beryllide materials with higher mechanical strength and thermal stability than the current binary beryllides. Therefore, the objective of this study is to fabricate a new ternary beryllide and evaluate its mechanical and thermal properties.

The beryllide blocks used in this study are made from ternary mixed powders by plasma sintering. Then, microstructure observations were carried out with using Scanning electron microscope (SEM) and Electron Probe Micro Analyzer (EPMA). To evaluate the mechanical properties, Micro Vickers hardness test and indentation fracture tests were carried out. Moreover, to investigate the thermal stability, thermal expansion and thermal conductivity were examined.

A detailed results regarding mechanical and thermal properties of newly developed advanced beryllides will be reported.

Keywords

Beryllide, mechanical properties, thermal properties.

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PS4-65 ABSTRACT-dbe7
D. Material Engineering for FNT

A Material Production and Qualification Plan for lead-lithium eutectic nuclear grades

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Basis for a complete Material Production and Qualification Plan (MPQP) for nuclear grades of lead-lithium eutectics is discussed. Production Plan includes: (1) the set of raw materials (Pb, Li) and LLE products handling and qualification established procedures; (2) the possible LLE alloying routes according to technical feasibility and industrial reliability; (3) the possible alloying routes for experimental demonstration at laboratory to be scale-up by design and demonstrated at required industrial scale. The provided Qualification Plan provides: (1) the specification of the material characteristics both constitutive (e.g. Li title, short-scale homogenization) and compositional (e.g.: allowable impurity limits); (2) the material database specification inputs and allowable uncertainties according to the material uses and design functionalities; (3) the material certification procedures, as the set of suitable experimental procedures to certify database properties. The two main goals are: (1) - the standardization by the generation and establishment of Norms and Standards (e.g. as UNE or ISO) in the production and qualification of lots of LLE(6) nuclear material grades; (2) - The establishment and management of a specific network of Services for LLE(6) characterization and standardized material qualification.

Keywords

Material Production and Qualification Plan (MPQP), Lead-lithium eutectics (LLE), Alloying routes, Material characteristics, Standardization and certification.

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PS4-66 ABSTRACT-ea83
D. Material Engineering for FNT

Excellent mechanical property of W-Y2O3 alloy processed by Kocks Rolling

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Tungsten materials have been attracting growing interest as a promising candidate for plasma facing materials (PFMs) based on its high melting point, high temperature strength, and good thermal conductivity. But the inherent disadvantages of tungsten materials, including poor machinability, low ductility, and high ductile-brittle transition temperature (DBTT), cannot be ignored for fusion reactor application. Some strategies studied in recent years have focused on composition and thermal processing to resist tungsten defects.

In this work, the mechanical properties and microstructure of W-Y_2O_3 alloy prepared by hydrogen sintering followed by Kocks rolling and annealing treatment were investigated. The W-1, 2, 3vol% Y_2O_3 mixed powders were prepared using a liquid-liquid (L-L) mixing process. The mixed powders were pressed into a cylinder by cold isostatic pressing, and then sintered in a hydrogen atmosphere at 2100 ℃. The sintered ingot was subjected to Kocks rolling mill at well controlled temperatures, by which slender rods of 12 mm in diameter (from the original diameter of 45 mm) were made. Density, hardness, and tensile strength at different temperature of W-Y_2O_3 alloy were tested respectively.The microstructure were analysed by scanning electron microscope and transmission electron microscopy. The results indicate the rolled W-Y_2O_3 alloy have excellent mechanical properties. At room temperature, The tensile strength of the three materials is up to 1.7 GPa and the strain is more than 10%, among which the strain of W-1%Y_2O_3 is up to 15%. This result is higher than most reported tungsten alloys. At 200℃, the tensile strength of W-1%Y_2O_3 is still as high as 1.2GPa and its strain is as high as 40%. The three materials also have high recrystallization temperatures, among which the recrystallization temperature of W-3%Y_2O_3 alloy is as high as 1800℃. The microstructure analysis shows that the rolled tungsten alloy has special texture.

Keywords

Tungsten, Plasma facing materials, Kocks rolling, mechanical property.

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PS4-67 ABSTRACT-ec41
D. Material Engineering for FNT

Effect of Ion Implantation on microstructure and critical current for YBCO superconductor

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Three kinds of ions (Ni, Fe, and Ag) were utilized for implantation of YBCO superconductor specimens for trying to increase the flux pining centers and thus to improve the current performance. The results show that, after Ag ion implantation, current cannot be detected, this means that Ag ion cannot increase current performance of YBCO superconductor. While after implantation with Fe and Ni ions, the critical current changes with the implantation doses. For the specimen with Ni ion implantation, the critical current decreased by 50-60% when the dose is $1*10^{13-14}$ ions/cm^2, and then increased when the dose $1*10^{15-17}$ ions/cm^2 but no longer reach to the same level as that before implantation. For the specimen with Fe ion implantation, the critical current increased significantly at the dose of $1*10^{13}$ ions/cm^2, by 40-60%. While after that as the dose increases, the critical current deceased by 30-60% first at $1*10^{14-15}$ ions/cm^2, and then increased slightly at $1*10^{16-17}$ ions/cm^2, but also no longer reach to the same level as that before implantation.

Microstructure analysis by transmission electron microscopy (TEM) show that, nanoscale crystallization behavior seemingly was found in the YBCO layer for the specimen with Ag ion implantation. This may be the reason why critical current can't be easily detected after Ag ion implantation. For the specimens with Ni and Fe ion implantation, latent track, stacking faults, and nano-sized particles were obviously observed in the YBCO layer. The detailed microstructure mechanism will be further analyzed by TEM for the specimens especially that with Fe ion implantation at dose of $1*10^{13}$ ions/cm^2, which seems effective for the critical current enhancement of the YBCO superconductor.

Keywords

Ion implantation, YBCO superconductor, nano-sized particles, crystallization, critical current.

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PS4-68 ABSTRACT-f64d
D. Material Engineering for FNT

Sputter yield of tungsten with strongly different microstructure

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With tungsten being the material of choice for the first wall in future fusion reactors, its sputtering behavior needs to be precisely understood to accurately assess the component lifetime. Therefore, numerous numerical as well as experimental approaches are carried out in the fusion community. At IPP Garching, the SIESTA experiment [1] with its possibility to create a mass and energy-separated ion beam is used for various controlled erosion experiments. Also, the numerical code SDTrimSP, a Monte-Carlo code for calculating collision phenomena, was developed and is refined continuously. Recently its capabilities were expanded to three dimensions [2].

For this study, a tungsten sample consisting of drawn tungsten wire embedded in a chemical vapor deposited (CVD) tungsten matrix [3] was polished to a mirror finish. The surface was subjected to different mass-selected ion beams. Mean sputtering rates derived from weight-loss measurements match the expected literature values reasonably. However, the spatially resolved erosion depth measured by confocal laser microscopy revealed a consistent difference in the sputtering yield between the tungsten matrix with large grains, and the severely deformed and fine-grained microstructure of the tungsten wire.

To identify the underlying reason for the observed differences in erosion behavior, careful pre- and post-experimental investigations were carried out. The influence of grain orientation [4] on the sputtering behavior is considered by evaluating electron backscatter diffraction measurements. The contribution of surface roughness and its evolution after exposure to the ion beams at SIESTA is calculated using atomic force microscopy results as input for 2-dimensional SDTrimSP simulations. Additionally, differences in the microstructure of the composite constituents are considered as an influencing factor on the sputtering yield. The contribution identifies the proportionate contribution of these three effects on the sputtering rates.

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[2] http://dx.doi.org/10.1016/j.nme.2020.100749
[3] https://doi.org/10.1016/j.nme.2016.03.005
[4] https://doi.org/10.1103/PhysRevLett.125.225502

Keywords

Tungsten, sputtering, erosión.

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PS4-68b ABSTRACT-fae3
D. Material Engineering for FNT

Calculation of production of Mo-99 in the IFMIF-DONES facility

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Molybdenum-99 (^{99}Mo) is the precursor of Technetium-99m (^{99}Tc), which is the most demanded radiopharmaceutical in nuclear medicine. ^{99}Tc is used to diagnose many types of cancers. Due to supply shortages, it is currently urgent to find alternatives to conventional production in nuclear fission reactors.

The main function of the International Fusion Irradiation Facility-DEMO Oriented NEutron Source (IFMIF-DONES) will be to irradiate materials with a flux of high energy and intensity neutrons. These neutronic features potentially enable the use of alternative nuclear reactions for ^{99}Mo production, such as ^{98}Mo(n, γ)^{99}Mo and ^{100}Mo(n, 2n)^{99}Mo, which have a high cross section with the IFMIF-DONES spectrum.

In this work we used transport simulation codes and nuclear inventory to calculate the specific activity of ^{99}Mo under varying conditions of irradiation time, target position, sample purity and isotopic composition. A production distribution map was prepared on an area of 50x200 cm^2, centered in the area of maximum exposure. Based on these calculations, we identified the position right behind the high test flux module as the most promising for ^{99}Mo production. We propose two possible solutions for the extraction of ^{99}Mo every 6 days, which is the optimal extraction frequency for the practical use of the isotope in nuclear medicine. We estimate that one of the two solutions would enable sufficient production to meet the needs of a country such as Spain.

Keywords

Molybdenum-99 , Technetium-99m, IFMIF-DONES, production.

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PS4-69 ABSTRACT-b0c8
E. Vacuum Vessel and Ex-vessel Systems

Final design of ex-vessel components for the Wide Angle Viewing System diagnostic for ITER Equatorial Port 12

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The ITER visible and infrared Wide Angle Viewing System (WAVS) provides surface temperature measurements of plasma facing components by IR thermography being one of its main roles to protect them from damage. It also images the plasma emission in the visible spectral range. The diagnostic that comprises 15 views distributed in four equatorial ports (3, 9, 12 and 17), will contribute to Machine protection, Basic control and Physics analysis. The final design of the system in the equatorial port 12 - including three lines of sight - has been developed within a contract granted by F4E to the Consortium formed by CEA, Ciemat, INTA and Bertin Technologies. The Final Design Review of EP12 held in the first quarter of 2023, is expected to be completed in the coming months. After approval of the final design, the manufacturing phase will be launched.

The final opto-mechanical design for the ex-vessel components in EP12 will be presented in the paper. The main function of the ex-vessel optics is to relay the exit image of the port plug at the vacuum window, up to the detectors placed at the back-end of the optical change in the Port Cell, while keeping the required performance. The image relay through the Interspace, Bioshield and Port Cell covers around 10 m in length before reaching the cameras. The ex-vessel optics is arranged in several optical modules, assembled on the EP12 support structures, i.e. Interspace Support Structure (ISS) and Port Cell Support Structure (PCSS).

A detailed description of the opto-mechanical design that supports the optical modules, including the alignment capabilities, will be presented in the paper. Moreover, the assembly sequence and alignment strategy planned for the ex-vessel subsystems, taking into account all constraints imposed on the site, will be also included in the paper.

Keywords

ITER, WAVS, diagnostics, infrared, visible, ex-vessel optics.

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PS4-70 ABSTRACT-b659
E. Vacuum Vessel and Ex-vessel Systems

Cooling design and analysis of the EU DEMO EC Launcher port plug modules

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The EU Demonstration Power Plant (DEMO) will be equipped according to its present baseline design with six EC (Electron Cyclotron) launchers of maximum 36 MW microwave power each, aiming at plasma heating and to counteract plasma instabilities during operation. The launcher antennas will be installed into six equatorial ports of the DEMO vacuum vessel. All in-vessel microwave components of an EC antenna, comprising several sets of mirrors and waveguides are mounted either into so-called port plug modules or in between those and the closure plate, which forms the First Containment System (FCS) at the rear side of the port extension. The port plug modules are basically massive blocks with dedicated cut-outs for mirror integration and beam propagation. They fit the ports as integrated built-in components, which guarantee optimum performance and simplify assembly and maintenance of the EC system. During operation the EC port plug modules and their internals will be exposed to nuclear heating from neutrons and photons and thermal radiation from the plasma. Also stray radiation and power losses from microwave components can create local heating of the port plug modules. This is why they must be equipped with a powerful heat removal system. Beside optimum dissipation of the total heat applied to the launcher components, also good flow characteristics, adequate distribution of coolant in parallel branches and steady temperature gradients between particular cooling channels must be ensured by design. This paper outlines the general layout of the cooling circuits, and particular design features of the cooling channels. Relevant results of thermo-hydraulic analyses for different operation scenarios are presented as well as integration, manufacturing and maintenance aspects.

Keywords

DEMO, ECRH, Heating and Current Drive, Plasma stabilization, Cooling design.

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PS4-71 ABSTRACT-c23e
E. Vacuum Vessel and Ex-vessel Systems

Plasma disruptions in the design of DTT and strategies for their mitigation

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The DTT (Divertor Tokamak Test) facility is a medium size nuclear fusion experimental device currently under construction at the ENEA Research Centre in Frascati, Italy. Its main mission is investigating the feasibility of different approaches to handle power and particles exhaust in a tokamak reactor, realizing high performance plasmas in a very flexible environment able to achieve different magnetic configurations with large external heating in integrated conditions, as part of the European Roadmap to Nuclear Fusion. The challenging design goals of DTT (plasma current up to 5.5 MA, pulse duration up to around 100 s, capability of accessing Single Null, Double Null, X-Divertor and Negative Triangularity magnetic configurations) have been achieved in a relatively compact (major radius 2.19 m) tokamak with superconducting coils and tungsten, actively cooled first wall and divertor.

In DTT, as in other tokamaks with high plasma current, the impact of plasma disruptions is a key driving factor in the design of vacuum vessel and components inside it, because of the large induced EM and thermal loads on them. This work describes the criteria adopted in the definition of the reference disruptions parameters and expected frequency: the worst-case events estimated in DTT are Major centred Disruptions (MD) and upward and downward hot Vertical Displacement Events (VDE), with a current quench duration in the range from 5 to 40 ms. These disruptions have been simulated by Maxfea code and the resulting plasma evolution and EM loads have been used to assess the tokamak structure. Even if DTT has been designed to sustain a relatively high number of worst-case disruptions, a Disruption Mitigation System has been proposed, relying on the use of Shattered Pellet Injection, and its conceptual design is in progress: some preliminary requirements are also reported.

Keywords

Disruptions, DTT, EM loads, MAXFEA, DMS.

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PS4-72 ABSTRACT-6f82
E. Vacuum Vessel and Ex-vessel Systems

Thermo-mechanical Analysis of the Components in the Launcher Transmission Line of ITER Ex-Port Plug Collective Thomson Scattering

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ITER Collective Thomson Scattering (CTS) diagnostic is designed to diagnose the alpha particles resulting from Deuterium-Tritium fusion reactions. It consists of a one launcher and nine receiver transmission lines. The launcher line transports the high-power microwave emission of 1MW, from the gyrotron source in the Assembly Building to the front-end. On the other hand, the receiver lines transport the collected microwave emission from the front-end and distribute it to the instrumentation in the Diagnostic Building.

Due to the high power transported in the launcher line, many of its components have to be water-cooled to minimize their thermal expansion resulting from the heating caused by the power losses. Therefore, a detailed thermo-mechanical analysis of the main components in the launcher line has been performed to evaluate their behavior under every load combination defined in the System Load Specification.

The thermo-mechanical analyses performed in this work take into account the loads on every component calculated previously in a global analysis of the ex-PP transmission lines [1]. The material considered initially in the CTS was aluminium 6061-T6 for Non-SIC components and stainless steel SS316L for the SIC ones. But the results obtained in this work show that miter-bends, sliding-joints and rotational-joints should be made entirely of CuCrZr, no matter their classification.

[1] E. Rincón, E. Blanco, M. Medrano, JJ. Imaz, Y. Villalobos, L. Maldonado, P. Varela, Y. Liu, V. Udintsev; "Structural Analysis of the Ex-Port Plug Collective Thomson Scattering Transmission Lines for ITER". Submitted to Fusion Engineering and Design in October 2022.

Keywords

ITER, CTS, diagnostic, thermo-mechanical analysis, FEM, miter-bend, cooling.

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PS4-73 ABSTRACT-ee57
E. Vacuum Vessel and Ex-vessel Systems

Preparation for Korean procurement scope of ITER sealing flanges

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ITER is considering four stages for the overall research activities: First Plasma, Pre-Fusion Power Operation 1(PFPO-1), PFPO-2, and FPO. Based on these steps, a sealing strategy for the ITER vacuum vessel was established. The ITER accommodates three basic configurations: elastomer seals, spring energized-metal seals, and lip seals. The sealing types have determined depending on what is installed in the port at each stage. The Korean scopes are to cover 17 equatorial ports including the NB ports and three lower ports. The procurement activities for the Korean scopes will be implemented soon through contracts with industries. This paper deals with not only the Korean procurement scopes of the sealing flanges but also key requirements and other information such as the latest status.

Keywords

ITER, vacuum vessel, sealing flanges, metal gaskets, elastomer gaskets, lip seals.

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PS4-74 ABSTRACT-f325
E. Vacuum Vessel and Ex-vessel Systems

First Experiences of the new Cryo Vacuum Pumps at Wendelstein 7-X during Hydrogen Operation

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In the first Operation Phases of Wendelstein 7-X (W7-X) the vacuum system mainly operated with a set of 30 Turbo Molecular Pumps (TMP) with a nominal pumping speed of 2000 l/s each. Meanwhile, W7-X is equipped with NBI heating and the commissioning of a new, efficient pellet injector will be finished soon. Both components introduce high H_2- and D_2-throughput into the plasma vessel by design.

To cope with the expected high H_2-throughput, a set of 10 cryo vacuum pumps (CVP) was installed in the plasma vessel during the last maintaining phase. The CVPs are designed for a total pumping speed in the vessel of 60 m^3/s (H_2) at an operation temperature of approx. 3.8 K.

The paper describes the installation process as well as the intense leak testing of the CVPs. Experiences of the first cool down with an estimation for the duration necessary to get stable conditions will be presented. The measured pumping speed of approx. 65 m^3/s (varying slightly with the operation parameters) meets the design value. The measured vessel pressure for loaded CVPs of 7e-6 to 1.3e-5 mbar will be discussed in dependence on the CVP temperature and other operating conditions.

The interplay of the CVP with the TMPs is reported with special focus on the regeneration process of the CVPs. First values for the duration of the regeneration process will be presented and discussed taking into account possible scenarios of the regeneration process.

For explosion safety reasons the maximum pressure of H_2 allowed inside the plasma vessel is presently restricted to 1.6 mbar, which limits the amount of H_2 bound on the CVPs for future experiments or will make a more frequent regeneration process necessary. Planned changes in the vacuum system to overcome these restrictions are discussed.

Keywords

Cryo vacuum pump, pumping speed, regeneration process.

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PS4-76 ABSTRACT.- af9a
E. Vacuum Vessel and Ex-vessel Systems

Final design analysis of the Optical Hinge of the Wide Angle Viewing System for ITER Equatorial Port 12

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The ITER Visible/Infrared Wide Angle Viewing System (WAVS) is a machine protection optical diagnostic that is being developed for ITER. The first two ex-vessel components of WAVS are the Optical Hinge (OH) and Optical Relay Unit (ORU) and both share a common support structure. Equatorial Port 12 (EP12) WAVS should be operational for the first plasma and its facing its Final Design Review (FDR). It is being developed by the Consortium constituted by CEA, CIEMAT, INTA and Bertin Technologies through a Grant Agreement financed by F4E.

To compensate vacuum vessel vertical displacements respect to the building during operation, an Optical Hinge (OH) will be placed at the beginning of the ex-vessel area for each Line of Sight (LoS). Each OH is formed by a pair of mirrors one of which is moveable using a piezoactuator. The OHs are attached to the common support which is, in turn, fixed to the Interspace Support Structure.

To assure the optical performance, the common support structure has to withstand operational loads with low deviation and tilt. In addition, the structural integrity of the common structure has to be maintained in case of seismic or accidental loads as it is classified as a Safety Relevant component.

This work exposes the analyses performed to face the FDR of ex-vessel WAVS in EP12. It comprises the structural analyses performed by CIEMAT for the OH-ORU common support structure to assure its integrity according to RCC-MR Code and the detailed model of one OH LoS to check also the diagnostic optical requirements.

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Keywords

ITER diagnostic, WAVS, Optical Hinge, structural analysis, FEM.

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PS4-78 ABSTRACT-aec8
F. Nuclear System Design

Fatigue analysis based on LEFM for a HTS CS coil of the next generation fusion reactor

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Conceptual design of the next generation fusion reactor Central Solenoid (CS) coil had been started in Institute of Plasma Physics, Chinese Academy of Sciences. The Central Solenoid (CS) is one of the important sub-systems of the Tokamak Magnet System and it contributes to the inductive flux to drive the plasma, to the shaping of the field lines in the diverter region and to vertical stability control. The CS coil consists of a HTS sub-coil and a LTS sub-coil. The peak field of is 19.2 T at a current of 46.5 kA. A fatigue crack growth rate analysis based on LEFM assessment for the CICC jackets is carried out to ensure that a life of 60,000 cycles. For the fatigue assessment of the HTS and LTS CICC jackets, the chosen route is based on the cut-boundary method. The displacements and loads are transferred from a global model to a local one. And a detailed model with a semi-elliptical surface defect is modeled. The properties c and m of different materials were measured by experiments in the Paris law, and the relationship between the number of cycles and cracks was calculated by Paris formula.

Keywords

Fatigue assessment, central solenoid, jacket, LEFM, Paris law, tokamak.

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PS4-79 ABSTRACT-b302
F. Nuclear System Design

IFMIF DONES lithium pump design studies

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Electromagnetic (EM) induction pumps are used in the industry to transport liquid metals. This type of pumps has no moving parts in melt resulting in a hermetic construction ensuring higher safety. Over last two decades EM pumps based on permanent magnets have proven to be an efficient and relatively simple solution to many industrial and research applications.

Current research deals with design study of IFMIF DONES 100 l/s lithium electromagnetic induction pump with rotating permanent magnets. The aim is to search for the optimum balance of pump flow rate against the drive unit power consumption, hydraulic losses in the pump channel, and thermal losses in the channel walls. The pump design is based on the existing 50 l/s flow rate sodium pump. Its 6800 mm^2 cross-section Ω-shaped channel has three parallel internal flow passes and a laminated outer ferrous yoke. Pumps 385 mm outer diameter magnetic rotor with six magnetic pole pairs is assembled from SmCo magnets and formed in a Halbach array. The pump is tested on the DN 125 Tesla hydraulic test loop gaining data on pump flow rate, developed pressure, temperature regime, vibration, and acoustic spectrum data. Pumps design parameter extrapolation is based on analytical models and numerical simulations with OpenFoam, COMSOL and ANSYS Fluent software.

The use of EM pump based on permanent magnets allows easy modification of rotors moment of inertia and thus the control of the flowrate decay for safe shutdown in case of electricity dropout. The results of pump runout experiments are presented alongside with theoretical model showing a good agreement and demonstrating the feasibility of effective flowrate decay time control.

Keywords

Magnetohydrodynamics.

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PS4-80 ABSTRACT-b80b
F. Nuclear System Design

TBM Piping Systems: Design according to RCC-MRx code

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The ITER Test Blanket Modules (TBMs) are installed and operated inside the vacuum vessel (VV) at equatorial ports located within port plugs (PP). The Pipe Forest (PF) is a network of pipes connecting the Ancillary Equipment Unit (AEU) to the TBMs. Depending of the TBM, the Pipe Forest includes Water Cooled (350°, 188 bars), Helium Cooled (500°, 100 bars), Lithium Lead and Tritium Extraction System piping. During operating states, the PF main function is to withstand loads in normal, incidental and accidental situations according to RCC-MRx code criteria (including design rules for mechanical components for fusion installations). It includes severe thermomechanical analysis due to the thermal expansion added to seismic and vertical displacements events. In addition, the PF layout must satisfy constructability in constraint environments and maximize the space available for man access, which constitutes a real challenge for design.

Since no existing RCC-MRx post-processing tool exists off the shelves, a specific methodology for pipe stress analysis is being developed by CEA and ASSYSTEM. This methodology starts with a post-processing calculation tool of stresses and reaction forces for looping back into the structural calculation. Moreover, a specific extraction module from 3D model to processing tool allows the conversion of CAD data to calculation entry files. At last a feedback loop is also implemented in the basic CAD model through an import module, to trace back displacements and evaluate the impact of integrating pipe movements.

This paper describes the pipe stress analysis methodology and its application for Pipe Forest system design life cycle. It covers history of the routing evolution towards introduction of expansion loops and more recently integrated calculation of the complete port cell from the back of the TBM sets up to the connection pipes of the vertical shaft.

Keywords

ITER, Test Blanket Modules, Pipe Forest, Piping Stress Analysis, RCC-MRx.

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PS4-81 ABSTRACT-bafa
F. Nuclear System Design

Conceptual Design Study of a Large Bore Superconducting Test Facility Magnet, SUCCEX

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Conceptual design modification studies of the 16 T steady-state Korean fusion demonstration reactor (K-DEMO) toroidal field (TF) coil have been performed since 2018. A high field superconducting magnet for testing the K-DEMO TF conductor samples are also being designed in parallel to prepare for the development of the K-DEMO TF conductor. A 16 Tesla, large bore superconducting split pair solenoid magnet is considered as the superconducting conductor test magnet. Its background field is expected to be over 15 T in a large bore of 1 m diameter and can reach 16 T with the self-field of K-DEMO TF conductor sample. To reach the target magnetic field, the high current density Nb3Sn strands (Jc > 2400 A/mm^2 at 4.2 K, 12 T) is applied in the design of 2 types of CICC - high field inner coil (IC) CICC and low field outer coil (OC) CICC. They connected in series with each other and the operating current is about 35 kA. In this study, the preliminary conceptual design results of the thermo-hydraulic analysis of 16 T conductor test magnet are presented.

Keywords

Fusion magnet, Cable-in-conduit conductor, high field magnet, Thermo-hydraulic análisis.

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PS4-82 ABSTRACT-c08f
F. Nuclear System Design

Development of compact gyrotron for start-up scenario in KSTAR ECH system

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Korea Superconducting Tokamak Advanced Research (KSTAR) is planning to install an electron cyclotron (EC) system for a total of 6MW Radiofrequency (RF) output and is currently in progress. A total of 4 EC systems were installed in the 2022 KSTAR campaign, but only 2 EC systems are in operation. Both EC systems have a problem with the gyrotron and are currently being repaired. The 4 gyrotrons currently installed are 1MW, 105/140Ghz dual-frequency gyrotrons that can operate for up to 300 seconds. In the 2020 KSTAR campaign, an EC system called EC4 provided about 1450 shots to the tokamak. Of these, about 500 shots were used for start-up with a pulse length of less than 0.5 seconds. This is about 35% of the total shot. We conclude that it is inefficient to use a high-spec gyrotron for low power less than 500kW and short pulses. Therefore, we are developing a start-up gyrotron with an RF power of 500kW and a pulse length of less than 0.5 seconds. The design value was determined based on the specifications of the gyrotron, which are mainly required by KSTAR. We will explain the specifications of the Gyrotron for startups in this paper. In addition, we will present in detail the design of the core components of the gyrotron for startups, such as cavity, electron gun, window, mode converter, and collector.

Keywords

KSTAR, ECH, Gyrotron.

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PS4-85 ABSTRACT-2616
F. Nuclear System Design

Neutronics assessment of the radar diagnostic for the IFMIF-DONES lithium target

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The scope of the IFMIF-DONES facility (International Fusion Materials Irradiation Facility – DEMO-Oriented NEutron Source) is to qualify materials under equivalent nuclear fusion irradiation conditions. The neutron source (up to ~5 × 10^{18} $m^{-2}\cdot s^{-1}$) is produced by means of the deuterium–lithium (D-Li) nuclear reactions that take place in a liquid Li target when it is bombarded by a 125 mA of 40 MeV deuteron beam, with a maximum footprint of 200 mm × 50 mm.

The Li target is one of the most critical parts of the facility, where the high power beam (5 MW) will impinge on a 15 m/s Li jet at 300 ºC. Target thickness needs to be kept within 25 ± 1 mm for the safe absorption of beam power, avoiding critical damage in the back plate. On-line inspection of the liquid Li thikness and superficial stability is crucial for the proper operation of the IFMIF-DONES facility (including accident scenarios). The real-time monitoring and diagnostic of the Li target conditions are of primary importance, but the extreme irradiation conditions in the Li target area requires the development of new special instrumentation. A diagnostic system, based on radar techniques in the millimetre wave (mmWave) band, has been proposed to continuously monitor the lithium thickness variations in the beam impact area, as well as to detect and study instabilities and perturbations in the liquid lithium jet.

It is essential to determine if the functional properties of the diagnostics materials will not be affected/compromised during an irradiation campaign. This work summarizes particle transport calculations performed to determine the neutron and gammas doses absorbed by the materials foreseen for the antennas, waveguides and RF-window, which are located inside the Test Cell. Besides, isotopic inventory calculations are provided to define the type of handling maintenance protocols for these components

Keywords

IFMIF-DONES, Neutronics, diagnostics, MCNP, Materials.

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PS4-86 ABSTRACT-d463
F. Nuclear System Design

Thermal and structural analysis of the DONES Target System under steady state and transient loading conditions

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One of the crucial steps in the European Roadmap to the realisation of fusion energy is the design and construction of the Demo-Oriented NEutron Source (DONES) facility. DONES is a fusion-like neutron source, based on the International Fusion Materials Irradiation Facility (IFMIF) concept, aimed at qualifying and testing the materials to be used in fusion reactors. The Target System (TSY) is crucial subsystem as it is the component where the interactions between deuterons and Li flow take place to produce neutrons for irradiating the testing materials. In this regard, an investigation campaign has been performed to evaluate the thermo-mechanical behaviour of the DONES TSY under selected steady state and transient loading conditions, in light of the most recent design updates (new bellows and connection systems).

The first part of the assessment concerned the analysis of the TSY under the normal operation steady state loading scenario. A detailed 3D FEM model has been set up, considering the newest updates of the TSY geometry and of the operational parameters, and a thermo-mechanical analysis has been launched. Results have been mainly checked in terms of TSY global deformation, focussing on the Back Plate and on the connecting bellows displacements, in order to exclude overlapping and misalignments with the High Flux Test Module and the Li footprint.

Afterwards, since the Lithium inlet temperature has been increased from 250°C to 300°C, a transient thermo-mechanical analysis of the TSY under the start-up loading conditions has been performed to check if it is able to withstand the hotter lithium flow according to the RCC-MRx criteria. Therefore, the reference pre-heating strategy has been adopted, assessing the thermal and mechanical behaviour of the TSY, focussing on the Back Plate.

The obtained results are presented and critically discussed, giving clear indication for the follow-up of the DONES TSY design.

Keywords

IFMIF, DONES, thermo-mechanics, FEM, Target System.

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PS4-87 ABSTRACT-d66c
F. Nuclear System Design

Nuclear and sensitivity analyses in Equatorial Port Plug, Port Interspace and Port Cell #12 in ITER

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The International Thermonuclear Experimental Reactor (ITER) Equatorial Port (EP) #12 is a diagnostic port based on the long-modular Diagnostic Shielding Module (DSM). The Equatorial Port Plug (EPP) #12 will host six diagnostics which are integrated together in the three DSM cassettes. The EPP is not only a diagnostic port, but also it has to provide the necessary shielding to allow for the maintenance operations in Port Interspace (PI) and in Port Cell (PC). The necessary shielding shall be obtained with B_4C tiles mounted on trays which are inserted inside the DSMs. In this work a very detailed MCNP model of the PI and PC have been developed and integrated within a 40° ITER sector (C-model).

The model has been used to perform nuclear analyses, employing the D1SUNED v3.1.4 code aimed at evaluating the nuclear responses in the PI and in the PC areas. The impact of radiation cross-talk with neighbouring ports has been assessed during and at the end of ITER operations. A sensitivity study has been also performed by replacing the B_4C tiles in the first two columns with different shielding materials. Results are presented and discussed.

Keywords

ITER, Neutronics, Equatprial Port, Monte Carlo.

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PS4-88 ABSTRACT-dc8d
F. Nuclear System Design

Deterministic neutronics for stellarator system design

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Stellarators are complex, 3D shaped, yet promising candidates for a fusion power plant that can come in different variations – spanning a large design space. To investigate this design space, comprehensive systems codes are necessary that can model the complex stellarator geometries. However, appropriate stellarator models have only been introduced recently [1] and a blanket model that can generate important neutronics parameters, like the inhomogeneous wall load, tritium breeding ratio, or fast neutron flux at the coils in any stellarator geometry, is still lacking.

Conventional Monte-Carlo methods can be used to assess these neutronic parameters, but require substantial manual setup and heavy computational resources, incompatible with broad system studies. To mitigate these circumstances, we propose the use of deterministic methods, i.e. exploring mathematical methods to directly solve the neutron transport equation. For example, a recently developed matrix formulation for the neutron streaming can estimate the neutron wall load in stellarator geometry within a few seconds [2]. Similarly, a method for solving the neutron transport equation in automatically generated blanket geometry is currently under development. It uses a discrete ordinates, multigroup velocity space discretisation and a discontinuous Galerkin spatial discretisation. These deterministic methods can then be used to calculate the relevant neutronics design parameters in arbitrary stellarator geometry quickly enough for stellarator system studies.

In the first method, a wall region did not block the neutron flux from propagating to another wall behind it. A raytracing method resolves this, but did not change the resulting neutron wall load by more than a few percent, showing that the method can be applied even without this raytracing correction. The latter method has been benchmarked successfully inside the plasma domain and is currently being extended to the blanket domain, after which it will be benchmarked with existing Monte-Carlo methods.

Keywords

Stellarator, neutronics, system design.

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PS4-89 ABSTRACT-eb05
F. Nuclear System Design

Manufacturing Demonstration of a Step Inboard Neutron Shield

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A key challenge in the concept development for the Spherical Tokamak for Energy Production (STEP) is the manufacture of the inboard neutron shielding. The main requirements of this component are to reduce neutron flux on the inboard magnets, manage nuclear heating, and extract thermal power. These requirements must also be met whilst minimising the overall thickness of the shield to maintain the size and aspect ratio of STEP. As such, few materials are suitable for this application, where two primary candidates are Tungsten and Tungsten Carbide. This is due to their high neutron and gamma attenuation, high density, low-activation properties, and thermomechanical properties.

Assessment of the possible manufacturing strategies for the inboard shield identified several limits and gaps in knowledge. Such limits mainly arose from the modularisation of this component, as is required due to its large size compared to typical size constraints for tungsten ceramic manufacture. This highlighted gaps in knowledge for geometric constraints for tungsten ceramic forming, joining with low-activation interlayers, and dissimilar joining. These unknowns were addressed via a tungsten and tungsten carbide manufacturing demonstration for the main features of the inboard shield. This included permanent and semi-permanent joints, dissimilar joints, bonded coolant pipes, and cross-joint coolant channels.

Keywords

Neutron Shielding, Tungsten, Tungsten Carbide, Manufacturing, Low-activation Interlayers, Dissimilar Joining.

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PS4-90 ABSTRACT-6f53
F. Nuclear System Design

Overview of European Fusion Neutron Source activities within the IFMIF/EVEDA Project

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The fusion roadmap defines as a key facility for the fast track of fusion development a fusion-like neutron source for testing the candidate materials for fusion reactors. The Fusion Neutron Source (FNS) is conceived to generate fusion-relevant neutrons through Li(d,xn) stripping nuclear reactions, with a linear particle accelerator to obtain an intense deuteron beam (125 mA, 40 MeV) impinging onto a liquid lithium target. Neutrons will irradiate, under controlled conditions, the candidate samples in the Test Systems.

Since 2021 Europe and Japan are developing different FSN facilities. However, some common FNS design activities have been defined in the frame of a new international collaboration, the Broader Approach Phase Two (BA-II), in addition to the International Fusion Materials Irradiation Facility/Engineering Validation and Engineering Design Activities (IFMIF/EVEDA). In the BA-II the Engineering Design (ED) and Lithium Facilities (LF) design activities required for advancement in an FNS design have been defined, including the use of the Linear IFMIF Prototype Accelerator (LIPAc), and the analysis of the operational experience reliability data collection, as a key reference facility for future FNS construction.

The engineering, modelling, calculation, and experimental activities inside ED Procurement Arrangements include the tritium migration estimation, erosion/deposition modelling in the lithium loop, accident analysis in Safety, optimization of the Li-oil heat exchanger, and the use of LIPAc as a testing facility. The LF Procurement Arrangements activities include the Li purification system validation activities by means of pilot plants, Li target diagnostics design and validation, and erosion/corrosion analysis and modelling on the materials of a dismantled test loop, the EVEDA Lithium Test Loop (ELTL).

The status of the activities carried out for the European FNS (EU-FNS) activities are depicted and the forthcoming stages are described in this publication.

Keywords

IFMIF/EVEDA, Broader Approach, Fusion Neutron Source, LIPAc.

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PS4-92 ABSTRACT-f904
F. Nuclear System Design

An approach to improve the tension constancy of a TF coil shape

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An ideal TF coil shape or centerline should be such that the TF coil are subjected to constant tension. The Princeton-D and the Modified-D are widely used as constant-tension centerlines for designing TF coils, however, our analyses show that their tensions are not fully constant. The reason is that the magnetic fields used to calculate the centerlines are inaccurate. In this work, we propose an approach of calculating the Optimized-D shape with improved tension constancy. First, the Princeton-D and the Modified-D are introduced and their tension non-constancies are revealed. Then, a code is written to calculate the accurate toroidal fields. Next, the Optimized-D coil shape is obtained through a semi-analytical iteration process. Finally, for the three D shapes, the relationship between the TF coil parameters are discussed. Our results indicate that the Optimized-D shape realize a noteworthy improvement in tension constancy; In addition, from the perspectives of reducing budget and enlarging space between coils, the TF designer should maximize the current carried by one TF coil and thus minimize the number of TF coils, as long as the ripple is permissible.

Keywords

Tokamak, TF coil, superconducting, constant tension, Princeton-D, centerline, toroidal field.

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PS4-93 ABSTRACT-cbf6
G. Safety Issues and Waste Management

Potential of radioactive isotopes production in DEMO for commercial use

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There is widespread use of nuclear radiation for medical imagery and treatments. Worldwide almost 30 million treatments are performed per year. There are also applications of radiation sources in other commercial fields such as industry for weld inspection or in steelmaking processes, in consumer products, in food and agriculture. The large amount of neutrons generated in a fusion reactor such as DEMO could potentially contribute to the production of the required radioactive isotopes. The associated commercial value of these isotopes could mitigate the capital investments and operating costs of a large fusion plant.

There are numerous radioactive isotopes with commercial application. These include both short- and long-lived nuclides such as ^{213}Bi ($T_{1/2}$=43 min) and ^{60}Co ($T_{1/2}$=5.27 years) respectively. The majority are produced in fission reactors with an exposure time from hours to years. The efficiency of the isotopes production depends directly on the neutron fluence applied for this purpose. Material pieces arranged in a DEMO port will be subjected to the exposure of the intensive neutron spectrum suitable for a high isotope production rate.

The potential of producing radioactive isotopes in the DEMO equatorial port plug has been studied. For this purpose, the full 3D geometry model of the 22.5° DEMO toroidal sector with irradiation chambers arranged in an equatorial port plug has been modelled in an MCNP code to perform neutron transport simulations. Subsequent activation calculations provide detailed information on the quality and composition of the produced radioactive isotopes. The technical feasibility and the commercial potential of the production of various isotopes in the DEMO port are assessed.

Keywords

DEMO, neutronics, radioactive isotopes, applications.

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PS4-94 ABSTRACT-ce81
G. Safety Issues and Waste Management

ACABLoop simulation tool: improving the activation prediction of flowing PbLi alloy in support of DEMO fusion reactor design

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UNED

PbLi alloy as breeder material is considered within the EUROfusion Programme in the Water-Cooled Lithium-Lead (WCLL) blanket concept of DEMO fusion reactor. The PbLi travels along loops entering and leaving the Breeding Blanket (BB) with several missions: i) to recover the produced tritium, ii) to remove generated impurities and activated corrosion products, iii) to produce gravitational draining of the BB modules, iv) to maintain the PbLi in liquid state, and v) to transport the blanket heat to the steam generators. PbLi is activated by neutrons inside the BB, undergoing decay along the rest of the loop and mass incorporations and extractions, being this process repeated along many cycles.

Prediction of activation-related responses in the flowing PbLi is a key safety issue in support of DEMO design, being a demanding calculation because of the complex features of the loops.

Traditionally, the activation inventory generated in the flowing PbLi has been calculated considering a simplistic approach, valid only for a pre-conceptual analysis. Additionally, the simulation of some phenomena is not possible when using that simple methodology.

ACABLoop has been conceived as a tool to overcome such limitations predicting more realistically the activation of the PbLi alloy. ACABLoop, which is based on the validated ACAB activation code for fusion applications, can simulate the activation of the flowing fluid considering the above-mentioned issues, providing all the information related to the generated isotopic inventory in the fluid.

Status of ACABloop development is presented as well as some applications for PbLi activation in DEMO loops, proving its suitability for fusion activation calculations. Additionally, a first validation of ACABLoop using a water loop and a D-T fusion neutron spectrum shows very promising results. Last improvements of the code are devoted to allowing incorporation of CFD information as a tool for increasing the reliability in some specific situations.

Keywords

Activation, Fluid-activation, DEMO, code-depelopment.

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PS4-95 ABSTRACT-d8db
G. Safety Issues and Waste Management

The LiFIRE experimental facility: final design, construction and experimental campaign

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Recently, nuclear fusion-related experiments and dedicated facilities worldwide relying on the usage of liquid metals are experiencing a renewed interest.

This is the case of the International Fusion Materials Irradiation Facility – DEMO-Oriented Neutron Source (IFMIF-DONES), a radioactive facility which will use several cubic meters of pure liquid lithium as a target for deuterons to produce high neutron fluxes with a similar energy spectrum expected in future fusion reactors. The main goal of IFMIF-DONES will be to irradiate selected candidate structural materials for building a comprehensive database on fusion material properties.

Generally, alkali metals, such as lithium, exhibit safety issues related to their reactive nature and material compatibility. Due to the wide uncertainty regarding the lithium behavior confirmed by an extensive literature review, a dedicated experimental facility has been conceived to primarily study lithium ignition conditions in support of the licensing process of IFMIF-DONES. The ultimate purpose of the LiFIRE facility will be to provide experimental evidences for demonstrating the set of requirements on lithium fire protection to be applied to the final engineering design of IFMIF-DONES.

This work describes the final design, construction and experimental campaign of the LiFIRE facility currently under development at CIEMAT’s premises.

Keywords

Liquid metal, Lithium, ignition, fire protection, fusion safety, IFMIF-DONES.

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PS4-96 ABSTRACT-4b32
G. Safety Issues and Waste Management

Safety Issues of Lithium-Water Reaction for Fusion Reactor

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The liquid lithium (Li) limiter is considered as a promising plasma facing component (PFC) for fusion reactors. And meanwhile water is used as the coolant for the shield blanket. However, when the Loss of Coolant Accident (LOCA) happened, the Li will probably react with water. The energy caused by the reaction will probably damage the reactor structure and result in the release of radioactive products to the environment. The study on Li-water reaction is very necessary and important for blanket design and reactor safety operation. It also provides the scientific evidence for risk analysis and prevention of the accidents.

The small-scale lead alloy/water direct contact test was performed and investigated. The small-scale apparatus was constructed to investigate the mechanism through releasing Li droplet into water tank. The interaction has been conducted at different initial injection pressures, drop temperatures and droplet masses. The results showed that the pressure increased sharply to the maximum value and then decreases slowly to the constant pressure. The pressure peak increased slightly with the increasing temperature of Li. The temperature increases, strictly connected to the mass of Li. A hydrogen production rate model with the initial condition, such as the temperature and the amount of Li, initial injection pressure, the temperature difference, etc, was established.

Keywords

Li-Water reaction, reaction mechanism.

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PS4-97 ABSTRACT-dfae
G. Safety Issues and Waste Management

Simulation of tritium trapping on impurities in beryllium

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Beryllium is envisaged as a neutron multiplier material for the Test Blanket Module of ITER and DEMO fusion reactors. Beta-radioactive tritium is generated as a by-product of neutron multiplication and can be accumulated in beryllium in considerable amounts. As far as total tritium inventory inside any fusion reactor should be limited, excessive tritium accumulation in beryllium can present a serious safety issue and even hinder the utilization of beryllium after the end-of-life of a fusion reactor.

In this work, the interaction of hydrogen isotopes with impurities in beryllium is studied at the atomic scale by application of first principles methods. Binding energies of interstitial hydrogen with typical substitutional impurity atoms (Al, Fe, Si, Mg, Cr, etc.) in beryllium are calculated. It was found that oversized impurities like Al and Si repel interstitial hydrogen, while undersized ones can slightly attract it. Hydrogen trapping in vacancies is significantly affected by impurities being the nearest neighbors of a vacancy. We show that some impurities can increase hydrogen binding, whereas others decrease it with respect to binding with a single vacancy.

These results are in accordance with the experimental findings showing the association of nanoscale bubbles with Al-Fe-Be precipitates as observed by TEM. The implications of tritium trapping at impurities for overall tritium accumulation in beryllium at different irradiation temperatures are discussed.

Keywords

Tritium, beryllium, ab initio, computer simulation.

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PS4-99 ABSTRACT-ef14
G. Safety Issues and Waste Management

Improving the efficiency of MELCOR modelling for accident analysis of a tokamak

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Fusion power is a critical part of the future energy landscape yet is all too often depicted with the focus on the fusion reaction itself with little attention on supporting systems. In practice of course a real power plant will have the tokamak surrounded by a plethora of ancillaries: the fuel cycle, cooling systems, diagnostics, radiation management, controls, and so on. Understanding the consequences and mitigation for possible accidents is crucial for public acceptance but requires navigating this complex web of support systems.

MELCOR is a software tool for accident analysis with a long history in the fission sector. A model has been developed for use in MELCOR, using proposed layouts of DEMO and assumptions based on current design choices, to facilitate safety design and development.

We have streamlined the running of MELCOR tasks to allow simultaneous execution on a computing cluster. The input decks are version-controlled within UKAEA's internal GitLab repository, and shell scripts wrap the separate MELCOR invocations, embed specific job parameters, and keep separate the inputs and outputs of multiple separate instances. We illustrate the improved workflow with two examples studying tritium release following a Postulated Initiating Event of a guillotine break in the pumping line of a torus pumping system: a study of the effect of various locations for safety relevant valves around the system, and a 20-point parameter sweep looking at the effect of varying delays in valve closure times on the amount of tritium released from primary pipework.

Results of the sensitivity analysis for valve closure delay indicate counter-intuitively that as the delay initially increases up to 120 s the total amount of tritium released decreases; with response delays of 180 s and longer the tritium release increases as expected. This parameter sweep outcome demonstrates the advantage of readily running MELCOR tasks in bulk.

Keywords

MELCOR, job execution, accident analysis, safety, safety design, fission-fusion synergy, tritium.

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PS4-100 ABSTRACT-f3e6
G. Safety Issues and Waste Management

Assessment of radiation dose rate in main building area of A-FNS facility

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The A-FNS design activity aims to construct a neutron irradiation facility for fusion neutron irradiation tests on the fusion reactor materials. As one of radiation shielding analyses taking into account the construction, the radiation dose rate in the A-FNS has been calculated by Monte Carlo simulation. While the base model of A-FNS adopted from IFMIF project for the simulation, however, it is foreseen that A-FNS has a different dose rate distribution than IFMIF due to design changes such as the installation of beam ducts for application cells and lateral access cells which IFMIF did not have. It is required to evaluate the dose rate due to the design changes, and to perform radiation shielding design that can satisfy the required values using the detailed calculation model. The calculation model was created based on the 3D CAD drawings of A-FNS main building. Monte Carlo calculations were performed for estimation on the detailed dose distribution using the variance reduction method with MCNP6.2 and ADVANTG codes. In the reference design, which has no beam duct for the application usage, the neutron dose rate in the application cell is below 1e-2 microSv/h. However, the dose rate increases by eight orders of magnitude to a few Sv/h for the design with the duct of 10 cm in diameter. As a point of view of radiation zoning for A-FNS facility, the application cell should be "yellow" which means that allows restricted access under 2 mSv/h dose rate. Since the beam duct could make high dose rates on the outside wall of the building expected than an uncontrolled area (<0.6 microSv/h), improved shielding design by increasing additional local shields and building walls should be considered. In this symposium, the results of the dose rate distribution in the A-FNS main building, the strategy of shielding design are presented.

Keywords

Radiation Dose rate, MCNP, ADVANTG, A-FNS, Shielding análisis.

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PS4-101 ABSTRACT-f63b
G. Safety Issues and Waste Management

Instantaneous Risk Monitoring Method for Fusion Reactors based on Level Three Probabilistic Safety Assessment

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With a series of achievements in the R&D of fusion devices in recent years, the safety of fusion reactors is receiving more and more attentions. However, risk monitoring, as an important technical basis for the operation safety, has been seldom studied. Conventional risk monitors for fission reactors generally support the core meltdown risk assessment based on the level-1 probabilistic safety assessment, which cannot fulfill the operation demands of fusion reactors. Furthermore, some special design features of fusion devices resulted in different safety concerns, for example, the possibility of a large amount of tritium release from the Vacuum Vessel or Tritium Plant.

Focusing on the safety supervision of fusion reactors, an improved real-time risk monitoring method based on the level-3 probabilistic safety assessment is proposed in this contribution. In order to supports the calculation of source terms and off-site dose risk, event tree method is used to model the accidents progress and the safety systems response, and fault tree method is used to reflect plant configuration changes in the operation for the analysis of accidents frequencies. The accidents sequences in event tree models are categorized according to the severity of their consequences, which is determined by the radionuclides release to the environment. The effectiveness of the proposed method can be demonstrated by case study and application, with the support of self-developed Reliability and Safety Assessment Program RiskA. Results show that tritium process and related components are very important for the operation safety, which are different from fission reactors. The proposed method will provide

Keywords

Safety Regulation, Probabilistic Safety Assessment, Risk Monitoring.

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PS4-102 ABSTRACT-fd13
G. Safety Issues and Waste Management

Assessment of air activation discharge from JET during DT operations

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During its most recent tritium operational campaign, DTE2, the Joint European Torus (JET) broke the international record for sustained fusion energy, achieving 59 MJ. Whilst an historic achievement for JET and a huge milestone in the Nuclear Fusion industry, large, high energy neutron yields induced significant levels of neutron-induced activation in the tokamak and its surrounding environment.

As well as the activation of structural and functional materials, the activation of air surrounding JET is also an important consideration, as the radionuclides produced are circulated and subsequently released to the environment. This discharge is monitored by the Environment Agency (EA) and it is the responsibility of UKAEA, as the operator of JET, to ensure the annual discharge of activity remains within specified limits.

It has been determined from inventory studies that the dominant radionuclide at decay times > 1 minute considered by the EA for annual reporting arises principally from neutron capture in argon-40, producing argon-41 with a half-life of 109 minutes. Following an internally conducted best available technique (BAT) assessment, it was determined that due to the relatively low activities produced, annual discharges should be calculated using Monte-Carlo based neutronics assessments in favour of a commercially available stack monitoring system.

In this submission, an overview of the calculation methodology is presented, which includes the application of a mesh-based discretisation approach to stochastically determine the spatial distribution of radioisotope production from air within the JET torus hall. Experimentally determined pulsed neutron yields for operational campaigns including DTE2 are subsequently utilised to explicitly calculate spatial activity distributions of dominant radionuclides. From these calculated distributions, bounding cases for activity discharge are subsequently determined to inform UKAEA's annual reporting requirement to the EA.

Keywords

JET, neutronics, waste, activation, simulation.

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PS4-103 ABSTRACT-97b4
H. Models and Experiments for FNT

Methods for characterization of voids and bubbles in tungsten using S/TEM

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Materials serving as plasma-facing components suffer from severe environmental degradation. Defects generated during 14 MeV neutron irradiation and interactions with plasma degrade the properties of materials, finally leading to the need of replacement. One of the most dangerous defects are voids or bubbles, as they may lead to a brittle cracking or swelling. These objects are usually nanosized, thus, the only experimental technique which enables their direct visualization is (scanning) transmission electron microscopy (S/TEM). The high-performance capabilities of modern microscopes seem to make possible observations of very fine objects on a daily basis. However, the complexity of available techniques, modes or other effects make it hard to find proper conditions for direct visualization, which enables qualitative as well as quantitative characterization. Especially the latter is very important and usually is neglected in electron microscopy studies due to difficulties related with understanding of a spatial distribution of observed objects. Here, we propose a combination of TEM and STEM techniques which allows to obtain results which shed light on nanovoids formation and their distribution in the material. Output of such experiments can be further used for advanced modelling or combined with other experiments. Our work focuses on tungsten, however proposed techniques may be easily applied for studies on other materials.

Keywords

Bubble, voids, electron microscopy, microstructures.

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PS4-104 ABSTRACT-9ca5
H. Models and Experiments for FNT

Optimizing Plasma Parameters for High Negative Ion Density in a Two-Region Arc Plasma ion source

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The Korea Atomic Energy Research Institute (KAERI) is developing a novel Two-Region Arc Plasma Ion Source (TRAP) as a Cs-free negative hydrogen (deuterium) ion source for a Neutral Beam Injection (NBI) system in a fusion tokamak. The TRAP ion source is based on a two-region configuration, comprising a high electron temperature region that creates highly vibrationally excited molecules and a low electron temperature region that generates negative ions by attaching electrons to molecules. This configuration can be achieved by optimizing the filament position and magnetic cusp field. In order to optimize the TRAP configuration, we investigate the plasma parameters under various operating conditions, such as filament position, gas pressure, and arc power. Electron density and temperature are determined using Langmuir probe measurements, while negative ion density is evaluated by photo-detachment measurements. Additionally, we analyze the uniformity of negative ion density to ensure a consistent ion beam. In this paper, the detailed experimental results are described and discussed.

Keywords

Neutral beam injection, Heating and current drive, Plasma arc source, Negative ion beam, Cs-free.

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PS4-106 ABSTRACT-a564
H. Models and Experiments for FNT

Current Status and Plan of Fusion Material and Blanket R&Ds using Various Ion and Neutron Irradiation Facilities at KAERI

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KAERI

In Korea, a fusion energy roadmap for the development of DEMO and fusion reactors is being embodied considering fusion neutrons sources (FNSs) to support engineering R&D like material, blanket, system engineering, licensing, code & standard and so on. Currently, various neutron sources such as 40 MeV accelerator-based neutron sources, assembly of cyclotron-based neutron sources, spherical tokamak-based neutron sources etc. are being considered. However, prior research is indispensable for the construction and operation of such large facilities. Therefore, KAERI examines various existing or proposed neutron sources, and focuses on small equipment that can be built or built in a short period of time. We are systematically carrying out the necessary previous studies (1) Irradiation test of fusion material using heavy ion accelerator (2) Neutron irradiation test using 30 MeV cyclotron based neutron source (3) Neutron irradiation test with research reactor, HANARO under ITER TBM program (4) Establishment of a reasonable neutron source selection and construction plan among the various options mentioned above. The overall status and plans related to these were briefly introduced in this paper.

Keywords

Neutron source, ion irradiation test, fusion material test, breeding blanket test.

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PS4-107 ABSTRACT-8922
H. Models and Experiments for FNT

IFMIF-DONES radiofrequency conditioning

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The IFMIF-DONES facility will be a relevant neutron source providing the irradiation conditions foreseen for the first wall of the DEMO fusion power plant. Its liquid lithium target bombarded with a 125 mA and 40 MeV deuteron beam, accelerated by a 5 MW RF linear accelerator (Linac) operating in continuous wave (CW) at 175 MHz, will generate neutrons by stripping reactions for candidate materials irradiation.

The IFMIF-DONES accelerator uses a first acceleration stage based on a Radiofrequency Quadrupole (RFQ) up to 5 MeV and a second stage, based on a five-cryomodules Superconducting Radiofrequency (SRF) Linac containing forty-six Half Wave Resonator (HWR) cavities, up to the final 40 MeV energy. To match both stages in longitudinal mode, the Medium Energy Beam Transport (MEBT) system will use two re-buncher cavities. All those cavities, and their corresponding power couplers, are designed to operate under Ultra-High Vacuum (UHV) and 6.5 MW of continuous RF power at 175 MHz, making them prone to some phenomena that can damage the components, like multipacting, vacuum bursts, or arcs, with a considerable radiation generation associated to these physical processes.

To prepare these components for nominal operation, a conditioning process based on applying RF in a controlled way to process the inner vacuum-exposed surfaces will effectively clean the component and reduce its reactivity to the RF power. Moreover, an optimized conditioning process including the necessary hardware is key to be able to complete the accelerator commissioning. Not only because a successful conditioning could even remove completely the dangerous phenomena, but also due to the need of completing this process in a reasonable timeframe within the accelerator commissioning phase of the project.

The design of the IFMIF-DONES RF conditioning process to be applied, including the proposed methodology, the hardware architecture, and the software automation, will be presented in this paper.

Keywords

IFMIF, DONES, RF, conditioning, multipacting.

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PS4-108 ABSTRACT-c1c8
H. Models and Experiments for FNT

Assessment of a millimeter-wave radar system for real-time diagnosis of the IFMIF-DONES lithium target

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IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO (DEMOnstration power plant) Oriented Neutron Source) is a fusion-like neutron source currently under design and development, where a high-power deuteron beam (5 MW) will impinge on a liquid lithium jet flowing at 15 m/s and 300 ºC. Neutrons will be produced from the deuteron-lithium stripping nuclear reaction with a similar nuclear spectrum to that obtained in a future fusion reactor, and they will be used to irradiate candidate materials to be studied for fusion applications.

The lithium target is one of the most critical parts of the facility, where keeping the lithium thickness within 25 ± 1 mm is essential for the safe absorption of the beam power. Due to the unprecedented extreme environment in the lithium target area (i.e., nuclear and gamma radiation, high temperature, lithium evaporation) and the space limitations, very few diagnostics are presently capable to operate in real-time during beam operation with the robustness and fast response time defined by the present IFMIF-DONES top-level requirements.

A diagnostic system, based on radar techniques in the millimeter wave (mmWave) band, has been proposed to continuously monitor the lithium thickness variations in the beam impact area, as well as to detect and study instabilities and perturbations in the liquid lithium jet. This system is expected to be in communication with the Machine Protection System (MPS) for the fast shut down of the deuteron beam in case of emergency.

This contribution describes the assessment of this radar-based lithium thickness diagnostic system for the IFMIF-DONES facility, taking into account the equipment compatibility and degradation under the harsh conditions expected in IFMIF-DONES.

Keywords

IFMIF-DONES, Lithium Target Diagnostic, Radar-based Diagnostic System, Nuclear Instrumentation, Novel Diagnostics for Hazardous Nuclear Reactor Environments.

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PS4-109 ABSTRACT-650c
H. Models and Experiments for FNT

LITEC: an experimental facility for the validation of the IFMIF-DONES Impurity Control System

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The Impurity Control System (ICS) of IFMIF-DONES is devoted to assuring lithium purity. After its circulation through the target, some undesired transmutation products will contaminate the lithium: deuterium, hydrogen, tritium, and beryllium. In addition, due to the high velocity of the flowing lithium, the target assembly, piping system, and other auxiliaries are subjected to a significant corrosion rate, thus leading to metallic impurities (iron, nickel, molybdenum…). Finally, nitrogen, oxygen, and carbon will also be present.

These impurities may compromise the operation of the facility: they may enhance the erosion/corrosion effect, shortening the operational life of equipment; they may be activated having an important impact on safety strategy, maintenance, and operational procedures; finally, they may affect fluid-dynamics, cavitation, and thermo-mechanics.

The ICS extracts a fraction of the liquid from the main loop to be purified in dedicated traps to maintain impurities and activated corrosion products below permissible levels to achieve the required safety conditions.

The validation of the ICS at a relevant circuit size will be performed in a new experimental facility called LITEC. It consists of a forced circulation lithium circuit. An electromagnetic pump provides a wide range of operations for testing all the purification systems envisioned in the ICS. It has four test sections, where trap prototypes will be installed (H trap, cold trap, etc.). The possibility to introduce impurities to simulate IFMIF-DONES conditions is also foreseen by an impurity injection system. Finally, a parallel branch will monitor online and offline impurities.

LITEC will contribute significantly to clarifying key aspects of the design and operation of IFMIF-DONES lithium systems and will be a reference facility for research activities in lithium technologies. In this work, its engineering design, operational parameters, and characteristics are presented.

Keywords

Ithium, purity, technologies, traps, experiments.

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PS4-110 ABSTRACT-c921
H. Models and Experiments for FNT

Neutronic analyses for EU DEMO upper limiter

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Design and integration of various systems into a functioning DEMO fusion power plant represents several challenges. These systems have to fit within spaces allocated for them, perform their main function while not interfering with other systems, and often also provide secondary functions to make sure the whole machine can operate reliably. In the case of plasma limiters for DEMO their main function is to provide protection for the first wall from both planned and unplanned plasma transients while also provide sufficient shielding to make sure that components located behind them, e.g. vacuum vessel and superconducting magnets, can perform their functions. Furthermore, the design must allow for a relatively easy extraction of the limiter independently from the tritium breeding blanket which dictates the size of the openings and complicates the routing of various cooling pipes.

Analyses were performed to provide assessments of the nuclear loads in Upper Limiter (UL) to guide the development of its design and integration. Nuclear loads such as nuclear heating, neutron induced damage to the material (DPA), and He production were assessed and the effect of using different cooling strategies and tritium breeding blankets quantified. More detailed concept of the UL was prepared based on initial results and higher fidelity analyses were performed with this model. These results informed 3D thermohydraulic analyses and are being used for the development of a suitable limiter cooling strategy as well as to come up with new more realistic limiter concepts.

Keywords

Nuclear heating, plasma limiter, neutron transport.

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PS4-111 ABSTRACT-ca86
H. Models and Experiments for FNT

Design and Testing of Windowless Gas Target for High Intensity Neutron Sources

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The High Intensity Neutron Sources employs high current D^+ beam to bombard tritium nuclides so as to generate high intensity fusion neutrons, which can be applied in fields such as neutron physics, nuclear medicine, radiation protection, and nuclear technology application, etc. A windowless gas target for neutron production with low energy particle beams will allow beam currents higher than will a solid or windowed target, and thus will allow greater neutron yield， is an important choice for high intensity neutron sources target.

The windowless deuterium gas target has been designed and constructed for High Intensity Neutron Sources. The gas target is capable of operation at 1000Pa deuterium gas pressure based on a differential pump vacuum system, and can admit a beam of 300keV, 30mA. With this target, it is expected to produce neutron yield of 3×10^{11}n/s. Testing result shows that the differential pump vacuum system can be match well with D+ ion accelerator with the expected performance.

Keywords

D-T neutron source, gas target, high intensity neutron sources.

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PS4-112 ABSTRACT-cfd8
H. Models and Experiments for FNT

Experiments for testing the H-trap and proposal of experiments in LITEC.

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The DONES/IFMIF Li Impurity Control System (ICS) is conceived as a sub-loop by passing the main loop. It will contain traps for the retention of impurities, monitors for on-line measurement of impurity content and a sample extraction system for off-line determination of impurity content by chemical analysis.

One of the most relevant impurities to be removed from the lithium liquid is the hydrogen. The reference procedure for purification of liquid alkali metals is by means of "getter" trapping. A reactive metal (V, Nb, Ti, Zr, Y…) is used as getter metal which forms a thermodynamically stable compound with this impurity.

Preliminary studies showed that the best metal to trap hydrogen from lithium is yttrium. Yttrium shows greater affinity for hydrogen than lithium forming more stable hydrides and therefore, it is an excellent getter material for hydrogen impurities present in liquid lithium.

An experimental activity is under development for the study of Y as getter material and in parallel a 1:1 prototype of the DONES-ICS lithium loop (LITEC facility) is under construction for validation of the DONES ICS design. Experiments at laboratory-scale on deuterium-loaded yttrium samples and their characterization using Thermal Desorption Spectroscopy (TDS), X-ray Photoelectron Spectroscopy (XPS) and chemical analysis will be presented. Results will enable a better understanding of the behavior of yttrium as a getter material. Based on the results and taking into account the state of the art, a preliminary experimental plan to test the hydrogen trap in LITEC facility will be proposed in this paper. The role played by parameters such as the working temperature, the amount of getter or the flow rate will be analyzed in this experimental plan in order to find the most appropriate values to achieve the highest efficiency of the trap.

Keywords

Liquid metal, lithium, yttrium getter, hydrogen trap, DONES-ICS lithium loop, LITEC facility..

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PS4-113 ABSTRACT-5e37
H. Models and Experiments for FNT

Towards an improved version of Fluorescence Profile Monitor for IFMIF-DONES

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The International Fusion Materials Irradiation Facility - DEMO Oriented Neutron Source (IFMIF-DONES) has as main objective experimental tests of materials interesting for fusion reactors. The irradiation neutron source is based on a 125 mA, 9 MeV Continuous Wave (CW) deuteron beam impinging on a liquid lithium target. The beam is shaped along the accelerator to obtain the required profile at the target. Transversal beam profile has a crucial role in high-power accelerators, being mandatory to reduce beam losses and inhomogeneity, and ensuring the desired neutron field over the samples, hence the importance of beam profile diagnostics.

Several diagnostics monitor the beam. Some of them are interceptive, thus only useful during commissioning at low power operation. Others are non-interceptive and capable of uninterrupted operation at full power, providing beam lumped parameters such as current, position, losses and bunch length. Fluorescence Profile Monitors (FPM) are positioned as the main candidates for non-interceptive measurement of beam transversal profile in IFMIF-DONES.

However, there are limitations. Firstly radiation (neutrons and gammas) passing through photon sensor and optical components induces noise and damage. Another limitation is the reduced set of projections (in LIPAc prototype only two: horizontal and vertical), too few to reconstruct a 2D transversal profile.

To address the first limitation, an experimental campaign with radiation facilities available at CIEMAT was carried out. An FPM prototype was radiated with gammas at a ^{60}Co source called NAYADE, CIEMAT, Spain. Characterisation of unwanted fluorescence from optical components and interference in detector provides better knowledge of operational behaviour, to assess shielding requirements and sensor performance. For the 2D transversal profile limitation, alternative optical configurations were explored to allow basic tomographic techniques while keeping shielding constrains.

In this contribution, results of the experimental campaign will be reported and assessment for FPM diagnostics to operate in IFMIF-DONES will be discussed.

Keywords

IFMIF, FPM, Fluorescence profile monitor, LIPAc.

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PS4-114 ABSTRACT-d25b
H. Models and Experiments for FNT

Gas flow modelling of the IFMIF-DONES beamline

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IFMIF, the International Fusion Materials Irradiation Facility, is a test facility for materials foreseen in fusion reactors. High neutron fluxes are generated with an energy spectrum and intensity similar to the conditions at the burn phase inside a fusion reactor. The high energy neutrons result from accelerating deuterons onto a lithium target. IFMIF-DONES (DEMO-Oriented Neutron Source) is an IFMIF variant using just one of the two IFMIF 40 MeV accelerators.

Simulations of the molecular gas flow inside the full IFMIF-DONES beamline were performed with the Monte-Carlo simulator package code Molflow+[1]. The model is based on the design of the LIPAc (Linear IFMIF Prototype Accelerator), which is a 1:1 mockup of the IFMIF accelerator frontend (the injector, the RFQ and the SRF LINAC first cryomodule) being built and tested in Japan, and the new designs of the superconducting LINAC modules, the High Energy Beam Transport subsystem (HEBT and the liquid lithium target). Different simulation models for the beamline were prepared, which include different sets of boundary conditions for the pumping of deuterium, hydrogen and argon originating from beam losses, outgassing and gas injection near the target, respectively.

The simulation shows pressure profiles inside the beamline vacuum system, which are mainly determined by beam losses in the Low Energy Beam Transport line, the Radiofrequency Quadrupole, the Medium Energy Beam Transport line as well as by outgassing in the SRF LINAC modules and by injection of gas in the ion source, and in the HEBT from the target. This contribution will show the result of the gas simulations.

[1] M. Ady, R. Kersevan, Recent developments of Monte-Carlo codes MolFlow+ and SynRad+, 10th Int. Particle Accelerator Conf., Melbourne, Australia - doi:10.18429/JACoW-IPAC2019-TUPMP037

Keywords

IFMIF, DONES, TPMC, Vacuum.

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PS4-115 ABSTRACT-d368
H. Models and Experiments for FNT

Validation of Modular Cryogenics code (ModCryo) developed for lab-size test cryostats

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Cryogenics plays a vital role in fusion devices, and not only in superconductive magnet cooling, but also in the formation of cryogenic pellets that are used for Disruption Mitigation and fueling of fusion reactors. The development of efficient and reliable cryogenic systems is essential for the success of fusion research. The Centre for Energy Research has developed a Modular Cryogenics code (ModCryo) that helps in the design process of lab-sized test cryostats. The code is specifically designed to aid in the design of the Injector cryostat for the ITER DMS Shattered Pellet Injector Support Laboratory. First validation experiments on the cryostat of the Centre for Energy Research ITER DMS SPI Support Laboratory were performed. The experimental results were compared with the simulations obtained from the code. The validation demonstrated that the model could predict the gas flows and temperature distribution in the cryogenic system and provided a better understanding of the heat transfer mechanisms involved. The model's ability to handle complex components and network configurations enabled the identification of potential design improvements. The validation results indicate that the modular, object-oriented Python code is a reliable tool for extensive modelling of cryogenic laboratory size experimental systems and can aid in the design of heat exchangers, ensuring their optimal sizing for efficient operation.

Keywords

ModCryo, modular Python code, cryogenics, cryostat, validation, modelling.

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PS4-116 ABSTRACT-f3c8
H. Models and Experiments for FNT

Vacuum Thermal Contact Conductance determination between CucrZr and TZM electrodes coated by alumina and copper

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Thermal Contact conductance expresses the ability to transfer heat between the surfaces of two bodies in contact. In this work TCC is measured in vacuum conditions using an 1D setup at different contact pressure (3-20MPa) and using different materials in contact, simulating the Glow Discharge Cleaning permanent electrode of the Fusion Reactor ITER. For that, two samples have been prepared simulating the electrode and the heat sink (CuCrZr, TZM respectively) of the component. In order to electrically isolate the contact between the components, an alumina coating has been applied to the surface of the heat sink by thermal spraying (300 microns thickness). In the case of the electrode, two different conditions have been tested, coated by copper coating (10 microns thickness) deposited by magnetron sputtering and uncoated. The measurements show that the best result in terms of heat transferred between the electrode and the heat sink is obtained in the case of copper coated electrode, with TCC values around 5400W/m^{2}K.

Keywords

TCC, Surface, Electrodes, Thermal Contact Conductance,Coating, Magnetron Sputtering, Thermal Spraying.

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PS4-119 ABSTRACT-f682
H. Models and Experiments for FNT

Electromagnetic computer codes validation for ITER related analyses

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In this paper we describe and show the results of a new activity on the validation of numerical codes to be used for the low frequency electromagnetic analysis of ITER components. In order to be used for the design and the verification of protection important components (PIC) and for the development of protection important activities (PIA), computer codes (commercial and not) should demonstrate the accuracy of their numerical approximation. The aim of the activity is to propose a database of problems focused on induced currents in non linear materials and on hysteresis behaviours. For this purpose already existing benchmark problems are considered and new ones are developed for specific investigations. The TEAM problems 10 and 32 (*Testing Electromagnetic Analysis Methods*) from the COMPUMAG society have been tested, together with an ad hoc problem proposed by one of the authors. Four computer codes (EleFAnT3D, ANSYS APDL, ANSYS Maxwell, Comsol Multiphysics) have been used and results obtained have been compared. Comparison of results related to the TEAM problem 10 show a good agreement among all numerical results but a discrepancy with the experimental data. The TEAM 32 problem has been implemented only with the two codes allowing the simulation of hysteretic effects (ANSYS Maxwell, Comsol Multiphysics). Results of the benchmark show the limitation of the Jiles-Atherton Model implemented in COMSOL to capture all the effects represented with the experimental data. The last problem instead, related to the eddy current analysis in ferromagnetic structure, showed some limitation of the ANSYS APDL edge element. In the paper a complete and consistent description of the benchmark problems, and of the numerical codes used will be given together with a detailed description with the results obtained.

Keywords

Codes validation, benchmark, induced currents, non linear materials.

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PS4-120 ABSTRACT-623b
I. Repair and Maintenance

Logistics and Maintenance Applied to Building Plant System of Ifmif Dones

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DONES plant is defined to provide an accelerator-based D-Li neutron source to produce high-energy neutrons at sufficient intensity and irradiation volume to simulate as closely as possible the first wall neutron spectrum of future nuclear fusion reactions. For this purpose, a series of buildings and the necessary facilities have been designed to support the neutron source in achieving its objectives.

The maintenance of the facilities of the IFMIF DONES buildings is aimed at ensuring the proper functioning of the same in order to ensure the safety of its users. This is why a maintenance plan is being developed for the facilities, which includes: maintenance frequency, duration, necessary spare parts, tools to be used, staff involved, and other relevant information. All this will be reflected in Maintenance Matrix (MMX) that will serve as a basis for the aforementioned maintenance plan, once the facilities are completed.

Keywords

IFMIF-DONES, maintenance, facilities.

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PS4-121 ABSTRACT-9bba
I. Repair and Maintenance

The Remote Handling maintenance in IFMIF-DONES Project: status of the activities and future development program

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The International Fusion Materials Irradiation Facility - DEMO Oriented Neutron Source (IFMIF-DONES) is a facility which will be built in Granada (Spain). It is an accelerator-driven intense neutron source designed for the study and qualification of structural materials under severe irradiation condition of a neutron field having an energy spectrum similar to the one present in a fusion power reactor. IFMIF-DONES consists of complex systems and massive components that need to be assembled and maintained on site. For several of them it is required to perform maintenance, inspection and monitoring tasks over many years in a hostile environment and in efficient, safe and reliable manner. The maintenance of IFMIF-DONES systems and components, located mainly in the Test Systems, in the Lithium Systems and in the Accelerator Systems, is classified as a Remote Handling (RH) 1st Class activity. Over the last 5 years, much progress has been made in the definition of the maintenance activities to be performed for such a facility. The main achievements include: definition of the RH maintenance requirements; implementation of the maintenance strategy; classification of components from the maintenance point of view; and development of the maintenance procedures and the design of the Remote Handling System (RHS). This latter system comprises the whole set of Remote Handling equipment and tooling for the execution of the maintenance tasks. In addition, a wide experimental program is ongoing to validate the RH maintenance operations and the custom and special purpose devices used to implement them. In this paper, an overview of the present status of the IFMIF-DONES RHS design is given together with a description of the several validation activities, either underway or planned

in the next coming years, for the RH maintenance of various systems and components of the facility.

Keywords

IFMIF-DONES, Maintenance, Remote Handling Maintenance.

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PS4-123 ABSTRACT-c850
I. Repair and Maintenance

Kinematics analysis of a novel hybrid manipulator for optomechanical modules assembly in SG-Ⅲ

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A novel hybrid manipulator is proposed in this work, in order to assemble optomechanical modules in an inertial confinement fusion facility (SG-Ⅲ). Differ from general serial and parallel manipulators, both the inverse and forward kinematics problems of hybrid manipulators are difficult to be solved. Based on an analytic method, the position and orientation mapping functions between the optomechanical module and the parallel manipulators are established, and then the method of transformation matrix is employed to obtain the inverse kinematics solution. The forward kinematics of this hybrid manipulator is highly nonlinear and coupled, so a back propagation neural network using all class one network strategy is trained to approximate the mapping relation. Aiming to improve the accuracy, the network is split into two sub-networks corresponding to the natural property of the outputs. In the end, an improved back propagation neural network is proposed to optimize the network, results illustrate that the kinematic accuracy is increased significantly.

Keywords

Inertial confinement fusion, hybrid manipulator forward kinematics, inverse kinematic, neural networks

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PS4-124 ABSTRACT-ced0
I. Repair and Maintenance

Data Driven Modeling of Heavy-duty Joint System for DEMO Manipulators

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Remote maintenance (RM) plays critical role in the successful operation of future DEMO plant, where various robotic systems will be employed for various tasks, ranging from the in-vessel mover, pipe joint handling cask, vertical maintenance manipulator, multipurpose deployer to divertor cassette deployer etc. The design of heavy-duty robotic systems must deal with the heavy payload where the joint design is a critical task.

This study investigates the dynamic modeling of heavy-duty joint in a close loop system using the data driven method. The heavy duty joint often comprises high reduction planetary gear, cycloidal gear or harmonic gear or a compound system of those gears, which makes the transmission characteristics highly nonlinear and empirical in terms of force outputs and sensing, positioning accuracy, efficiency etc. Through studying the abundant data of input and output of joint system, this study utilized data process techniques, namely SVD (singular value decomposition) and PCA (principal component analysis) and filtering, and machine learning technics, namely deep neural networks, developed the surrogate model of the joint from the simulation, which can be used in the close loop position control system for overall system performance evaluation.

The obtained data driven model has been compared with the analytical model, and result shows the pretrained data driven model can be extrapolated into the new domain after few post-trainings. The close loop system integrated with the data driven model can recognize the force abnormality successfully in a deliberate collision case. Utilizing merely measurement data, the methods and results of this study would benefit dynamic modeling scenarios in the real DEMO RM manipulators which is hurdled by the complexity in deriving the analytical model of real systems.

Keywords

Dynamic modeling, robotics, artificial intelligence, remote maintenance.

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PS4-125 ABSTRACT-3717

I. Repair and Maintenance

Alignment strategy for IFMIF-DONES facility

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The IFMIF-DONES facility is based on a deuterón linear particle Accelerator, impinging on a liquid lithium target, generating neutrons for the irradiation of material samples of future fusión reactors. The facility includes equipment from different systems that require alignment. This equipment is located in three separate rooms. Due to the level of residual doses, two of these rooms, known as Targert Interface Room (TIR) and Test Cell (TC), cannot be accessed for hands-on maintenance and only access from the top of the room is granted. Hence a network of references for periodic alignment on the room above them shall be foreseen. Moreover, the radiation levels during beam operation in TIR and TC precludes the use of conventional spherically mounted retroreflectors that would be damaged by neutron flux during operation. The challenges of connecting networks of references in different rooms and the use of equipment compatible with the radiation conditions in some areas will be discussed. The deformations in some equipment due to relevant change in temperature and pressure during operation will also be addressed. Finally, the baseline strategy for alignment in IFMIF-DONES and the uncertainties associated will be presented.

Keywords

IFMIF-DONES, alignment, maintenance.

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PS4-126 ABSTRACT-db02
I. Repair and Maintenance

Development of Lower Manipulator System for Breeding Blanket Maintenance within Large Port-Based Tokamaks

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Large port-based tokamaks for energy production are expected to be developed around breeding blanket technology, the purpose of which is to breed Tritium, provide shielding to the vessel walls and carry heat transfer loops. These blankets are expected to be split into five segments, each weighing up to 60 tons and will require periodic replacement through the Upper Port for maintenance. During replacement, the blankets need to be guided through the vacuum vessel through highly restricted kinematic paths to avoid collisions. The Lower Manipulator System (LMS) is expected to operate through the Lower Port to guide the blankets and react to the severe loads arising from the offset between the blankets' Centre of Gravity (CoG) and the lifting point. Several mechanisms for the LMS have been developed to comprehensively explore blanket handling capabilities using the two-port approach. This paper presents the different LMS concepts and discusses the respective salient features.

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Keywords

RM, DEMO, Blanket handling.

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PS4-128 ABSTRACT-e992
I. Repair and Maintenance

Assessment and recovery from the damage of SPIDER RF driver Faraday shield lateral wall

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The SPIDER experiment, hosted in the Neutral Beam Test Facility of Padua (IT), is the full-scale prototype of the negative ion source for ITER neutral beam injectors. Inside the ion source, eight RF drivers are responsible for producing plasma via electromagnetic induction. The Faraday shield lateral wall (FSLW) is a cup-shaped component of SPIDER RF drivers made out of electrodeposited copper, protecting the surrounding components from direct contact with plasma. Electromagnetic coupling with plasma is not perfect inside the driver: RF power in the order of tens of kWs is deposited on the driver components during nominal operation instead, thus requiring them to be actively cooled. During SPIDER 2018-2021 experimental campaign, an unexpected issue with the cooling system resulted in overheating and structural failure of the FSLW of four drivers. Numerical analyses and measurement of the hardness of the copper surface highlight the localized nature of the damage, which is restricted to the cylindrical section of the FSLW. Refurbishment of the components is undertaken by removing the damage-affected material and restarting electrodeposition. Measurements of hardness, grain size and shape deformation are collected across the surface of the FSLWs to obtain a high-spatial-resolution map of the damage, from which spatial distribution of RF power deposition can be estimated. These provide new insight on possible improvements in the thermo-structural design of the faraday shield and represent useful inputs for studying the coupling with plasma in SPIDER drivers. This paper presents a summary of the analyses leading to the full comprehension of the status of the damaged parts and the route undertaken for the refurbishment.

Keywords

Neutral beam injectors – negative ion RF sources – RF driver – Faraday shield – refurbishment.

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PS4-129 ABSTRACT-c6b6
J. Burning Plasma Control and Operation

Bismuth Hall sensors for magnetic measurements in the HL-2A/2M tokamak

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Measurement errors will inevitably arise when using conventional inductive sensors to measure magnetic fields in future long-pulse fusion reactors. Metal Hall sensors, a new technology for steady-state magnetic field measurement, have been developed for resolving this problem.

Recently the design and manufacturing technology of metal Hall sensors has been developed in the HL-2A/M tokamak. Metal Hall sensors, involving three metals (bismuth, antimony, and copper), and four thicknesses (30nm, 50nm, 100nm, and 150nm), have been manufactured and tested. The test results show that the sensitivities of the sensors are consistent with theoretical expectations approximately, and the output voltages are proportional to the magnetic field. But when the sensitive layer thickness is lower than 100nm, the sensitivity shows a nonlinear relationship in the range of 0-100Gs. 100nm bismuth Hall sensor offers the largest sensitivity among them, but it is temperature-dependent.

Due to the excellent performance of the 100nm bismuth Hall sensor, 38 bismuth Hall sensors have been installed on the HL-2A tokamak for the radial magnetic field, B_{rad}, and poloidal magnetic field, B_{pol}, measurements. Each selected sample bismuth sensor is individually calibrated before installation, including the specific value of sensitivity K_H and stable calibration accuracy. The calibration accuracy test results show that the bismuth-based metal Hall sensor has good stability in the range of 1T magnetic field. When the magnetic field is larger than 30Gs, the calibration accuracy is better than ±1%. In addition, the identification of the advanced divertor configuration of the HL-2M tokamak for plasma control remains a prominent issue for further research and development.

Keywords

Metal Hall sensor, Bismuth, magnetic measurement, HL-2A, tokamak.

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PS4-130 ABSTRACT-dc58
J. Burning Plasma Control and Operation

Design and operation results of KSTAR 105/140GHz ECH system

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The Electron Cyclotron Heating (ECH) system at Korea Superconducting Tokamak Advanced Research (KSTAR) as one of important heating devices from the first plasma operation has been used in various researches such as plasma startup, wall cleaning, Neo-classical Tearing Mode (NTM) control, long pulse operation, and hybrid scenarios development. KSTAR has installed and operated four 105/140 GHz dual frequency ECH systems and has a plan to add two more 170 GHz ECH systems. The ECH wave used in the tokamak device is a highly localized, high-power wave in a specific polarization state, and must be transmitted from the high-power gyrotron to the antenna with minimal loss. If it is incident in an unintended state, not only the ECH/ECCD efficiency will drop even if the gyrotron can output the high power, but also it can cause some damage to the ECH system itself (gyrotron, transmission line, antenna, etc.), tokamak device, and other devices installed on tokamak. In order to prevent this, mode purity, beam alignment, polarization, antenna mirror control, etc. are very important factors. This paper briefly describes the KSTAR ECH system and its operation results, and discusses problems that occurred during ECH system operation and experimental methods to solve them.

Keywords

KSTAR, ECH, Gyrotron, TL, CW.

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PS4-133 ABSTRACT-f3f9
J. Burning Plasma Control and Operation

Time series methods for the control of fusion plasma instabilities and disruption prediction

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Tokamak plasmas are the site of various instabilities which develop naturally and so far unavoidably. Regions of small perturbations may grow rapidly, leading to a rapid loss of the stored thermal and magnetic energy, which in the end may cause harmful effects on the plasma facing components. Disruptions may be accompanied by the generation of runaway electrons, which could also lead to severe accidents such as the melting of the first wall or leaks in water cooling circuits. Various pacing techniques have been developed and tested on several devices, for the mitigation of ELM and sawtooth instabilities. These techniques are based on the use of external perturbations, aiming to reduce the instability amplitude by triggering the instability at a higher rate than its natural frequency. Since these instabilities of the "integrate and fire" type, this approach meant to lower the energy lost in each crash. However various details of the pacing processes remain poorly understood, in particular the assessment of the relative contribution of triggering phase and amplitude. We report a data analysis methodology based on the wavelet decomposition of the signals and the following implementation of information theoretic indicators, which has been developed to determine the influence of amplitude/phase components for an efficient pacing scheme. A series of time series analysis methods, including recurrence plots, monitoring the evolution of the embedding dimension, and the identification of the transition to a chaotic regime have been applied for the determination of the time moment when significant changes in the plasma dynamics occur, indicating the onset of drifts towards the plasma disruptive regions of the operational space. These methods may help the construction of significantly more appropriate training sets for various kinds of disruption predictors. Some of these methods presented may also be implemented as stand-alone disruption predictors for real time deployment.

Keywords

Time series analysis, instability pacing, disruption prediction, fusion plasma.

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PS4-134 ABSTRACT-7d26

L. Fission-Fusion Synergy and Cross Cutting Technologies

Tokamaks as Convenient Neutron Sources for Nuclear Materials Testing

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Fusion needs the development of radiation resistant and low activation materials. An experimental device for testing radiation resistance is deemed necessary by many studies on fusion power demonstration. Plasma-facing materials, in fact, could undergo a fast-neutrons high fluence, with annual doses in the range of 20 dpa (displacement per atoms), and total fluences of about 200 dpa. Testing of candidate materials, therefore, requires a reliable high-flux source of high energy neutrons.

High-output neutron sources (NS) are required, besides nuclear materials testing, also in fundamental science and innovative technologies. To date, the most powerful NS are fission reactors and charged particle accelerators. Due to the high cost and engineering problems, fission reactor and accelerator NS may not surpass 10^{18} n/s. Fusion is neutron-rich and may become the most intensive neutron source. A tokamak neutron source could be designed and built soon, extrapolating present designs of fusion tokamaks, paying attention to some additional R&D, such as emphasize quasi-steady state or at least longer operation. Compact high-field tokamaks can be a candidate for being a neutron source in a materials testing device, essentially due to their design characteristics, such as compact dimensions, high magnetic field, high production of neutrons, flexibility of operation, etc.

This study addresses the development of a tokamak neutron source for a materials testing facility using compact high-field tokamaks. The ARC-SPARC design family is the starting point: revision of its operating parameters in order to act a as a suitable neutron source in a materials testing device are discussed and new operating scenarios are proposed.

The simulations show the potential of this neutron-rich device for fusion materials testing. Some full-power months of operation are sufficient to obtain relevant radiation damage values in terms of dpa for the tested materials.

Keywords

Neutron source, tokamak, SPARC-ARC, materials testing.

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L. Fission-Fusion Synergy and Cross Cutting Technologies

Numerical Prediction of the Physical Properties of Beryllium and Long-Lived Fission Products Fluoride Mixtures for Transmutation Using Fusion Neutrons Sources

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The disposal of high-level radioactive waste is one of the most important issues in nuclear energy use. Long-lived fission products (LLFP) are factors in high-level radioactive waste's radioactivity and decay heat. Earlier studies have proposed the transmutation of LLFP into short-lived nuclides by using fusion reactors. Fusion reactors are the ideal systems for transmutation because they can provide high energy and monochromatic neutrons. For actual transmutation in fusion reactors, it is practical to provide space only for transmutation in a part of a blanket system, where tritium is not generated.

Based on the background above, we propose new molten salts—beryllium fluoride and LLFP mixtures—as the transmutation targets. Our previous study has performed neutron transport and burn-up simulations and indicated that LLFPs in the molten salts are sufficiently transmuted by fusion neutron sources. Whereas the molten salts seem to be promising for transmutation, there are no reports of physical properties. It is necessary to establish an evaluation method that can predict the properties for discussing the feasibility of system design using the molten salt targets.

This study evaluates the physical properties of molten salts by using numerical simulation. More specifically, we perform classical molecular dynamics (MD) simulation to predict the properties—density, specific heat, viscosity, and thermal conductivity. Results have indicated that the viscosity increases according to the ratio of BeF_2, and the thermal conductivity declines in a system rich in LLFP fluorides. We further conduct machine learning MD simulations to extend the applicability to various LLFP compounds.

Keywords

Molten salt, partitioning and transmutation, high level radioactive waste, molecular dynamics, density functional theory, machine learning.

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